

**NUCLEAR REGULATORY COMMISSION
ISSUANCES**

**OPINIONS AND DECISIONS OF THE
NUCLEAR REGULATORY COMMISSION
WITH SELECTED ORDERS**

January 1, 1997 – June 30, 1997

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PREFACE

This is the forty-fifth volume of issuances (1 - 495) of the Nuclear Regulatory Commission and its Atomic Safety and Licensing Boards, Administrative Law Judges, and Office Directors. It covers the period from January 1, 1997 to June 30, 1997.

Atomic Safety and Licensing Boards are authorized by Section 191 of the Atomic Energy Act of 1954. These Boards, comprised of three members conduct adjudicatory hearings on applications to construct and operate nuclear power plants and related facilities and issue initial decisions which, subject to internal review and appellate procedures, become the final Commission action with respect to those applications. Boards are drawn from the Atomic Safety and Licensing Board Panel, comprised of lawyers, nuclear physicists and engineers, environmentalists, chemists, and economists. The Atomic Energy Commission first established Licensing Boards in 1962 and the Panel in 1967.

Beginning in 1969, the Atomic Energy Commission authorized Atomic Safety and Licensing Appeal Boards to exercise the authority and perform the review functions which would otherwise have been exercised and performed by the Commission in facility licensing proceedings. In 1972, that Commission created an Appeal Panel, from which are drawn the Appeal Boards assigned to each licensing proceeding. The functions performed by both Appeal Boards and Licensing Boards were transferred to the Nuclear Regulatory Commission by the Energy Reorganization Act of 1974. Appeal Boards represent the final level in the administrative adjudicatory process to which parties may appeal. Parties, however, are permitted to seek discretionary Commission review of certain board rulings. The Commission also may decide to review, on its own motion, various decisions or actions of Appeal Boards.

On June 29, 1990, however, the Commission voted to abolish the Atomic Safety and Licensing Appeal Panel, and the Panel ceased to exist as of June 30, 1991. In the future, the Commission itself will review Licensing Board and other adjudicatory decisions, as a matter of discretion. *See* 56 Fed. 29 & 403 (1991).

The Commission also has Administrative Law Judges appointed pursuant to the Administrative Procedure Act, who preside over proceedings as directed by the Commission.

The hardbound edition of the Nuclear Regulatory Commission Issuances is a final compilation of the monthly issuances. It includes all of the legal precedents for the agency within a six-month period. Any opinions, decisions, denials, memoranda and orders of the Commission inadvertently omitted from the monthly softbounds and any corrections submitted by the NRC legal staff to the printed softbound issuances are contained in the hardbound edition. Cross references in the text and indexes are to the NRCI page numbers which are the same as the page numbers in this publication.

Issuances are referred to as follows: Commission--CLI, Atomic Safety and Licensing Boards--LBP, Administrative Law Judges--ALJ, Directors' Decisions--DD, and 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 to have any independent legal significance.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 40-8027-EA

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

January 22, 1997

The Commission grants two petitions for review challenging the Licensing Board's approval of a settlement agreement. The Commission also establishes a briefing schedule.

MEMORANDUM AND ORDER

The State of Oklahoma, Native Americans for a Clean Environment, and the Cherokee Nation have filed petitions for Commission review of the Atomic Safety and Licensing Board's Memorandum and Order, LBP-96-24, 44 NRC 249 (1996), in which a majority of the Board approved a settlement agreement between the NRC Staff and General Atomics (GA) in this proceeding. In a dissenting opinion, Judge Bollwerk raised questions that, in his view, merited further inquiry. The NRC Staff and GA oppose Commission review. In accordance with the considerations set forth in 10 C.F.R. § 2.786(b)(4), the Commission has decided that review of LBP-96-24 is appropriate.

Pursuant to 10 C.F.R. § 2.786(d), the Commission sets the following briefing schedule:

1. Intervenor and the State shall file their briefs within 21 calendar days after service of this Order. Their briefs shall not exceed thirty pages each.
2. The Staff and GA may file responsive briefs within 21 calendar days after service of the Petitioners' brief. Their responses shall not exceed thirty pages each.
3. Within 10 calendar days after service of the responsive briefs, Intervenor and the State may file reply briefs. Their replies shall not exceed ten pages each.

The parties' briefs should address (1) what the role of the Board should be in reviewing settlements; (2) what factors the Board should consider when applying the "public interest" standard governing review of settlements (*see Sequoyah Fuels Corp. (Gore, Oklahoma Site), CLI-94-12, 40 NRC 64, 71 (1994)*); (3) the arguments set forth in the petitions for review; and (4) the questions raised by Judge Bollwerk. Briefs exceeding ten pages must contain a table of contents, with page references, and a table of cases (alphabetically arranged), statutes, regulations, and other authorities cited, with references to the pages of the brief where they are cited. Page limitations on briefs are exclusive of pages containing a table of contents, table of cases, and of any addendum containing statutes, rules, regulations, etc.

IT IS SO ORDERED.

For the Commission¹

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 22d day of January 1997.

¹ Commissioner Diaz was not available for the affirmation of this Order. If he had been present, he would have approved the Order.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 70-3070-ML

LOUISIANA ENERGY SERVICES, L.P.
(Calborne Enrichment Center)

January 29, 1997

The Commission denies a motion, filed by the Intervenor, requesting partial reconsideration of CLI-96-8, 44 NRC 107 (1996). In CLI-96-8, the Commission granted in part and denied in part the Intervenor's petition for review of Atomic Safety and Licensing Board Initial Decision LBP-96-7, 43 NRC 142 (1996), which resolved all contentions on emergency planning in the Applicant's favor.

RULES OF PRACTICE: RECONSIDERATION MOTIONS

Motions for reconsideration may not rest on a new thesis that could have been raised earlier in a petition for review.

RULES OF PRACTICE: RECONSIDERATION MOTIONS

NRC rules contemplate petitions for reconsideration of a Commission decision on the merits, not petitions for reconsideration of a Commission decision to decline review of an issue. *See* 10 C.F.R. § 2.786(e).

ORDER

The Intervenor, Citizens Against Nuclear Trash (CANT), has filed before the Commission a Motion for Partial Reconsideration of CLI-96-8, 44 NRC 107 (1996). Both the NRC Staff and the Applicant, Louisiana Energy Services (LES), oppose the Intervenor's motion. For the reasons stated in this Order, we deny the motion.

In CLI-96-8, the Commission granted in part and denied in part CANT's petition for review of Atomic Safety and Licensing Board Initial Decision LBP-96-7, 43 NRC 142 (1996). The Licensing Board's decision resolved all contentions on emergency planning in favor of LES. The Commission in CLI-96-8 granted review of only one issue raised in CANT's petition for review: whether the Licensing Board erred in directing the NRC Staff to clarify the intended role of the Applicant's onsite fire brigade. 44 NRC at 108. The Commission went on to hold, based on the pleadings and record before it, that the emergency plan description of the onsite brigade's size and training meets Commission requirements. 44 NRC at 110.

We deny CANT's motion for partial reconsideration for three independent reasons:

First, motions for reconsideration may not rest on a "new thesis."¹ Both LES and the NRC Staff argue that CANT is now raising for the first time before the Commission the issue of the qualifications and training of the offsite fire department, an issue they say that CANT failed to raise in its Petition for Review of LBP-96-7. We agree with the NRC Staff and LES. While CANT's petition for review contained references to the offsite fire department (at 2, 5), the petition failed to articulate any explicit challenge to the Board's findings on the department's training and qualifications. A "cursory assertion" is insufficient to raise an issue for appeal. *See Yankee Atomic Electric Co. (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 272 (1996).*²

Secondly, even if CANT had intended in its petition for review to raise the offsite fire department question, the Commission in CLI-96-8 explicitly denied review of the Licensing Board's decision, "except for a single issue," involving "the intended role and training of the Applicant's onsite fire brigade." 44 NRC at 108. CANT, in requesting reconsideration of CLI-96-8, does not challenge the Commission's findings on the role and training of the onsite fire brigade. CANT

¹ See, e.g., *Central Electric Power Cooperative (Virgil C. Summer Nuclear Station, Unit 1), CLI-81-26, 14 NRC 787, 790 (1981)*, quoting *Tennessee Valley Authority (Hartsville Nuclear Plant, Units 1A, 2A, 1B, and 2B), ALAB-418, 6 NRC 1, 2 (1977)*.

² The NRC Staff and LES also claim that the motion for reconsideration was filed 2 days late, a point disputed by CANT. We do not decide this timing issue. Even taking into account all of CANT's arguments in its motion for reconsideration and its reply brief, and assuming *arguendo* that the motion is timely, we find no reason to grant it.

instead requests the Commission to review the qualifications and training of the *offsite* fire department, a subject the Commission did not accept for review. Our rules contemplate petitions for reconsideration of a Commission decision on the merits, not petitions for reconsideration of a Commission decision to decline review of an issue. *See* 10 C.F.R. § 2.786(e).

Third, CANT's motion for reconsideration simply does not raise any compelling argument calling into question the Licensing Board's findings on the training and qualifications of the offsite fire department. The Board made several findings about the training and qualifications of the offsite fire department. *See* LBP-96-7, 43 NRC at 151-52, 158, 159-61, 164-65. It found the information outlined in the LES plan adequate under the "brief description" requirement of NRC rules (10 C.F.R. §§ 40.31(j)(3)(i), 70.22(i)(3)(i)); it properly based its findings on information contained in the record and found in either expert testimony or the LES emergency plan itself.

We are not persuaded by CANT's argument that the Licensing Board was unaware that the offsite fire department would be ultimately responsible for fighting a severe onsite fire at the LES facility. The Board assumed from the SAR and the SER that the offsite fire department would be the primary organization responsible for controlling fires at the plant, and that the onsite fire brigade would merely "supplement" but not replace the local fire department. 43 NRC at 161. In referring the onsite brigade issue to the Staff, the Board sought merely to confirm that the *onsite* brigade would not have a bigger firefighting role than the Board had found reflected in the SAR and SER, and accordingly to ensure that the brigade did not need additional training or members. 43 NRC at 160-61. The Board expressed no concerns about the adequacy of the offsite fire department.

The Intervenor's Motion for Partial Reconsideration of CLI-96-8 is *denied*. It is so ORDERED.

For the Commission

JOHN C. HOYLE
Secretary of the Commission

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Dr. Charles N. Kelber
Dr. Peter S. Lam

In the Matter of

Docket No. 50-219-OLA
(ASLBP No. 96-717-02-OLA)

GENERAL PUBLIC UTILITIES
NUCLEAR CORPORATION
(Oyster Creek Nuclear Generating
Station)

January 31, 1997

In this proceeding concerning challenges by Intervenors Nuclear Information Resource Service (NIRS) and the Oyster Creek Nuclear Watch (OCNW) to a technical specification change regarding heavy load handling over the Oyster Creek Nuclear Generating Station spent fuel pool, the Licensing Board grants summary disposition in favor of Licensee General Public Utilities Nuclear Corporation (GPUN) on the sole intervenor contention, ruling that (1) prior to the requested revision, the technical specification did preclude the heavy load activity now at issue; (2) as they embody the agency's "defense-in-depth" philosophy, the provisions of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (July 1980), which Intervenors assert preclude authorizing the requested technical specification change, establish guidance rather than regulatory requirements for handling heavy loads; and (3) nothing in the provisions of NUREG-0612 and later NRC Staff generic letters intended to promote compliance with that document's recommendations bars the adoption of the requested technical specification change.

LICENSE: CONSTRUCTION OF TERMS (PLAIN MEANING)

The first interpretational tool for discerning the meaning of the terms of a license is the plain meaning of the language of the provision in question.

LICENSE: CONSTRUCTION OF TERMS (SUBSEQUENT REVISION)

A subsequent enactment that declares the intent of an earlier provision generally is to be given “great weight” in resolving a construction problem. *See Red Lion Broadcasting v. FCC*, 395 U.S. 367, 380-81 (1969); *cf.* 17A Am. Jur. 2d *Contracts* § 388, at 415-16 (1991) (when contract terms are ambiguous and parties have made other contracts concerning the same subject matter, those instruments can be examined together to aid in interpretation). The relevance of such a subsequent enactment seems particularly telling when the parties who drafted and approved the revision declare it was intended to clarify any ambiguity in the prior version.

LICENSE: CONSTRUCTION OF TERMS (“EXCEPT”)

In a technical specification paragraph that sets forth a general prohibition, the use of the term “except” to describe a specific activity sanctioned in a subsequent paragraph establishes that, but for its specification as an exception, that activity would be covered by the general prohibition.

REGULATORY GUIDES: APPLICATION; STATUS

A Staff report bearing the NUREG designation does not fall into the category of a regulatory “requirement,” such as a statute, regulation, license condition, or order. *See Curators of the University of Missouri*, CLI-95-1, 41 NRC 71, 98 (1995). Instead, at best, “it serves as guidance, setting forth but one method for meeting the applicable regulatory requirements In other words, that document ‘is treated simply as evidence of a legitimate means for complying with regulatory requirements.’” *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant), ALAB-852, 24 NRC 532, 544-45 (1986) (quoting *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), ALAB-698, 16 NRC 1290, 1298-99 (1982), *aff’d in part on other grounds*, CLI-83-22, 18 NRC 299 (1983)).

GENERIC COMMUNICATIONS: APPLICATION; STATUS

In a generic letter that both “requested” that licensees take various actions and “required” that licensees provide a report detailing their compliance efforts, in contrast to the reporting component of a generic letter, which seemingly would constitute a “requirement,” *see* 10 C.F.R. §§ 2.204, 50.54(f), the generic letter’s compliance request would not constitute a “requirement” in the absence of some additional regulatory directive such as an order or a regulation mandating compliance. *Cf.* 60 Fed. Reg. 34,381, 34,392 (1995) (agency expects licensees to adhere to commitments resulting from administrative actions such as confirmatory action letters and will issue appropriate orders to ensure commitments are met), *reprinted in* Office of Enforcement, U.S. Nuclear Regulatory Commission, NUREG-1600, “General Statement of Policy and Procedures for NRC Enforcement Actions” at 14 (July 1995).

ATOMIC ENERGY ACT: LICENSE AMENDMENTS

LICENSE: AMENDMENT

OPERATING LICENSE(S): TECHNICAL SPECIFICATIONS (AMENDMENT)

OPERATING LICENSE AMENDMENT HEARING: ISSUES FOR CONSIDERATION

A technical specification that is not subject to revision would not be the norm. By providing in section 187 of the Atomic Energy Act that agency-issued licenses are “subject to amendment,” 42 U.S.C. § 2237; *see also, e.g.*, 10 C.F.R. § 50.90, the Congress contemplated that any license provision could be changed, at least so long as the revision sought was not inimical to the public health and safety or the common defense and security. Consequently, in the absence of language in the license (or some other regulatory requirement) that makes manifest a license provision’s immutability, the question in a license amendment proceeding generally is whether the requested change is consistent with applicable agency regulatory strictures and any suitable guidance.

MEMORANDUM AND ORDER **(Ruling on Summary Disposition Motion)**

Pending before the Licensing Board is a motion filed by Licensee General Public Utilities Nuclear Corporation (GPUN) requesting that summary disposition be entered in its favor on the sole contention at issue in this proceeding.

This contention, which is sponsored by pro se Intervenor Nuclear Information and Resource Service (NIRS) and the Oyster Creek Nuclear Watch (OCNW), poses a single legal issue that can be summarized as follows:

Whether a technical specification revision for GPUN's Oyster Creek Nuclear Generating Station (OCNGS) permitting a dry shielded canister (DSC) shield plug to be moved over irradiated fuel in a DSC as a prerequisite to sealing and removing the DSC from the OCNGS spent fuel pool for transport to an onsite independent spent fuel storage installation (ISFSI) is foreclosed under the terms of a 1980 NRC staff report, Office of Nuclear Reactor Regulation (NRR), U.S. Nuclear Regulatory Commission (NRC), NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," (July 1980) [hereinafter NUREG-0612], as it embodies the agency's "defense-in-depth" risk management precepts.

The NRC Staff supports the Licensee's motion, while Intervenor NIRS and OCNW oppose it.

For the reasons set forth below, we grant the Licensee's summary disposition motion, finding that (1) the "heavy load" limitation in OCNGS Technical Specification 5.3.1.B encompasses a shield plug movement over irradiated fuel in a DSC; (2) as it embodies the agency's defense-in-depth philosophy, NUREG-0612 provides guidance rather than requirements regarding the control of heavy loads at nuclear power plants; and (3) nothing in this NUREG-0612 guidance precludes the adoption of the requested OCNGS technical specification change.

I. BACKGROUND

As we outlined in our October 25, 1996 ruling admitting Intervenor NIRS and OCNW and their legal contention into this proceeding,¹ *see* LBP-96-23, 44 NRC 143, 147-56 (1996), the license amendment at issue here involves a change in OCNGS Technical Specification 5.3.1.B. When this proceeding began in June 1996, and through early November 1996, that provision stated "[I]oads greater than [the] weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility." NRC Staff Response in Opposition to Request for Hearing and Petition to Intervene of [NIRS/OCNW/Citizens Awareness Network (CAN)] (June 26, 1996) unnumbered attach. 2 (OCNGS Technical Specification, p. 5.3-1 (Apr. 10, 1995)). On November 7, 1996,

¹In that memorandum and order, we also concluded that although a third petitioner, the Citizens Awareness Network (CAN), had failed to establish its standing to intervene either as of right or as a matter of discretion, we would permit CAN to participate as an *amicus curiae* if it wished to do so. *See* LBP-96-23, 44 NRC at 159-61. We then established a deadline for CAN to advise the Board and the other parties that it wanted to participate as an *amicus*. *See id.* at 161 n.13. CAN, however, has neither appealed this ruling to the Commission nor shown any further interest in participating in this proceeding before the Board.

pursuant to a Staff “no significant hazards consideration” finding,² that provision was revised so that it now reads:

- B. 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.
2. The shield plug and the associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

Letter from Ann P. Hodgdon, NRC Staff Counsel, to Licensing Board (Nov. 12, 1996) encl., at encl. 1, attach. at unnumbered p. 2 (OCNGS Technical Specification p. 5.3-1, Amendment No. 187) [hereinafter Amended Technical Specification 5.3.1.B].

GPUN proposed this change to facilitate the off-loading of spent fuel from the OCNGS spent fuel pool into dry cask storage in the OCNGS ISFSI. As we described in some detail in our earlier opinion, *see* LBP-96-23, 44 NRC at 148-50, while submerged in one corner of the spent fuel pool within the confines of a GPUN-developed cask drop protection system (CDPS) and a 60-ton onsite transfer cask (TC), the 14-ton DSC is loaded with up to fifty-two spent fuel assemblies, each weighing approximately 800 pounds. To close the DSC before removing it and the accompanying TC from the fuel pool in preparation for transport to the OCNGS ISFSI, a 4-ton shield plug attached to a crane by a 3-ton yoke is moved over the DSC and the fuel assemblies it contains and then lowered into place atop the DSC. The technical specification amendment at issue in this proceeding explicitly allows the shield plug — which weighs many times more than a fuel assembly — to be moved over the fuel assemblies in the DSC while those assemblies and the DSC are in the CDPS in the corner of the spent fuel pool.

In LBP-96-23, 44 NRC at 156-66, we found that in challenging the GPUN technical specification change, in accordance with the requirements of 10 C.F.R. § 2.714(a), (b)(2), NIRS and OCNW had both established their standing to intervene and jointly put forth a single litigable contention concerning that amendment. Their sole contention states:

² In its initial notice of opportunity for hearing regarding the Licensee’s amendment request, the Staff advised that it proposed to find the change involved “no significant hazards consideration.” *See* 61 Fed. Reg. 20,842, 20,848 (1996). Under Atomic Energy Act (AEA) section 189a(1)(A), (2)(A), 42 U.S.C. § 2239(a)(1)(A), (2)(a), and the implementing regulations in 10 C.F.R. §§ 50.91-92, upon making such a finding the Staff can issue an amendment notwithstanding the pendency of a hearing request challenging the proposed license change. On November 7, 1996, based on its conclusion the GPUN proposed technical specification revision involved “no significant hazards consideration,” the Staff issued the technical specification amendment effective immediately. *See* 61 Fed. Reg. 66,702, 66,720 (1996).

The GPUN application fails to provide defense-in-depth against the risks of a heavy load drop onto irradiated fuel and fails to satisfy NRC regulatory guidance as provided in NUREG-0612 "Control of Heavy Loads At Nuclear Power Plants" pertaining to defense-in-depth risk management to assure that a heavy load drop does not impact or encroach on irradiated fuel.

Supplemental Petition of [NIRS/OCNW/CAN] (July 18, 1996) at 2. Further, although the Intervenor put forth several bases in support of this contention, we determined only one was adequate to support its admission, which we summarized as follows:

The NRC's fundamental regulatory defense-in-depth principle is implemented in NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," which is the equivalent of a regulatory guide. Because OCNCS does not employ a single failure proof crane for shield plug movement, consistent with NUREG-0612 guidelines as described in enclosure 1 to NRC Generic Letter 85-11 (June 28, 1985), GPUN must rely on analyzed safe load paths and restricted load limits for movement of heavy loads "to assure, to the extent practical" that heavy loads are not carried over or near irradiated fuel. Although GPUN claims in its safety evaluation regarding the proposed technical specification change that a shield plug drop accident is not credible because of GPUN administrative controls (e.g., rail stops), operator training, and inspections concerning dry-storage related spent fuel movements, this does not adequately address human error or mechanical/electrical failure issues. Rather, the most effective way to avoid such failures is to restrict both human-directed activity and prohibit the movement of heavy loads as is done with current Technical Specification 5.3.1.B. As such, consistent with the agency's NUREG-0612 defense-in-depth guidance, the existing provision cannot be revised as the Licensee has requested.

LBP-96-23, 44 NRC at 151-52.

In considering the admissibility of the Intervenor's contention, we observed that the contention and this supporting basis are premised on the Intervenor's assertions that (1) NUREG-0612 provides binding regulatory guidance for implementing the agency's overall defense-in-depth principle in the context of heavy load control; and (2) the then-existing technical specification with its one fuel assembly heavy load limit cannot be changed consistent with NUREG-0612 because that limit is a vital control necessary for compliance with the defense-in-depth principle underlying NUREG-0612. Although recognizing GPUN and Staff assertions that NUREG-0612 is not a regulatory requirements document and declares only that moving heavy loads over or near irradiated fuel should be avoided "to the extent practical," we nonetheless found two factors established a dispute regarding the technical specification change that warranted further inquiry. The first was the apparent adoption of the then-existing GPUN technical specification with its absolute single fuel assembly load limit after the publication of NUREG-0612 with its "to the extent practical" language. The second concerned various statements in Licensee and Staff documents regarding NUREG-0612 "requirements." See *id.* at 165-66. We also concluded this contention apparently presented a legal issue so that summary disposition

provided the appropriate procedural avenue for seeking to resolve its merits in the first instance. We thus established a schedule for dispositive motions and responses by the parties. *See id.* at 166-67.

In a November 15, 1996 motion, which is accompanied by a statement of material facts not in dispute and the supporting affidavit of GPUN Licensing and Regulatory Affairs Director John C. Fornicola, Licensee GPUN seeks summary disposition in its favor on this contention. *See* Licensee's Motion for Summary Disposition (Nov. 15, 1996) [hereinafter GPUN Dispositive Motion]; Licensee's Statement of Material Facts as to Which There Is No Genuine Dispute (Nov. 15, 1996) [hereinafter GPUN Material Facts Statement]; Affidavit of John C. Fornicola (Nov. 15, 1996) [hereinafter Fornicola Affidavit]. In a December 6, 1996 response, which includes the supporting affidavits of NRC Senior Project Manager Ronald B. Eaton and NRC Senior Reactor Engineer Harold Walker, the Staff agrees that GPUN's summary disposition request should be granted. *See* NRC Staff Response in Support of Licensee's Motion for Summary Disposition (Dec. 6, 1996) [hereinafter Staff Response]. On the same date, Intervenor NIRS and OCNW filed a response opposing GPUN's summary disposition request, albeit without any supporting affidavits. *See* Petitioner[s'] Opposition to GPUN Motion for Summary Disposition (Dec. 6, 1996) [hereinafter NIRS/OCNW Response]. Thereafter, in accordance with the pleading schedule we established, on December 20, 1996, GPUN filed a reply to the Intervenor's response. *See* Licensee's Reply to Petitioners' Opposition to Motion for Summary Disposition (Dec. 20, 1996) [hereinafter GPUN Reply].³

II. ANALYSIS

A. Standards Governing Summary Disposition

Under Rule 56(c) of the Federal Rules of Civil Procedure, a party is entitled to seek summary judgment in its favor on the merits of any claim for which "there is no genuine issue as to any material fact." The Commission's administrative counterpart to this judicial rule is found in 10 C.F.R. § 2.749(d), which provides in pertinent part:

The presiding officer shall render the decision sought if the filings in the proceeding, depositions, answers to interrogatories, and admissions on file, together with the statements of the parties and the affidavits, if any, show that there is no genuine issue as to any material fact and that the moving party is entitled to a decision as a matter of law.

³ By memorandum issued January 3, 1997, we advised the parties we had decided not to hold an oral argument on GPUN's dispositive motion. *See* Licensing Board Memorandum (Oral Argument on Dispositive Motion) (Jan. 3, 1997) at 1-2.

A number of the central procedural requirements governing the summary disposition process were recently summarized as follows:

The party filing the summary disposition motion has the burden of demonstrating the absence of any genuine issue of material fact. In this regard, [10 C.F.R. §] 2.749(a) requires that the moving party include a statement of material facts about which there is no genuine issue to be heard. In contrast, the opposing party must append to its response a statement of material facts about which there exists a genuine issue to be heard. If the responding party does not adequately controvert material facts set forth in the motion, the party faces the possibility that those facts may be deemed admitted. If, however, the evidence before the Board does not establish the absence of a genuine issue of material fact, then the motion must be denied even if there is no opposing evidence. Nevertheless, a party opposing a motion cannot rely on a simple denial of the movant's material facts, but must set forth specific facts showing there is a genuine issue of material fact.

Yankee Atomic Electric Co. (Yankee Nuclear Power Station), LBP-96-18, 44 NRC 86, 92-93 (citations omitted), *petition for review denied*, CLI-96-9, 44 NRC 112 (1996).

B. The Parties' Positions

1. GPUN's Arguments

In seeking summary disposition, GPUN declares that the two factors identified by the Board as potential support for the Intervenor's position that OCNCS Technical Specification 5.3.1.B cannot be changed in fact provide no justification for that claim. The Licensee asserts that while, as the Board observed, NUREG-0612 does indicate that in 1980 OCNCS did not have a technical specification governing the movement of heavy loads over spent fuel, NUREG-0612 was incorrect. According to GPUN, Technical Specification 5.3.1.B was adopted initially in 1977, some 3 years before NUREG-0612 was issued. *See* GPUN Dispositive Motion at 19 & n.13; GPUN Material Facts Statement at 1-2. As a result, GPUN concludes that any concern the language of Technical Specification 5.3.1.B prior to its recent amendment was reflective of a Licensee/Staff judgment regarding the application of defense-in-depth principles is misplaced.

As to the second concern about the language of various Licensee and Staff documents referring to NUREG-0612 "requirements," GPUN cites agency authority and language in NUREG-0612 it asserts establishes that a NUREG document, like a Staff regulatory guide, merely serves as guidance and cannot prescribe requirements. *See* GPUN Dispositive Motion at 8; GPUN Material Facts Statement at 1. GPUN further declares that while the Staff requested in two Staff generic letters that licensees conform to certain NUREG-0612 recommendations, the NUREG-0612 recommendation that licensees adopt a technical specification like OCNCS Technical Specification 5.3.1.B to govern

heavy load handling was not among them. See GPUN Dispositive Motion at 9-10; GPUN Material Facts Statement at 1.

In support of its summary disposition request, GPUN also claims that the Intervenor's position is legally untenable because Technical Specification 5.3.1.B "only applies to heavy loads moved over stored fuel in the spent fuel storage racks and is no legal impediment to the movement of heavy loads over spent fuel in the CDPS." GPUN Dispositive Motion at 2. According to GPUN, by "wording and intent" Technical Specification 5.3.1.B has always applied only to "stored" spent fuel, which does not include fuel assemblies placed in the CDPS prior to being removed from the spent fuel pool. *Id.* at 11. GPUN asserts that it requested the amendment at issue "at the suggestion of the NRC staff and out of an abundance of caution, only to make this meaning more explicit." *Id.* GPUN argues that various factors support this interpretation including (1) the use of the terms "stored" and "storage" in Technical Specification 5.3.1.B prior to its recent amendment; (2) a purported Staff/Licensee understanding about this meaning under Technical Specification 5.3.1.B that permitted GPUN in the mid-1980s to place a "heavy load" lid over fuel assemblies while loading and unloading a transportation cask in the CDPS as the cask was being sent to and later returned from a reprocessing facility; (3) a Staff interpretation of a similar technical specification at the Palisades Nuclear Plant; (4) language in the Safety Evaluation issued by the Staff in support of the November 7, 1996 "no significant hazards consideration" amendment; (5) regulatory history relative to the OCNCS spent fuel pool indicating there was a clear differentiation between the spent fuel pool and the CDPS; and (6) the language of and the interpretation accorded the agency's standard technical specification (Standard Technical Specification 3.9.6.2) regarding heavy load handling at boiling water reactors (BWRs). See *id.* at 12-22.

Finally, GPUN asserts that interpreting Technical Specification 5.3.1.B and NUREG-0612 in the manner suggested by the Intervenor is untenable because this would lead to an "absurd" result. To read these two items as the Intervenor suggest would mean GPUN is precluded from ever placing a shield plug over a loaded DSC while the cask is in the CDPS. This, GPUN declares, would violate numerous agency regulatory requirements that require shielding for spent fuel moved out of a spent fuel pool. GPUN maintains that sanctioning such an untoward result is inconsistent with the NUREG-0612 and its "to the extent practical" language, which in summarizing its recommended defense-in-depth measures declared that licensees should "'define safe travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment.'" *Id.* at 23 (quoting NUREG-0612, at 5-2). Pointing to the Staff's use of the same language in a 1985 generic letter in which the Staff recognized the need to handle the reactor vessel head over spent fuel in an open reactor vessel head

during refueling, GPUN asserts that without such an interpretation spent fuel can never be removed from the spent fuel pool. Because there is no other alternative, GPUN declares, the only conclusion is that this “to the extent practical” language sanctions the shield plug movement. *See id.* at 23-24.

2. *The Staff’s Response*

In its response supporting GPUN’s motion, the Staff likewise declares that, as with a Staff regulatory guide, NUREG-0612 is only a guidance document that does not prescribe requirements. *See* Staff Response at 6-7. The Staff further asserts that any technical specification, including OCNGS Technical Specification 5.3.1.B, can be changed so long as the amended provision provides reasonable assurance of protection of the public health and safety. *See id.* at 7-8. In addition, addressing the Licensee’s argument that GPUN really did not need the requested amendment, the Staff cites an October 5, 1995 Staff-issued amendment for the Rancho Seco Nuclear Power Station similar to that recently granted GPUN and concludes “not only may a licensee move a shield plug over spent fuel despite a Technical Specification like [Technical Specification] 5.3.1.B (prior to the Nov. 7th amendment) (Palisades), it may amend that Technical Specification to clarify that it can move a shield plug over spent fuel in the canister/cask (Rancho Seco).” *Id.* at 8-9.

3. *The Intervenors’ Arguments*

Intervenors NIRS and OCNW oppose the Licensee’s summary disposition motion. They declare that the intent of Technical Specification 5.3.1.B with its prohibition on carrying a load heavier than a single spent fuel assembly over irradiated fuel was to ensure OCNGS operations were within the facility’s engineering design basis, which included the offsite dose limitations set forth in 10 C.F.R. Part 100. The subsequent issuance of NUREG-0612, the Intervenors claim, was not intended to alter this design basis, but rather to provide guidance for handling loads greater than a single fuel assembly. According to the Intervenors, with its “to the extent practical” qualifier, NUREG-0612 specified two permissible options for dealing with these loads: (1) safe load paths that precluded heavy load transportation over irradiated fuel; or (2) use of a single-failure-proof crane. Before it was amended in November 1996, OCNGS Technical Specification 5.3.1.B with its prohibition on moving heavy loads over irradiated fuel satisfied the first option. If, however, GPUN wants to move heavy loads over irradiated fuel, the Intervenors argue that the Licensee must comply with the second option by installing a single-failure-proof crane. *See* NIRS/OCNW Response at 6-7.

The Intervenor also declare that, notwithstanding the Licensee and Staff attempts to obscure various references to NUREG-0612 "requirements" by rendering those references interchangeable with the term "guidelines," the provisions of NUREG-0612 embody the fundamental regulatory mandate of defense in depth that must be complied with. *See id.* at 8-9. Further, Intervenor NIRS and OCNW describe as "legalistic semantics" the GPUN attempt to establish that Technical Specification 5.3.1.B never applied to the movement of the DSC shield plug based on the purported distinction between whether fuel assemblies are in the spent fuel pool for "storage" or for "transport." *Id.* at 9. They also suggest that the prior cask movement described by GPUN either was an undetected noncompliance or, at best, could be sanctioned under language of the pre-November 1996 technical specification because that movement involved offsite shipments, as opposed to the presently proposed activities that will involve the onsite ISFSI. Finally, the Intervenor question why it was necessary to seek this amendment at all if, as GPUN asserts, the mid-1980s transfer of fuel assemblies from the reprocessing facility was in compliance with the prior, unamended language of Technical Specification 5.3.1.B. *See id.* at 10. NIRS and OCNW conclude that GPUN's motion should be denied.⁴

4. GPUN's Reply

In reply,⁵ GPUN labels the Intervenor's various claims "unpersuasive" because they are based on mere allegations, without supporting affidavits, evidence, or other authority. GPUN Reply at 2. The Intervenor's attempt to lend regulatory significance to NUREG-0612 is, according to GPUN, a totally unsupported allegation that contradicts long-standing agency precedent regarding the weight to be given to such documents. GPUN also declares that, in light of this precedent and the Staff's uncontroverted confirmation that NUREG-0612 was not intended to impose regulatory strictures, there is no genuine material issue regarding the references to NUREG-0612 "requirements" in various Licensee and Staff documents. *See id.* at 3-4.

Further, according to GPUN, both it and the Staff have established Technical Specification 5.3.1.B was not adopted in response to NUREG-0612 and, in any event, was never intended to prevent moving a shield plug over a DSC containing spent fuel. In this regard, the Licensee classifies as "mere allegation and suspicion" the Intervenor's charge that an earlier offsite cask movement was

⁴ In establishing a schedule for summary disposition filings, we noted that the Intervenor could, if they wished, seek to establish their need for discovery to respond to the Licensee's motion. *See* LBP-96-23, 44 NRC at 166 n.20. The Intervenor's response makes no mention of the need for discovery.

⁵ Under our schedule governing dispositive motion filings, the Intervenor were entitled to file a reply to the Staff's response to GPUN's motion. *See* LBP-96-23, 44 NRC at 166. They made no such filing, however.

an undetected noncompliance and maintains the Intervenor's asserted distinction between offsite and onsite transportation is meaningless. *Id.* at 5-7.

Finally, GPUN argues the Intervenor's claim that consistent with NUREG-0612 it must use a single-failure-proof crane to move any heavy load over spent fuel should be rejected. This assertion is deficient, GPUN declares, because it is based on a misreading of NUREG-0612 and is an untimely new basis for the Intervenor's contention that they have failed to show meets the late-filing standards of 10 C.F.R. § 2.714(a). *See id.* at 7-9.

C. Discussion

1. *Applicability of Technical Specification 5.3.1.B to DSC Shield Plug Movements*

In assessing the various arguments made by GPUN in support of its dispositive motion, we begin with the Licensee's assertion the requested amendment is really unnecessary because Technical Specification 5.3.1.B, as it existed prior to the November 1996 "no significant hazards consideration" amendment, already permitted GPUN to place the shield plug over the irradiated fuel in a DSC. As described above, GPUN has put forth a host of explanations as to why this is so, including references to Staff and Licensee interpretations of that language and Staff interpretations of similar language in the agency's standard technical specification and other facility technical specifications relating to movement of heavy loads.

As GPUN acknowledges, however, the first interpretational tool is the plain meaning of the language of the provision in question. *See* GPUN Dispositive Motion at 12 & n.6. In this instance, GPUN asserts, the references in Technical Specification 5.3.1.B to "stored irradiated fuel" in the "the spent fuel storage facility" settle the issue of its meaning. According to GPUN, the CDPS containing the DSC is not a "storage" area nor is irradiated fuel in the assemblies in the DSC "stored."

The problem with this claim, at least insofar as it is asserted to establish a clear and unambiguous meaning, is that it does not account adequately for the physical circumstances regarding spent fuel handling at OCNGS as they have been presented to us. As we noted in our previous determination, *see* LBP-96-23, 44 NRC at 149, the CDPS is a cylinder physically located within and attached to the walls of one corner of the OCNGS spent fuel pool — i.e., the OCNGS "spent fuel storage facility" — in which irradiated fuel is stored. The CDPS is configured this way so that while spent fuel assemblies are being loaded into a DSC, those assemblies can remain submerged in the water that fills the spent fuel pool and provides shielding and residual heat removal for the stored spent fuel. Given this physical configuration, at least so long as the

irradiated fuel remains within the confines of the spent fuel pool, the distinction between “storage” and “packaging/transfer” upon which GPUN seeks to rely is, in our estimation, too problematic to allow us to conclude the language of Technical Specification 5.3.1.B is “unambiguous” in this regard.

This ambiguity in the language of Technical Specification 5.3.1.B necessarily causes us to look for other clues to its meaning. GPUN asserts, and the Staff seemingly agrees, that a number of circumstances support its reading of this technical specification, including GPUN’s past practice under this provision and the Staff’s interpretation of similar provisions. The Licensee, however, does not make reference to one interpretational tool that has been found significant in resolving language construction issues — a subsequent enactment that declares the intent of an earlier provision. As the United States Supreme Court has noted, such later enactments generally are to be given “great weight” in resolving a construction problem. *See Red Lion Broadcasting v. FCC*, 395 U.S. 367, 380-81 (1969); *cf.* 17A Am. Jur. 2d *Contracts* § 388, at 415-16 (1991) (when contract terms are ambiguous and parties have made other contracts concerning the same subject matter, those instruments can be examined together to aid in interpretation). The relevance of such a subsequent enactment seems particularly telling here when the parties who drafted and approved the revision declare it was intended to clarify any ambiguity in the prior version. *See* GPUN Dispositive Motion at 11; Staff Response at 7.

The language of the recent revision to this technical specification makes it readily apparent the interpretation of its predecessor’s meaning now proffered by GPUN is not correct. After stating that heavy loads shall not be moved over stored irradiated fuel in the spent fuel storage facility, amended Technical Specification 5.3.1.B.1 adds the proviso “*except as noted in 5.3.1.B.2.*” Amended Technical Specification 5.3.1.B (emphasis supplied). Amended Technical Specification 5.3.1.B.2 then states that the shield plug may be moved over irradiated fuel in a DSC in the CDPS.

The use of the term “except” in paragraph one of amended Technical Specification 5.3.1.B to describe the shield plug heavy load activity sanctioned in paragraph two, plainly establishes that, but for its specification as an exception, this activity would be prohibited by paragraph one. Otherwise, there would be no reason to create the exception. As the GPUN technical specification is now worded, therefore, it indicates quite clearly that, without the specified exception, the DSC shield plug activity over irradiated fuel that is the focus of GPUN’s amendment request would be a prohibited heavy load activity. And because the prohibition language in amended paragraph 5.3.1.B.1 is indistinguishable from that in Technical Specification 5.3.1.B prior to that recent revision, the construction rule regarding subsequent enactments counsels that, affording considerable weight to an unambiguous expression of intent by the drafting and enacting parties, we give a parallel construction to these identical provisions.

We must, therefore, reject GPUN's claim it is entitled to summary disposition because the shield plug movement activity in question is not covered under the terms of Technical Specification 5.3.1.B prior to its revision in November 1996.⁶

2. *The Status and Meaning of NUREG-0612*

Having concluded that the technical specification at issue here would, unless amended, preclude the Licensee's planned shield plug movement activity, we next consider whether, as the Intervenor asserts, the amendment proposed by GPUN and adopted by the Staff in November 1996 is appropriate in light of NUREG-0612. As we have explained, the Intervenor's claim Technical Specification 5.3.1.B cannot be amended as GPUN has asked because to do so would violate the precepts of NUREG-0612 as it implements the agency's defense-in-depth approach to regulation.⁷

a. *Background on NUREG-0612*

In analyzing this assertion, we begin with an overview of NUREG-0612, the central focus of the Intervenor's contention before the Board. This 1980 document sets forth the results of a Staff attempt to make a systematic examination of the adequacy of then-existing measures for handling of "heavy loads" at nuclear power plants.⁸ In its initial summary, the report states:

This report provides the results of the NRC staff's review of the handling of heavy loads and includes the NRC staff's recommendations on actions that should be taken to assure safe handling of heavy loads. These recommendations include: (1) a program should be initiated to review operating plants against the guidelines developed in [this report]; (2) certain interim measures should be taken for operating plants until completion of this review program; (3) changes to certain Standard Review Plans and Regulatory Guides should be made to incorporate the guidelines in this report; (4) changes to technical specifications should be made after completion of the review; and (5) a task should be initiated to establish

⁶ In this connection, we are troubled by the Staff's apparent claim that under the language of Technical Specification 5.3.1.B before its recent revision, the Licensee was free to treat the movement of the shield plug over the DSC as either covered or not covered by that license requirement. See Staff Response at 8-9. Although we have no quarrel with the general proposition there may be more than one way to comply with a regulatory requirement, see *id.* at 6, as a matter of logic we are hard pressed to understand how a directive that states heavy loads "shall not" be moved over irradiated fuel can be read to both sanction and prohibit the same heavy load movement activity. From an enforcement perspective, such an interpretation renders that "requirement" essentially meaningless.

⁷ As we noted in our October 1996 issuance, "[t]he 'defense-in-depth' principle is the agency policy under which regulated entities are required to safeguard the public health and safety 'through multiple intermeshing and overlapping protections.'" LBP-96-23, 44 NRC at 162 n.14 (quoting *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-74-40, 8 AEC 809, 813 (1974)).

⁸ In using the term "heavy load" in this decision, we adopt the definition of that phrase found in NUREG-0612, which classifies a "heavy load" as "any load that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool for the specific plant in question." NUREG-0612, at 1-2.

guidelines for the control of small loads near spent fuel. The guidelines proposed include definition of safe load paths, use of load handling procedures, training of crane operators, guidelines on slings and special lifting devices, periodic inspection and maintenance for the crane, as well as various alternatives that include: use of a single failure proof handling system, use of mechanical stops or electrical interlocks to keep heavy loads away from fuel or safe shutdown equipment, or analyzing the consequences of postulated heavy load drops to show these are within acceptable limits.

NUREG-0612, at iii. The report then goes on to provide a generic analysis of the consequences of heavy load drops, including the "potential problem areas" of offsite releases from heavy load drops on spent fuel or safe shutdown equipment and recriticality from fuel reconfiguration; a survey of licensee information on load handling operations at reactor facilities; a review of historical data on crane operations; guidelines that describe alternative approaches for heavy loads control; and a program for implementing the suggested guidelines at operating facilities, including suggested standard review plan, regulatory guide, and technical specification changes. *See id.* at v-vi.

As highlighted by the parties in their various filings, several portions of this NUREG document potentially are pertinent to any resolution of the merits of the Intervenor's contention. For instance, as we previously noted, in describing the results of its survey on load handling procedures, NUREG-0612 indicates that OCGS was one of twenty-seven plants without a technical specification prohibiting handling of heavy loads over spent fuel. *See id.* at 3-8, 3-9 (Table 3.2-1).

Thereafter, in section 5 of the report entitled "GUIDELINES FOR CONTROL OF HEAVY LOADS," addressing the general problem of load drop accidents the report declares that although existing operating facility heavy load handling measures cover certain of the potential problem areas, they nonetheless varied widely and did not adequately address the major causes of load handling accidents. The report identifies these causes as operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. Subsequently, in section 5.1 of the report under the heading "*Recommended Guidelines*," NUREG-0612 sets forth a series of items designed to upgrade the measures already in effect "[t]o provide adequate measures that minimize the occurrence of the principal causes of load handling accidents and to provide an adequate level of defense-in-depth for handling heavy loads near spent fuel and safe shutdown systems." *Id.* at 5-1.

According to the report, the objectives of the alternative approaches it sets forth as guidelines for controlling heavy loads are to assure either (1) an extremely small load drop potential, or (2) for each of the potential problem areas, satisfaction of four "evaluation criteria." These criteria include keeping any damaged spent fuel releases well within 10 C.F.R. Part 100 limits; preventing fuel and storage rack damage from resulting in a configuration that creates an

effective multiplication factor (k_{eff}) larger than 0.95; keeping reactor vessel or spent fuel pool damage from resulting in water leakage that would uncover the fuel; and limiting damage to redundant or dual safe shutdown path equipment so as not to result in a loss of required safe shutdown functions. *See id.* at 5-1. NUREG-0612 then goes on to provide:

After reviewing the historical data available on crane operations, identifying the principal causes of load drops, and considering the type and frequency of load handling operations at nuclear power plants, the NRC staff has developed an overall philosophy that provides a defense-in-depth approach for controlling the handling of heavy loads. This philosophy encompasses an intent to prevent as well as mitigate the consequences of postulated accidental load drops. The following summarizes this defense-in-depth approach:

- (1) Provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system; and
- (2) Define safe load travel paths through procedures and operator training so that to the extent practical heavy loads avoid being carried over or near irradiated fuel or safe shutdown equipment; *and*
- (3) Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Certain alternative measures may be taken to compensate for deficiencies in (2) and (3) above, such as the inability to prevent a particular heavy load from being brought over spent fuel (e.g., reactor vessel head). These alternative measures can include: increasing crane reliability by providing dual load paths for certain components, increased safety factors, and increased inspection as discussed in Section 5.1.6 of this report; restricting crane operations in the spent fuel pool area (PWRs) until fuel has decayed so that off-site releases would be sufficiently low if fuel were damaged; or analyzing the effect of postulated load drops to show that consequences are within acceptable limits. Even if one of these alternative measures is selected, (1) and (2) above should still be satisfied to provide maximum practical defense-in-depth.

NUREG-0612, at 5-1 to -2.

Thereafter, under the heading of "General," in section 5.1.1 NUREG-0612 describes seven criteria that all plants should satisfy in handling heavy loads that could be brought over or in the proximity of safe shutdown equipment or irradiated fuel in any plant area. These include (1) defining safe load paths to minimize the potential that any dropped heavy load would impact irradiated fuel or safe shutdown equipment; (2) developing procedures, such as premovement inspection criteria, to cover heavy load handling operations over or in the proximity of irradiated fuel or safe shutdown equipment; (3) training crane operators to conduct themselves in accordance with applicable American National Standards Institute (ANSI) standards; (4) ensuring that special lifting devices, such as spent fuel cask yokes and slings, satisfy applicable ANSI

guidelines; (5) ensuring that lifting devices that are not specially designed meet applicable ANSI guidelines; (6) inspecting, testing, and maintaining cranes in accordance with ANSI standards; and (7) designing cranes to meet ANSI and Crane Manufacturers Association of America (CMAA) standards.

Finally, relative to reactor buildings for BWR facilities such as OCNCS, in section 5.1.4 NUREG-0612 declares:

To assure that the evaluation criteria of Section 5.1 are satisfied one of the following should be met in addition to satisfying the general guidelines of Section 5.1.1:

- (1) The reactor building crane, and associated lifting devices used for handling the above heavy loads, should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

- (2) The effects of heavy load drops in the reactor building should be analyzed to show that the evaluation criteria of Section 5.1 are satisfied. The loads analyzed should include: shield plugs, drywell head, reactor vessel head; steam dryers and separators; refueling canal plugs and gates; shielded spent fuel shipping casks; vessel inspection platform; and any other heavy loads that may be brought over or near safe shutdown equipment as well as fuel in the reactor vessel or the spent fuel pool. Credit may be taken in this analysis for operation of the Standby Gas Treatment System if facility technical specifications require its operation during periods when the load being analyzed should be handled. The analysis should also conform to the guidelines of Appendix A.

NUREG-0612, at 5-6 to -7. And, as an interim measure to provide reasonable assurance that no spent fuel shipping casks or other heavy loads were handled over the spent fuel pool until the section 5.1 guidelines were finally implemented, NUREG-0612 declares that facility technical specifications "should be upgraded to prohibit handling of heavy loads over the spent fuel pool." *Id.* at 5-18.

The parties' filings also suggest that two agency generic letters issued in the wake of NUREG-0612 are relevant to our inquiry here. The first, an unnumbered letter dated December 22, 1980, set forth a two-stage process for licensee responses regarding compliance with the recommendations of NUREG-0612.⁹ As outlined in the December 1980 letter, in Phase I licensees were to identify their load handling equipment within the scope of NUREG-0612 and describe how their use of that equipment complied with the six general criteria specified in NUREG-0612 section 5.1.1. Thereafter, in Phase II, BWR licensees like GPUN were to provide a second response showing that, consistent with NUREG-0612 section 5.1.4, either single-failure-proof lifting equipment was provided or such equipment was not needed, as demonstrated in a detailed load drop analysis. See GPUN Dispositive Motion, ex. B, encl. 3, at 2-7 (Letter

⁹ On February 3, 1981, the Staff's December 22 letter was supplemented by Generic Letter 81-07, which provided missing pages for one of the enclosures.

from Darrel G. Eisenhut, Director, Division of Licensing, to All Operating Plant Licensees, Operating License Applicants, and Construction Permit Holders (Dec. 22, 1980)). The generic letter, however, did not request that licensees undertake any technical specification change regarding heavy loads, as had been suggested in NUREG-0612.

The other correspondence of potential import is Generic Letter 85-11, dated June 26, 1985, in which the Staff described its resolution of Phase II. *See id.*, exh. D (Letter from Hugh L. Thompson, Jr., Director, Division of Licensing, to All Licensees for Operating Reactors (June 26, 1985)). In an enclosure to this letter, the Staff stated that, based on its comprehensive review of licensee Phase I responses, licensee satisfaction of the Phase I guidelines had assured that the potential for a load drop accident was extremely small. Thus, the Staff found that Phase I guidelines were "adequately providing the intended level of protection against load drop accidents." *Id.*, encl. 1, at 3.

In this generic letter, the Staff also noted that although all licensees had provided a Phase II submittal, because the Staff considered Phase II an enhancement of Phase I, it had decided to conduct a pilot program review of a limited number of plants to aid in deciding whether to undertake an equally extensive review of all Phase II submittals. According to the Staff, based on its pilot program review of twelve operating reactor sites as well as its review of five operating license applicants, it had concluded most risk associated with carrying heavy loads involved possible damage to spent fuel rather than safe shutdown systems. The Staff further declared that, as a result of licensee Phase I activities, the handling of heavy loads over spent fuel had been limited to the extent practical but, where necessary, was being performed in conformance with Phase I guidelines. *See id.* at 3-4.

There remained, however, the question of whether under Phase II licensees wishing to handle heavy loads over spent fuel would have to either install costly single-failure-proof cranes or perform costly detailed load drop analyses. The Staff concluded that with Phase I implementation improvements and based on its review of individual licensee Phase II submittals, it did not perceive a significant enough benefit in requiring costly conversion to single-failure-proof cranes or find any outstanding plant-specific concerns. Thus, the Staff declared Phase II was considered complete without further Staff or licensee action. *See id.* at 4-6.

b. Status of NUREG-0612

With this background in mind, we turn to the question of the status of NUREG-0612 as it impacts on the requested GPUN technical specification change. The Intervenor has asserted the provisions of NUREG-0612 effectively bar the requested revision. Although both the Licensee and the Staff vigorously oppose this notion, as we observed in accepting the Intervenor's

legal contention framing their NUREG-0612-based challenge to GPUN's license amendment, there are any number of instances in Licensee and Staff documents in which the terms "NUREG-0612" and "requirement" are linked. If NUREG-0612 did indeed establish "requirements," its provisions seemingly would be on a par with legally binding directives such as a statute, regulation, license condition, or order and so might, depending on its terms, preclude adoption of a requested technical specification change.

As both the Licensee and the Staff point out, however, the Commission previously has declared that a Staff report bearing the NUREG designation, such as NUREG-0612, does not fall into this category. See *Curators of the University of Missouri*, CLI-95-1, 41 NRC 71, 98 (1995). Instead, at best, "it serves as guidance, setting forth but one method for meeting the applicable regulatory requirements In other words, that document 'is treated simply as evidence of a legitimate means for complying with regulatory requirements.'" *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant), ALAB-852, 24 NRC 532, 544-45 (1986) (quoting *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), ALAB-698, 16 NRC 1290, 1298-99 (1982), *aff'd in part on other grounds*, CLI-83-22, 18 NRC 299 (1983)). Certainly, nothing in NUREG-0612 itself suggests the provisions of that document should have any other standing. See, e.g., NUREG-0612, at iii, 1-4 (NUREG-0612 provides the Staff's "recommendations" and "guidelines" for actions that should be taken to assure safe handling of heavy loads). See also Staff Response, attach. 2, at 4 (NUREG-0612 was intended to provide guidance and acceptance criteria, not regulatory requirements) (Affidavit of Harold Walker in Support of the NRC Staff's Response in Support of the Licensee's Motion for Summary Disposition (Dec. 6, 1996) at 4); *id.*, exh. 2, at 2 (NRC positions communicated to licensees in NUREG reports are not binding requirement unless formally issued as regulations or included in order or as part of a permit or license) (NRC Management Directive Handbook 3.7, at 8 (rev. Feb. 9, 1995)).

With this Commission explanation of the status of NUREG documents generally as well as NUREG-0612's own description of the scope of its provisions, the question becomes whether anything on the record before us establishes that report's terms should be given a different status. As we have pointed out, there are various Licensee and Staff references to NUREG-0612 "requirements." Nonetheless, when viewed against the Commission's clear declaration about the status of NUREG documents, we can only conclude these otherwise unexplained references do not accurately reflect the status of that document and its provisions. That they suggest an apparent misunderstanding of this document's status is unfortunate, but in this instance these misstatements do not change the fundamental nature of this NUREG document or its provisions.

NUREG-0612 does not itself contain "requirements," but rather Staff "guidance" on assuring safe handling of heavy loads.¹⁰

c. *Meaning of NUREG-0612*

Ultimately, however, whether the provisions of NUREG-0612 are found to constitute guidance or requirements, if GPUN's amendment does not violate that document's dictates, then, at least as the issue before us has been framed by the Intervenor, summary disposition should be entered in favor of GPUN. In accepting the Intervenor's contention as litigable, the factor the Board found significant in this regard was the apparent timing of the adoption of Technical Specification 5.3.1.B as reflected in NUREG-0612.

The seeming adoption of this technical specification after the publication of NUREG-0612 suggested that the heavy load movement prohibition it contained might, as the Intervenor has maintained, reflect the Staff's ultimate judgment about how GPUN should conform with the provisions of NUREG-0612. It is apparent, however, that as GPUN has asserted (without contradiction from the Intervenor or the Staff), the information in NUREG-0612 regarding the OCNCS technical specification was incorrect. In fact, Technical Specification 5.3.1.B was adopted in 1977, some 3 years before NUREG-0612 was published. *See* GPUN Dispositive Motion at 19 & n.13; Fornicola Affidavit at 3. Thus, the timing of this technical specification's adoption provides no support for the Intervenor's assertion the technical specification's language prohibiting the movement of heavy loads over stored spent fuel was intended to reflect a NUREG-0612-dictated irrevocable prohibition for OCNCS.¹¹

¹⁰As we have observed above, in several generic letters the Staff both "requested" that licensees take various actions to conform with the recommendations on handling heavy loads outlined in NUREG-0612 and "required" that licensees provide a report detailing their efforts in this regard. In contrast to the reporting component in these generic letters, which seemingly would constitute a "requirement," *see* 10 C.F.R. §§ 2.204, 50.54(f), the generic letters' compliance requests did not constitute "requirements" in the absence of some additional regulatory directive such as an order or a regulation mandating compliance. *Cf.* 60 Fed. Reg. 34,381, 34,392 (1995) (agency expects licensees to adhere to commitments resulting from administrative actions such as confirmatory action letters and will issue appropriate orders to ensure commitments are met), *reprinted in* Office of Enforcement, NRC, NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions" at 14 (July 1995).

¹¹To be sure, a technical specification that is not subject to revision would not be the norm. By providing in section 187 of the Atomic Energy Act that agency-issued licenses are "subject to amendment," 42 U.S.C. § 2237; *see also, e.g.*, 10 C.F.R. § 50.90, the Congress contemplated that any license provision could be changed, at least so long as the revision sought was not inimical to the public health and safety or the common defense and security. Consequently, in the absence of language in the license (or some other regulatory requirement) that makes manifest a license provision's immutability, the question in a license amendment proceeding generally is whether the requested change is consistent with applicable agency regulatory strictures and any suitable guidance.

As is apparent from a reading of Technical Specification 5.3.1.B, nothing on the face of that provision suggests there is any basis for finding it an irrevocable license condition. The same is true for the other regulatory requirements that the Staff has identified as potentially pertinent to GPUN's requested technical specification change. *See* Staff Response, attach. 2, at 4-5. These include General Design Criterion (GDC) 2, which establishes

(Continued)

This leaves only the provisions of NUREG-0612 as the supporting source for the Intervenor's assertion that OCNCS Technical Specification 5.3.1.B cannot be revised to permit hauling heavy loads such as the DSC shield plug over spent fuel, including the fuel inside a DSC within the CDPS in the spent fuel pool.¹² The problem for the Intervenor is that the NUREG-0612 guidance in fact contemplates there are instances when, with the proper safeguards, heavy loads can be hauled over spent fuel.

As we noted above, NUREG-0612 recommends that, consistent with the agency's defense-in-depth approach, in handling heavy loads, operator training, load handling instructions, and equipment inspections be provided sufficient to assure reliable handling system operation; safe load paths be defined through procedures and operator training so that "to the extent *practical*" heavy loads are not carried over or near spent fuel; and mechanical stops and electrical interlocks be provided to prevent movement of heavy loads over irradiated fuel. NUREG-0612, at 5-2 (emphasis supplied). NUREG-0612 then goes on to declare that if there are deficiencies concerning these measures "such as an inability to prevent a particular heavy load from being brought over spent fuel," alternative measures may be utilized, such as increasing crane reliability or analyzing postulated load drop effects to show that any consequences are within acceptable limits, so long as those measures in combination with the above-specified defense-in-depth measures, will provide the "maximum *practical* defense-in-depth." *Id.* (emphasis supplied).

With its repeated emphasis on "practicality," the upshot of this guidance is not that heavy loads can never be moved over irradiated fuel. Rather, NUREG-0612 seeks to ensure that through the use of a combination of preventative measures — including crane operator training, systems and equipment upgrades and inspections, load handling instructions and procedures, and load movement planning that sets practical limits on spent fuel exposure to heavy loads — the risks inherent in hauling large loads over spent fuel are reduced to permissible levels. NUREG-0612 clearly recognizes it sometimes is necessary to move heavy loads over spent fuel, as is the case with the DSC shield plug, but that such action should be taken only after the risks involved have been confined at acceptable levels through the implementation of appropriate safeguards.

design bases for protection against natural phenomena such as earthquakes; GDC 4, which concerns design bases for environmental and dynamic effects such as missiles; GDC 5, which sets forth design bases for shared structures, systems, and components that are important to safety; and GDC 61, which establishes design criteria for fuel storage and handling. See 10 C.F.R. Part 50, App. A, § I, Criteria 2, 4, & 5, § VI, Criterion 61.

¹² In responding to the Licensee's summary disposition motion, the Intervenor has made no claims regarding the applicability of Regulatory Guides 1.13 and 1.29, which concern the design basis for spent fuel storage facilities and seismic design classification, respectively. See Staff Response, attach. 2, exhs. 3 and 4 (Office of Standards Development, NRC, Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis (rev. 1, Dec. 1975) (for comment); *id.* Regulatory Guide 1.29, Seismic Design Classification (rev. 3, Sept. 1978)).

In contesting GPUN's summary disposition motion, with one exception the Intervenor's have not sought to challenge the adequacy of GPUN's implementation of the various preventative measures (such as ensuring that crane operators are adequately trained and load handling procedures are developed) that NUREG-0612 suggests should be put in place to ameliorate the risks inherent in heavy load hauling. This single exception is their argument that, consistent with NUREG-0612, GPUN can move the shield plug only by installing and using a "single-failure-proof" crane, which GPUN does not have.¹³

As we described in the background discussion above, the Staff once contemplated that for BWR facilities like OCNGS to comply with the guidance in NUREG-0612, besides providing the various preventative measures discussed above, a licensee would have to show (1) the reactor building crane and associated lifting devices met the single-failure-proof guidelines,¹⁴ or (2) the effects of any remaining potential heavy load drop events in the reactor building, including those involving shield plugs, would satisfy the evaluation criteria in NUREG-0612 section 5.1, including its specification that any releases fall within 10 C.F.R. Part 100 limits and any fuel reconfiguration not exceed an effective multiplication factor of 0.95. Indeed, as it was outlined in the Staff's December 1980 generic letter, this was to be the second phase of the Staff's NUREG-0612 guidance implementation program.

It also is apparent, however, that the Staff later determined, based on its assessment of the Phase I implementation activities of licensees such as GPUN and a pilot program review of a selection of the submittals provided by all licensees addressing the Phase II criteria, that this Phase II activity was not necessary. Describing the results of Phase I in Generic Letter 85-11, the Staff declared:

Our review has indicated that satisfaction of the Phase I guidelines assures that the potential for a load drop is extremely small. We have noted improvements in heavy load handling procedures and training and crane and handling tool inspection and testing. *These changes have been geared to limiting the handling of heavy loads over safety-related equipment and spent fuel to the extent practical, but where this can not be avoided, to accomplishing it with the operational and other features of the program implemented in Phase I.* We therefore conclude that the guidelines of Phase I are adequately providing the intended level of protection against load drop accidents.

¹³ Although GPUN has challenged this claim as a late-filed basis for the Intervenor's contention that they have not attempted to show meets the criteria for late-filed submissions, we consider this assertion within the confines of the Intervenor's admitted legal contention and basis.

¹⁴ As described in NUREG-0612, a "single-failure-proof" crane must have certain active components meeting improved redundancy or duality evaluation criteria that render the crane highly reliable. See NUREG-0612, at 5-7.

GPUN Dispositive Motion, exh. D., encl. 1, at 2-3 (emphasis supplied). At the same time, based on its Phase II pilot program review, the Staff found that with the Phase I improvements, there was no cost/benefit justification for requiring licensees to perform costly detailed load analyses or install costly single-failure-proof cranes. The Staff concluded:

[W]e believe the Phase I implementation has provided sufficient protection such that the risk associated with potential heavy load drops is acceptably small. We further conclude that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense in depth" is satisfied by the Phase I compliance, and that the Phase II analyses did not indicate the need to require further generic action at this time. This conclusion has been confirmed by the results obtained from the Phase II pilot program and additional Phase II reviews, which identified no residual heavy loads handling concerns of sufficient significance to demand further generic action. All plants have examined their load handling practices against the recommendations of Phase II and submitted the Phase II report. In this way, the utilities were required to identify any unexpected problems to the Staff.

Id. at 5-6. Thus, without installing a single-failure-proof crane, reactor licensees, including GPUN, were found by the Staff to have complied with the guidance in NUREG-0612 as it was intended to implement the agency's defense-in-depth principle.

The Intervenor has presented nothing that calls into question the efficacy of the Staff's June 1985 generic determination not to impose single-failure-proof crane installation on GPUN (or any other licensee) as a condition for compliance with the guidance it set forth in NUREG-0612.¹⁵ Nor have the Intervenor presented anything that would lead us to conclude relative to the technical specification at issue here that a different result is required in order to comply with the Staff's NUREG-0612 guidance as set forth in that document or the subsequent generic letters describing how that guidance was to be implemented.¹⁶ In the context of this case, therefore, we find nothing in NUREG-0612 (whether

¹⁵ Although it might be asserted the Staff's decision, as reflected in Generic Letter 85-11, not to mandate single-failure-proof crane installation for GPUN simply reflects a Staff recognition of the then-existing prohibition on heavy load handling found in Technical Specification 5.3.1.B, this does not account for the fact there apparently were numerous other facilities without such a technical specification that were not required to adopt such a license condition or to implement the initial NUREG-0612 guidance regarding single-failure-proof crane installation.

¹⁶ As outlined above, under the terms of the Staff's December 1980 generic letter, in the absence of a single-failure-proof crane GPUN would have been required to provide an analysis showing that any heavy load drop accident involving the spent fuel in the DSC/CDPS would satisfy the evaluation criteria in section 5.1 of NUREG-0612, including showings that any resulting releases would not violate 10 C.F.R. Part 100 limits and that any ensuing fuel reconfiguration would not result in an effective multiplication factor exceeding 0.95. As the Staff recently has made clear, however, the closeout of Phase II under Generic Letter 85-11 did not relieve licensees of the responsibility to evaluate any planned new heavy load activities under their existing technical specifications to ensure those activities do not involve an unreviewed safety question that would warrant a license amendment. See Hearing Petition, unnumbered attach. 6, at 5-6 (based on Staff audit of GPUN submission claiming no unreviewed safety issues in proposal to haul loaded DSC/TC over safety-related equipment while OCNCS is operating, Staff advises licensees of responsibility to evaluate heavy load activities and requires report discussing need for any technical specification changes

(Continued)

or not it is considered a regulatory requirement) that would, as a matter of law, preclude the adoption of GPUN's requested technical specification revision.

III. CONCLUSION

Contrary to the assertions of GPUN and the Staff, we find that, prior to its recent amendment pursuant to the Staff's November 1996 "no significant hazards consideration" determination, OCNGS Technical Specification 5.3.1.B did apply to the movement of heavy loads over irradiated fuel in a DSC within the spent fuel pool CDPS. We also find, however, that GPUN has established there is no genuine issue as to any material fact and it is entitled to a judgment as a matter of law on its claim that, contrary to the Intervenor's contention, nothing in the guidance in NUREG-0612 precludes the grant of the technical specification revision GPUN has sought.

For the foregoing reasons, it is, this 31st day of January 1997, ORDERED that:

1. The November 15, 1996 motion for summary disposition of GPUN is *granted* and, for the reasons given in this Memorandum and Order, a decision regarding the merits of the Intervenor's admitted legal contention is rendered in favor of GPUN.

2. Pursuant to 10 C.F.R. § 2.760, this decision will become the final decision of the Commission 40 days from the date of its issuance (i.e., on *Wednesday, March 12, 1997*), unless a petition for review is filed in accordance with section 2.786, or the Commission directs otherwise.

3. As the determination rendered herein terminates this proceeding before the Licensing Board, pursuant to 10 C.F.R. § 2.786(b)(1), within 15 days after service of this Memorandum and Order a party may file a petition for review with the Commission on the grounds specified in section 2.786(b)(4). The filing of a petition for review is mandatory in order for a party to have exhausted its administrative remedies before seeking judicial review. Within 10 days after service of a petition for review, any party to this proceeding may file an answer

to address planned heavy load activities) (NRR, NRC, NRC Bulletin 96-02: Movement of Heavy Loads over Spent Fuel, over Fuel in the Reactor Core, or over Safety-Related Equipment (Apr. 11, 1996)).

Subsequent to the deadline for filing contentions, GPUN made available to the Staff and the Intervenor several "worst case" analyses that appear to address the NUREG-0612 evaluation criteria. See LBP-96-23, 44 NRC at 155-56; see also GPUN Dispositive Motion, ex. A., encl. 2, at 3-5 (NRR, NRC, "Safety Evaluation of [NRR] Related to Amendment No. 187 to Facility Operating License No. DPR-16 [GPUN] and Jersey Central Power & Light Company [OCNGS] Docket No. 50-219" (Nov. 7, 1996)). The Intervenor has not made any attempt to contest the validity of those analyses in conformance with the standards governing late-filed contentions and bases. See LBP-96-23, 44 NRC at 163 n.16.

supporting or opposing Commission review. The petition for review and any answers shall conform to the requirements of section 2.786(b)(2)-(3).

**THE ATOMIC SAFETY AND
LICENSING BOARD**

**G. Paul Bollwerk, III, Chairman
ADMINISTRATIVE JUDGE**

**Charles N. Kelber
ADMINISTRATIVE JUDGE**

**Peter S. Lam
ADMINISTRATIVE JUDGE**

**Rockville, Maryland
January 31, 1997**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of

Docket Nos. 50-255
72-7

CONSUMERS POWER COMPANY
(Palisades Nuclear Plant)

January 23, 1997

The Acting Director of the Office of Nuclear Reactor Regulation is granting, in part, and denying, in part, a petition filed by the organizations Don't Waste Michigan and Lake Michigan Federation pursuant to 10 C.F.R. § 2.206. The Petitioners requested that the NRC (1) find that Consumers Power Company violated NRC requirements related to unloading procedures for dry storage casks for spent nuclear fuel, (2) suspend the Licensee's use of the general license provisions related to dry cask storage of spent nuclear fuel, (3) require a substantial penalty be paid by the Licensee, and (4) conduct hearings related to unloading procedures for dry storage casks at Palisades. To the extent that the NRC has determined that Consumers Power Company violated NRC regulations insofar as the original unloading procedure developed for unloading dry storage casks was not adequate, the petition is granted. However, the NRC has decided not to impose a civil penalty for the violation or to suspend Consumers Power Company's use of the general license for dry cask storage at Palisades. To that extent, the petition is denied.

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

I. INTRODUCTION

On September 19, 1995, the organizations Don't Waste Michigan and Lake Michigan Federation (Petitioners) filed a petition pursuant to section 2.206 of

Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206) requesting that the U.S. Nuclear Regulatory Commission (NRC) (1) find that Consumers Power Company (Licensee) violated NRC requirements related to unloading procedures for dry storage casks for spent nuclear fuel, (2) suspend the Licensee's use of the general license provisions related to dry cask storage of spent nuclear fuel, (3) require a substantial penalty be paid by the Licensee, and (4) conduct hearings related to unloading procedures for dry storage casks at Palisades.

On September 30, 1996, the Petitioners amended the petition by including additional information in support of their position that the Licensee did not have a workable unloading procedure before loading the thirteen dry storage casks currently in the Palisades independent spent fuel storage installation (ISFSI).

The petition has been referred to me pursuant to section 2.206. The NRC letter dated October 24, 1995, to Dr. Sinclair and Mr. Skavronck, on behalf of the Petitioners, acknowledged receipt of the petition. Notice of receipt was published in the *Federal Register* on October 31, 1995 (60 Fed. Reg. 55,388).

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

II. BACKGROUND

NRC regulations contain a general license that authorizes nuclear power plants licensed by the NRC, such as Palisades, to store spent nuclear fuel at the reactor site in storage casks approved by the NRC. (*See* 10 C.F.R. Part 72, Subpart K.) In regard to dry cask storage of spent nuclear fuel at Palisades, the Licensee opted to use the VSC-24 Cask Storage System designed by Sierra Nuclear Corporation. The VSC-24 Cask Storage System was added to the list of NRC-certified casks in May 1993 (58 Fed. Reg. 17,948). The associated certificate of compliance, Certificate No. 1007, specifies the conditions for use of VSC-24 casks under the general license provisions of Part 72. Section 1.1.2, "Operating Procedures," in the certificate of compliance for the VSC-24 casks, requires that licensees prepare an operating procedure related to cask unloading. Specifically, the condition states

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the SAR [safety analysis report] are considered appropriate, as discussed in Section 11.0 of the SER [safety evaluation report], and should provide the basis for the user's written operating procedures. The following additional written procedures shall also be developed as part of the user operating procedures:

1. A procedure shall be developed for cask unloading, assuming damaged fuel. If fuel needs to be removed from the multi-assembly sealed basket (MSB), either at the end of service life or for inspection after an accident, precautions must be taken against

the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This activity can be achieved by the use of the Swagelok valves, which permit a determination of the atmosphere within the MSB before the removal of the structural and shield lids. If the atmosphere within the MSB is helium, then operations should proceed normally, with fuel removal, either via the transfer cask or in the pool. However, if air is present within the MSB, then appropriate filters should be in place to permit the flushing of any potential airborne radioactive particulate from the MSB, via the Swagelok valves. This action will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

The Licensee for Palisades began loading casks in May 1993 after implementing pertinent certificate conditions, including those in section 1.1.2.

In July 1994, the Licensee discovered radiographic indications of possible defects in a weld in multiassembly sealed basket (MSB) No. 4. MSB No. 4 had been loaded with spent fuel earlier that month and placed, inside a ventilated concrete cask, on the ISFSI storage pad. The Licensee evaluated the flaw indications and determined that the MSB continued to meet its design basis and was capable of safely storing spent fuel for the duration of the certificate (20 years). Nevertheless, the Licensee stated that MSB No. 4 would be unloaded to support additional inspections and evaluations related to its future use.¹ In preparation for the unloading of MSB No. 4, the Licensee reviewed the unloading procedure issued in May 1993 (Revision 0) and identified several technical questions. A revision of the unloading procedure (Revision 1) was subsequently developed to resolve the identified technical questions.

The technical questions and the associated procedural changes were discussed during meetings with the NRC Staff, and additional information was provided in submittals from the Licensee to the NRC. Evaluation of the revised unloading procedure by the NRC Staff was initially made through the review of submittals from the Licensee and has continued through an inspection of the Licensee's revised unloading procedure.

As a result of its inspections and reviews, the NRC Staff recognized that some licensees, including Consumers Power Company, had developed unloading procedures that tended to be simplistic and lacked sufficient details and contingencies. In order to address these issues, an item related to cask loading and unloading procedures was added to the NRC dry cask storage action plan that was implemented in July 1995. Some issues, such as the thermal-hydraulic

¹The schedule for unloading MSB No. 4 remains indefinite. The Staff has recently learned that the Licensee may postpone the unloading until a multipurpose cask is available. This would allow the spent fuel currently stored in MSB No. 4 to be transferred to a cask that would support both storage and transportation of the spent fuel. The NRC Staff is reviewing this plan and will initiate discussions pertaining to this matter with the Licensee and other affected parties.

behavior of casks during the unloading process, were included largely as a result of questions related to the original unloading procedure at Palisades. Experience at other facilities using storage and transportation casks resulted in the identification of other issues. For example, as a result of the turbidity of the spent fuel pool during the unloading of a transportation cask at the Shearon Harris Nuclear Power Plant, the NRC Staff assessed the potential for and significance of deposits on fuel assembly surfaces becoming loose during the unloading of dry storage casks. Evaluations and inspections were used to resolve these issues for specific facilities, and revisions to NRC guidance documents have been prepared to resolve generic concerns.

Completion of the NRC inspection of the revised unloading procedure for Palisades was postponed following an event at the Point Beach Nuclear Plant.² Following the hydrogen ignition event at Point Beach, 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 document 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," and confirmation of corrective actions by the NRC.³

The NRC Staff is continuing to review the bulletin responses and corrective actions for the Palisades facility, and, therefore, the Licensee is restrained from loading or unloading additional VSC-24 casks. Completion of the ongoing NRC inspection of the revised unloading procedure at Palisades will be coordinated with the Staff's review of the Licensee's response to the bulletin. Further, the NRC has committed to state officials and members of the public that the exit meeting for the inspection at Palisades will be open to the public, the meeting will be noticed sufficiently in advance to allow interested parties to attend, and the NRC Staff will allocate time to discuss issues with the public following the meeting with the Licensee.

III. DISCUSSION

The petition requests four actions by the NRC on the basis of the contention that the original unloading procedure (Revision 0) implemented by the Licensee was inadequate, and therefore, the Licensee violated NRC regulations requiring

² On May 28, 1996, a hydrogen gas ignition occurred during the welding of the shield lid on 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.

³ On December 3, 1996, the NRC Staff informed the Licensee for the Arkansas Nuclear One facility in Russellville, Arkansas, that it had completed its reviews and inspections associated with that facility and found that the Licensee had satisfactorily completed the commitments documented in the CAL. Shortly thereafter, the Licensee initiated cask-loading activities.

the Licensee, prior to using an approved cask, to establish that all conditions in a dry storage cask certificate of compliance have been met (*see* 10 C.F.R. § 72.212(b)(2)).

1. Determine That the Licensee Violated NRC Requirements

In support of the petition's contention that the Licensee violated NRC requirements related to the original unloading procedure, the Petitioners claim that issues identified in Licensee documents dated November 11, 1994, and June 2, 1995, regarding revisions to the unloading procedure to support the planned unloading of Cask No. 4, demonstrate that the original procedure was inadequate. The amendment to the petition filed on September 30, 1996, included issues related to material compatibility identified in NRC Bulletin 96-04 as additional evidence that the Licensee's original unloading procedure was inadequate.

The primary information offered by the Petitioners in support of their claim that the original procedure violated NRC requirements is identified in the Licensee's document dated November 11, 1994. Although the issues identified by the Petitioners have been represented by the Licensee as improvements or enhancements to the original unloading procedure to support the planned unloading of Cask No. 4 at Palisades, a potential inference that might be drawn from the November 11 document is that the original unloading procedure could not adequately support the unloading of Cask No. 4. However, the Licensee's letter dated December 29, 1994, affirmed the Licensee's position that the original unloading procedure was adequate, and therefore complied with the certificate of compliance. Additional information, including the revised unloading procedure and the supporting engineering analyses, was provided in the Licensee's submittal to the NRC dated June 2, 1995. The NRC Staff requested additional information from the Licensee, and that information was provided by the Licensee in submittals dated October 16, 1995, December 20, 1995, and July 19, 1996.

On the basis of its review, the NRC Staff concluded that, had the Licensee attempted to unload a cask using the original unloading procedure, certain deficiencies associated with the original procedure would have prevented completion of the unloading process. The original unloading procedure's administrative limit for maximum cask pressure would have prevented the Licensee from establishing a continuous cooling cycle because the internal cask pressure would not have been sufficient to force steam to the outlet of the discharge piping at the bottom of the spent fuel pool. Other weaknesses in the original unloading procedure that would have hampered cask unloading included a restrictive venting capacity due to reliance upon a small vent line with an installed Swagelok fitting, scant guidance for personnel performing tasks such as drawing a gas sample from the MSB to check for damaged fuel, and several examples of references to

the wrong step within the procedure. Such deficiencies and weaknesses would have required the Licensee to suspend activities at one or more times during the unloading process in order to evaluate the problems encountered and implement necessary revisions to the procedure. Therefore, because the original unloading procedure would have required revision in order to complete the unloading process, this was a violation of requirements that all activities affecting quality be prescribed by procedures appropriate for the circumstances and that procedures are reviewed for adequacy. (See Criteria V and VI in Appendix B to 10 C.F.R. Part 50.)⁴ However, the Staff also determined that the deficiencies in the original unloading procedure would not have challenged the integrity of the cask or fuel contained in the cask and that the Licensee would have ultimately been able to safely unload a cask. Thus, given the limited safety significance of the procedural deficiencies and the fact that the Licensee identified and corrected the deficiencies, the NRC exercised its discretion to refrain from issuing a Notice of Violation or a civil penalty for the violation.

The purpose and objective of the NRC's enforcement program are focused on using enforcement actions (1) as a deterrent to emphasize the importance of compliance with requirements, and (2) to encourage prompt identification and prompt, comprehensive correction of violations. Mitigation of enforcement sanctions, such as refraining from issuing a civil penalty and/or a Notice of Violation, is described in section VII.B of the "General Statement of Policy and Procedures for NRC Enforcement Actions (Enforcement Policy)," for those cases in which a licensee identifies a problem and corrects it within a reasonable time. These mitigating factors were applicable to the subject Severity Level IV violation pertaining to the original unloading procedure at Palisades and the violation was, therefore, dispositioned as a Noncited Violation.⁵

As noted, the Licensee, in various correspondence, took the position that the original unloading procedure was adequate and that subsequent changes incorporated into the revised procedure were enhancements based on lessons learned from operating experience and additional evaluations. Several statements in the Licensee's correspondence appear to assert that unloading procedures for

⁴Section 1.1.3 of the certificate of compliance for the VSC-24 cask states that activities at the ISFSI shall be conducted in accordance with the requirements of 10 C.F.R. Part 50, Appendix B. Requirements related to quality assurance for ISFSIs are also contained in Subpart G to 10 C.F.R. Part 72. The requirements of Criteria V and VI in Appendix B to 10 C.F.R. Part 50 are the same as the requirements stated in 10 C.F.R. §72.150 and 10 C.F.R. §72.152. In the case of the original cask unloading procedure at Palisades, the number of problems in the original procedure and the failure of the Licensee to identify these problems during reviews performed prior to approval of the procedure resulted in the finding that a violation of NRC regulations had occurred. This finding is documented in NRC Inspection Report 50-255/96014.

⁵Although the NRC Staff has identified weaknesses and deficiencies in the unloading procedure developed by the Licensee, these problems resulted from the Licensee giving insufficient consideration to the complexity of the activity. As part of its evaluation pertaining to the mitigation of enforcement sanctions, the NRC Staff concluded that the Licensee had not knowingly and willfully violated NRC requirements related to having an unloading procedure for dry storage casks as was claimed by the Petitioners.

dry storage casks do not need to maintain fuel integrity during the unloading process in order to satisfy requirements of the certificate of compliance or NRC regulations. The NRC Staff disagrees with this interpretation. NRC requirements mandate that the unloading process should be developed with due consideration to maintaining fuel integrity (*see* 10 C.F.R. §§ 72.122(h), 72.122(l), and 72.236(h)). Unloading activities are required to prevent gross ruptures of the fuel cladding in order to prevent operational safety problems. Unloading procedures are also required to include contingencies in case fuel cladding has degraded during storage such that additional measures are necessary to address increased radiological hazards during the unloading process. The NRC Staff has concluded that the original unloading procedure would have supported unloading of undamaged fuel assemblies without causing a significant loss of fuel cladding integrity.

The issues identified by the Licensee in the document of November 11, 1994, and for which the Petitioners claim that the original unloading procedure was inadequate, are addressed below.

MSB Cooling Skid

The Licensee modified the configuration of the fill and vent piping and components from that used in the original unloading procedure. An increase in the venting capacity and the use of the previous vent path for instrumentation necessitated these modifications. The original unloading procedure included steps to remove a gas sample for analysis, connect the venting arrangement to the spent fuel pool, and connect the cooling water supply from the spent fuel pool to the vacuum drying system water pump and the MSB drain line. Neither the Petitioners nor the NRC Staff have identified fundamental safety concerns with the arrangement used in the original unloading procedure.

Thermal-Hydraulic Modeling

In order to verify that undamaged fuel could be safely removed from MSB No. 4 and to support preparing the revised unloading procedure, the Licensee performed multiple analyses by modeling the thermal-hydraulic behavior of the cask during the cooling process. These analyses were used to estimate the pressure response of the cask, to estimate the time requirements for cooling the cask, and to select the appropriate venting capacity in the revised unloading procedure. The analyses performed by the Licensee showed that the venting capacity available for the original unloading procedure would have supported the cooling and refill of the MSB. These analyses also showed that cask unloading using the original procedure would have taken significantly longer than the

time estimated for the revised procedure. However, no violations of regulatory requirements would have resulted from taking longer to complete the unloading process. The Licensee's performance of the analyses during preparation of the revised unloading procedure highlighted the lack of supporting analyses or evaluations for the original version of the unloading procedure and contributed to the Staff's finding that the Licensee had violated the requirements of Criterion VI of Appendix B to 10 C.F.R. Part 50 by issuing the original procedure without sufficient reviews to determine its adequacy.

Maximum Allowable Pressurization

During its review of the unloading procedure, the Licensee determined that the cask should be limited to 38.3 psig in order to satisfy criteria established by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. This value is conservative with respect to the pressure that would challenge the structural integrity of the MSB. The original unloading procedure included precautions to maintain the internal pressure less than 10 psig and thus was bounded by the subsequent evaluations and the acceptable conditions specified in the revised procedure.

However, the Staff has concluded that the procedural limitation of 10 psig in the original unloading procedure would have introduced problems in establishing the cooling cycle because the pressure would have been too low to force steam or water from the MSB to the coolant discharge at the bottom of the spent fuel pool. These problems, in turn, likely would have prevented completion of cask unloading without revising the procedure. However, the problems would not have challenged the integrity of the cask or otherwise introduced a safety concern. Rather, upon identifying the problems caused by the administrative limit of 10 psig, the Licensee could have revised the procedure, proceeded to establish the desired cooling cycle, and completed unloading of a cask.

Fuel Integrity During Cooling

In support of preparing the revised unloading procedure, the Licensee, with support from the nuclear fuel supplier, analyzed the allowable temperature differences between fuel assembly components and cooling water. Additional analyses determined maximum expected fuel temperatures before establishing the cooling flow to the MSB. These evaluations and the expected thermal response of the MSB and fuel assemblies following the introduction of coolant during the unloading procedure confirmed that thermal shocking would not challenge the integrity of the fuel assemblies in the MSB.

Fuel Heatup While the MSB Is in the Transport Cask

As previously mentioned, the Licensee and the contractors analyzed the maximum fuel temperatures that could be experienced during the time that the MSB is in the transfer cask before establishing the cooling flow from the spent fuel pool to the MSB interior. These analyses were performed for various heat loads and time periods and included conservative analysis assumptions. The analyses showed that fuel temperature limits would not be exceeded before establishing the cooling flow from the spent fuel pool using the original (or the revised) unloading procedure.

MSB Lid Removal

The revised unloading procedure uses more advanced cutting technologies in order to incorporate operating experience, ease lid removal, and minimize personnel exposure. The capability of the original unloading procedure to control removal of the MSB lid was verified by the Licensee during mockups before loading casks at Palisades. Some of the improvements in the revised procedure are related to problems experienced during that exercise. However, the Licensee has demonstrated that techniques for lid removal in the original unloading procedure were adequate to remove the lids and provide access to the fuel assemblies in compliance with NRC requirements.

Criticality Prevention

The original unloading procedure included steps for sampling the spent fuel pool boron concentration and establishing time limits for lid removal following termination of recirculation flow. The NRC Staff considers the original procedure's lack of a detailed contingency for preventing bulk boiling, as was incorporated into the revised procedure, a procedural weakness. However, the weakness does not translate into a concern related to public health and safety or personnel exposure because of the inherent conservatisms related to reactivity control for storage casks, such as assuming nonirradiated fuel assemblies in supporting calculations, and the time that would be available for the Licensee to implement compensatory actions.

Section 50.59 Evaluation Related to the MSB Cooling Skid

Modifications to the MSB cooling skid led the Licensee to question whether an unreviewed safety question was introduced by a possible break of the return line to the spent fuel pool. Upon further review, the Licensee determined that the cooling system configuration did not create the possibility for an

accident or a malfunction of a different type than any evaluated previously in the facility's final safety analysis report or otherwise exceed the criteria that define an unreviewed safety question under 10 C.F.R. § 50.59. The Licensee has stated that this conclusion is also applicable for the original unloading procedure. Neither the Petitioners nor the NRC Staff have identified a safety or compliance issue regarding the Licensee's conclusion.

Rigging Procedures

The Licensee investigated several minor changes to the rigging process during the development of the revised unloading procedure. These changes are intended to ease the operations and reduce personnel radiation exposures. However, the Staff determined that the guidance provided by the original procedure, combined with expected skill of Licensee personnel, would have been adequate to control the lifting of the various loads associated with unloading a cask.

Helium Sampling

During the development of the revised unloading procedure, the Licensee recognized possible difficulties in drawing a gas sample from the MSB before initiating the cooling operation. The original unloading procedure included a step to "remove a gas sample from the cask," but did not include the more detailed guidance that is incorporated into the revised procedure. This lack of guidance in the original procedure may have resulted in Licensee personnel underestimating the helium concentration in the MSB. The original unloading procedure included provisions to suspend the unloading process if the sampling indicated air within the MSB. Therefore, this potential weakness in the original unloading procedure would not have introduced adverse safety consequences but instead may have erroneously caused the Licensee to suspend cask unloading activities in order to conduct management briefings and determine compensatory measures due to the potential oxidation of the fuel cladding.

Summary for (1) "Determine That the Licensee Violated NRC Requirements"

On the basis of its evaluation of the Licensee's original unloading procedure, the NRC Staff affirmed the Licensee's determination that the procedure had numerous weaknesses. The Staff believes that the administrative limit of 10 psig for maximum cask pressure and other identified weaknesses in the original unloading procedure would have required the Licensee to suspend activities at one or more times during the unloading process in order to evaluate the problems encountered and implement necessary revisions to the procedure. Given the

number of weaknesses in the original unloading procedure and the Licensee's failure to perform the necessary levels of review and analysis to have determined its adequacy prior to its issuance, the NRC Staff found that the Licensee violated NRC requirements contained in Criteria V and VI of Appendix B to 10 C.F.R. Part 50. The first request in the petition, to find that the Licensee violated NRC requirements related to unloading procedures for dry storage casks for spent nuclear fuel, is therefore granted. The violation was dispositioned as a Noncited Violation consistent with the NRC Enforcement Policy.

The Petitioners' amendment to the petition dated September 30, 1996, claims that the original unloading procedure was inadequate because of its lack of controls related to the generation of hydrogen gas from a chemical reaction between coatings used on the VSC-24 casks and the borated water in the spent fuel pool. The chemical reactions and hydrogen issue were identified following an event that occurred during welding of the shield lid on a spent fuel storage cask at the Point Beach plant on May 28, 1996. The need to include special precautions in the unloading procedures for VSC-24 casks in order to prevent ignition of hydrogen gas had not been recognized by the cask vendor, licensees, or the NRC Staff prior to the event at Point Beach. The Licensee's original unloading procedure was developed before the event at Point Beach caused the recognition of the potential for ignition of hydrogen gas during the unloading of a VSC-24 cask. Accordingly, the NRC cannot reasonably fault the Licensee, by taking enforcement action, for not having accounted for an issue that was not known to the NRC Staff, the vendor, or the Licensee.

2. Suspend the Licensee's Use of the General License

On the basis of the contention that the Licensee's unloading procedure was inadequate, the Petitioners requested that the Licensee's use of the general license provisions of 10 C.F.R. Part 72 be suspended until such time as the significant issues described in the Licensee's document of June 2, 1995, have been resolved, the NRC has documented its review, approved the Licensee's revised procedure, and Cask No. 4 has been safely unloaded.

The Licensee's submittal of June 2, 1995, provided Revision 1 of the unloading procedure and supporting engineering analyses. The petition includes specific questions and comments regarding the Licensee's submittal of June 2, 1995, in support of the Petitioners' position that actions taken by the Licensee had not resolved significant safety issues. In response to questions from the NRC Staff, the Licensee provided additional information related to the submittal dated June 2, 1995. The subsequent submittals were dated October 16 and Decem-

ber 20, 1995.⁶ In addition, the NRC Staff was reviewing and will continue to review the issues included in the submittal dated June 2, 1995, as part of the ongoing NRC inspection of the revised unloading procedure. Further, as described above, the NRC Staff has already concluded that the deficiencies in the original unloading procedure violated NRC requirements, and that the violation should be treated as a Noncited Violation because of the limited safety significance of the procedural deficiencies and consideration of mitigating factors defined in the NRC Enforcement Policy.

On June 3, 1996, the NRC issued CALs to the Licensee and other users of the VSC-24 cask system. The CALs confirmed a commitment made by each licensee to the NRC Staff to refrain from loading or unloading a VSC-24 cask pending completion of investigations and implementation of corrective actions. On June 27, 1996, a supplement to the CAL was issued to confirm a further commitment by the Licensee to refrain from placing a VSC-24 cask into the spent fuel pool until after the NRC has reviewed and accepted applicable responses to NRC Bulletin 96-04 and verified corrective actions taken in response to the bulletin. CALs are among the administrative mechanisms that the NRC uses to supplement Notices of Violation, civil penalties, and orders in its enforcement program. CALs may be issued to confirm an agreement by a licensee or vendor to take certain actions to remove significant concerns about health, safety, safeguards, or the environment. The NRC expects licensees and vendors to adhere to stated obligations or commitments included in a CAL and will not hesitate to issue appropriate orders to ensure that such obligations or commitments are met.

The NRC issued the CALs and Bulletin 96-04 in recognition of the fact that the generation of hydrogen gas during the loading of VSC-24 casks at Point Beach was evidence that possible material compatibility issues were not fully addressed during the design or certification reviews associated with some spent fuel storage and transportation casks. It is not unusual for the NRC to use such administrative mechanisms to address generic issues. Given that the generation of flammable gases was a particular concern for the users of the VSC-24 cask system, those licensees, including Consumers Power Company, were issued CALs to confirm that VSC-24 casks would not be loaded, unloaded, or otherwise placed in a spent fuel pool before the resolution of issues identified in NRC Bulletin 96-04.

In regard to those issues contained in the amendment to the petition, the existing CAL documents the Licensee's commitment to refrain from loading, unloading, or otherwise placing a VSC-24 cask into the spent fuel pool pending

⁶These documents, like all others identified in this Decision, are available to the public at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and from the local public document room located in the Van Wylen Library at Hope College in Holland, Michigan.

verification of corrective actions related to NRC Bulletin 96-04. Given the Licensee's commitment not to load or unload a cask, the NRC does not, in this instance, envision the need to issue an order as requested by the Petitioners.

Those portions of the petition that address NRC's approval of the revised unloading procedure and include the unloading of Cask No. 4 as a condition for resuming normal activities under the general license are denied. The NRC Staff does not generally review and approve specific procedures developed by licensees. NRC regulations, facility licenses, and NRC-approved quality assurance programs require licensees to establish and maintain a formal process for the preparation and issuance of procedures and changes thereto. NRC assessments of Licensee procedures are generally conducted as part of the NRC's inspection program. In this instance, given the Licensee's commitment to refrain from action until completion of NRC's inspections, the inspections will confirm that applicable regulatory requirements are satisfied before use of the Licensee's revised unloading procedure. As previously mentioned, the NRC Staff will resume its inspection activities related to the revised unloading procedure when the Licensee has resolved the issues identified in NRC Bulletin 96-04. If, and provided that, there is satisfactory resolution of the issues identified in NRC Bulletin 96-04 and any other questions that may arise during the inspection of the Licensee's revised unloading procedure, then the NRC will have reasonable assurance of the Licensee's compliance with regulatory requirements. Accordingly, the Staff would not have any basis or reason to require the Licensee to unload Cask No. 4 before resuming normal activities under the general license at Palisades. Thus, following resolution of all issues to the satisfaction of the NRC Staff, the determination of the sequence of events related to the planned unloading of Cask No. 4 and the loading of additional casks at Palisades will be at the discretion of the Licensee. As noted above, the NRC Staff has committed to open the exit meeting with the Licensee to the public at the conclusion of the ongoing inspection and will document its review in an inspection report that will be available for public review.

3. Require the Licensee to Pay a Substantial Penalty

On the basis of the contention that the Licensee's original unloading procedure was inadequate, the Petitioners requested that the NRC levy a monetary penalty of \$1.3 million against the Licensee. As previously mentioned, the NRC Staff determined that, although finding that the deficiencies in the original unloading procedure violated NRC requirements, the violation satisfied the criteria to be treated as a Noncited Violation because of the limited safety significance of the procedural deficiencies and consideration of mitigating factors defined in the NRC Enforcement Policy. Enforcement sanctions, including issuance of civil penalties and orders, are normally used as a deterrent to emphasize the impor-

tance of compliance with requirements, and to encourage prompt identification and prompt, comprehensive correction of violations. In this case, the Licensee identified the deficiencies that constituted the violation of NRC requirements and subsequently revised the unloading procedure to resolve the identified technical issues. It was the judgment of the NRC Staff that the violation should be dispositioned as a Noncited Violation in order to convey the appropriate regulatory message in this case. Further, even if the violation had been cited, it is the NRC Staff's judgment that it would have been categorized as a Severity Level IV, for which a civil penalty would not ordinarily be issued.

In regard to the hydrogen issues identified in the amendment to the petition, the NRC Staff has utilized an administrative mechanism in its enforcement policy (CALs) to ensure that the Licensee takes certain actions to resolve this safety concern. As previously mentioned, the specific contentions raised by the Petitioners pertaining to hydrogen issues and the original unloading procedure do not warrant additional enforcement actions by the NRC.

4. Allow Petitioners to Review Procedure, Require NRC to Hold Hearings, and Allow Petitioners to Participate in Proceedings

The original unloading procedure and the first revision of the unloading procedure have been provided to the Petitioners. In addition, correspondence between the NRC and the Licensee regarding the procedures has been furnished to the Petitioners. Further, due to the course of events following the Licensee's decision to unload Cask No. 4 — including the Licensee's evaluation of the original unloading procedure, identification of improvements to the unloading process, and the submittal of this petition — the original and first revision of the unloading procedure and related documentation have been available for public review. Accordingly, Petitioners have had the opportunity to review the unloading procedure. Further, as noted elsewhere, it is the NRC Staff's intention to hold a public meeting in the vicinity of the Palisades Nuclear Plant at the conclusion of its ongoing inspection of the Licensee's revised unloading procedure.

The Petitioners' request for hearings and participation in proceedings has been addressed in previous correspondence with the Petitioners and the Attorney General for the State of Michigan. In that correspondence, the NRC Staff explained that neither the general licensing provisions of 10 C.F.R. Part 72 nor the petition process described in section 2.206 require the NRC to institute a proceeding. Under section 2.206, the NRC office director responsible for the subject matter of the request "shall either institute the requested proceeding in accordance with this subpart or shall advise the person who made the request in writing that no proceeding will be instituted in whole or in part, with respect to the request, and the reasons for the decision."

As set forth in this Director's Decision, the NRC has determined not to institute the proceeding as requested by the petition.

IV. CONCLUSION

Petitioners requested that the NRC determine that Consumers Power Company violated NRC requirements, suspend the Licensee's use of the general license, impose a substantial penalty, and hold hearings related to the Licensee's unloading procedure for dry storage casks. In response, the NRC determined that the Licensee violated NRC requirements insofar as the original unloading procedure (Revision 0) would have required revision in order to have completed the unloading process. Further, NRC Staff determined that the violation, which was identified and corrected by the Licensee, should be treated as a Noncited Violation consistent with the NRC's Enforcement Policy. Therefore, to this extent, Petitioners' request for a determination that the Licensee violated NRC requirements is granted. The available information is sufficient to conclude, however, that no substantial safety issue has been raised regarding the operation of Palisades or its associated ISFSI given the Licensee's commitment not to load or unload a cask until the NRC Staff is satisfied that the Licensee's procedures are adequate. Therefore, the NRC has determined that no adequate basis exists for granting Petitioners' requests for suspension of Consumers Power Company's use of the general license for dry cask storage of spent nuclear fuel at Palisades or imposition of a civil penalty.

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

As provided by this regulation, this Decision will constitute the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes a review of the Decision within that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

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

Dated at Rockville, Maryland,
this 23d day of January 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 70-3070-ML

LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center)

February 13, 1997

The Commission grants petitions filed by the Staff and Louisiana Energy Services for Commission review of the Atomic Safety and Licensing Board Partial Initial Decision, LBP-96-25, 44 NRC 331 (1996), and sets a briefing schedule pursuant to 10 C.F.R. § 2.786(d).

ORDER

The Nuclear Regulatory Commission Staff and Louisiana Energy Services (LES) have filed petitions for Commission review of the Atomic Safety and Licensing Board's December 3, 1996 Partial Initial Decision, LBP-96-25. 44 NRC 331 (1996). This proceeding involves LES's application for a license to construct and operate the Claiborne Enrichment Center (CEC) near Homer, Louisiana. The Intervenor, Citizens Against Nuclear Trash (CANT), opposes the petitions for Commission review. In accordance with the considerations set forth in 10 C.F.R. § 2.786(b)(4), the Commission has decided to grant the petitions and will review the issues raised in the Staff's and LES's petitions.

I. SCHEDULING OF BRIEFS

Pursuant to 10 C.F.R. § 2.786(d), the Commission sets the following briefing schedule:

1. The Staff and LES shall file their briefs on or before March 13, 1997. Each brief shall be no longer than 40 pages.
2. CANT shall file a single responsive brief on or before April 10, 1997. Its response shall not exceed 50 pages. We allow 50 pages for CANT's responsive brief so that CANT will have adequate space to respond to separate approaches that may be taken in the opening briefs of the Staff and LES.
3. On or before April 24, 1997, the Staff and LES may file reply briefs. Their replies shall not exceed 15 pages each.

To be timely, all documents must be served on the parties and on the Commission, so that they are received in the hands of the recipient no later than 4:15 p.m., Eastern Time, on the due dates for filing. Any means is permitted, including hand delivery, facsimile transmission, or e-mail. However, for service on the Commission, facsimile or e-mail transmissions shall be followed by a mailed original signed copy. Briefs in excess of 10 pages must contain a table of contents, with page references, and a table of cases (alphabetically arranged), statutes, regulations, and other authorities cited, with references to the pages of the brief where they are cited. Page limitations on briefs are exclusive of pages containing a table of contents, table of cases, and of any addendum containing statutes, rules, regulations, etc.

II. REMAINING ISSUES BEFORE THE BOARD

The Commission expects that the Board will be able to decide the remaining issues by May 1, 1997. If the Board cannot do so, the Board should advise the Commission and parties of an alternative, reasonable schedule for deciding these issues.

IT IS SO ORDERED.

For the Commission¹

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 13th day of February 1997.

¹Commissioners Dicus and Diaz were not available for the affirmation of this Order. If they had been present, they would have approved the Order.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

Peter B. Bloch, Presiding Officer
Peter Lam, Special Assistant

In the Matter of

Docket No. 55-20726-SP
(ASLBP No. 96-721-01-SP)
(Re: Operator License)

RALPH L. TETRICK
(Denial of Application for Reactor
Operator License)

February 28, 1997

The Presiding Officer determined that a reactor operator should be considered to have passed the written test for senior reactor operator.

He determined that one of the questions on the exam was ambiguous and should be disallowed. He also determined, in the absence of guidance from the Staff of the Commission, that examination scores are sufficiently imprecise that they should be rounded to the nearest integer. As a consequence, the score on the written examination was 80%, which the Presiding Officer considered a passing score. Since this was the last hurdle for the applicant in obtaining his license, the Presiding Officer directed the Staff to issue a Senior Reactor Operator's license to him.

INITIAL DECISION

Ralph L. Tetric, a reactor operator at the Turkey Point Nuclear Generating Plant, Units 3 and 4 ("Turkey Point"), operated by Florida Power & Light Company ("Florida Power"), is an applicant for a senior reactor operator's

(SRO's) license. On October 21, 1996, I granted Mr. Tetrick's request for a hearing concerning whether he had passed his SRO license examination.¹ An SRO is defined in 10 C.F.R. § 55.4 as "any individual licensed under this part to manipulate the controls of a facility *and to direct the licensed activities of licensed operators.*" (Emphasis added.)

The Nuclear Regulatory Commission (NRC) has jurisdiction of this request for a hearing, in which Mr. Tetrick appeals the result of his written examination. The NRC helps to assure the health and safety of the public by requiring reactor operators to successfully demonstrate their knowledge of nuclear power plant operation before they are licensed. See *Alfred J. Morabito* (Senior Operator License for Beaver Valley Power Station, Unit 1), LBP-88-10, 27 NRC 417 (1988), and LBP-88-16, 27 NRC 583 (1988); *Rodger W. Ellingwood* (Senior Operator License for Catawba Nuclear Station), LBP-89-21, 30 NRC 68 (1989).

The Commission's regulations in 10 C.F.R. §§ 55.43 and 55.45 require that an applicant for a senior reactor operator license pass both a written examination and an operating test. Written examinations taken by applicants for senior reactor operator licenses are developed and administered by the licensee, in this case Florida Power & Light Company, and are governed by 10 C.F.R. § 55.43. Written examination questions test "the knowledge, skills, and abilities needed to perform licensed senior operator duties." 10 C.F.R. § 55.43(a). In addition to information contained in a facility's training program, knowledge of "information in the Final Safety Analysis Report, system description manuals and operating procedures, facility license and license amendments, [and] Licensee Event Reports" may properly be tested. *Id.* Written examinations for senior operators include a representative sample of questions from fourteen subject areas specified for operator license applicants in 10 C.F.R. § 55.41(b)(1)-(14). In addition, written examinations for senior operators are to include a representative sample of questions from the seven areas specified in 10 C.F.R. § 55.43(b)(1)-(7).²

In addition to the written test, Mr. Tetrick took and passed the operating test, which involves a plant walkthrough and dynamic simulator evaluation during which various plant tasks, scenarios, and questions are presented to the applicants. See 10 C.F.R. § 55.45.

On the written examination, Mr. Tetrick was scored by the examiner as correctly answering 78 of 100 multiple-choice questions, for a score of 78%,

¹ This is an informal hearing under 10 C.F.R. Part 2, Subpart L. See 10 C.F.R. § 2.1201(a)(2). By letter of November 7, 1996, the NRC Staff ("Staff") submitted the Hearing File pursuant to 10 C.F.R. § 2.1231. On December 30, 1996, Mr. Tetrick filed his written presentation in this proceeding, pursuant to 10 C.F.R. § 2.1233 (Tetrick Presentation). Staff replied, pursuant to this same section of the regulations, on January 23, 1997 (Staff Presentation).

² See NUREG-1021, "Operator Licensing Examiner Standards," for further guidance on the administration and grading of the senior reactor operator written test.

which does not meet the 80% minimum score required to pass. See NUREG-1021, at 5 of 6. In response to Mr. Tetrick's request, the Staff completed an informal review that confirmed his failing grade. Hearing File item 21, attachment at 2-7.

Initially, Mr. Tetrick challenged the grading of Questions 24, 63, 84, and 96 on his examination. In its review, the Staff determined that Question 24 was invalid and should be deleted from the examination. However, the result of this determination was that Mr. Tetrick's score was raised only to 78.8%, which is short of the 80% required to pass. Mr. Tetrick continues to contest the scoring of his answers to Questions 63, 84, and 96 and he also is contesting the scoring of his answer to Question 90. Mr. Tetrick must be sustained in at least one of the four remaining challenges to pass the examination. Below, the challenges are considered one at a time.

I. QUESTION 63.

A. The Question

Examination Question 63 stated as follows:

Plant conditions:

- *Preparations are being made for refueling operations.*
- *The refueling cavity is filled with the transfer tube gate valve open.*
- *Alarm annunciators H-1/1, SFP LO LEVEL and G=9/5, CNTMT SUMP HI LEVEL are in alarm.*

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

- a. *Verify alarms by checking containment sump level recorder and spent fuel level indication.*
- b. *Sound the containment evacuation alarm.*
- c. *Initiate containment ventilation isolation.*
- d. *Initiate control room ventilation isolation.*

B. Staff Position

Staff contends³ that the correct answer to this question is “*b. Sound the containment evacuation alarm.*” It relies on Procedure 0-ADM-219, § 3.4.1

³ Affidavit of Brian Hughes and Thomas A. Peebles, January 23, 1997 (Staff Affidavit), Attachment 2 to Staff's Presentation, at 8, ¶ 20.

(Hearing File #20, attachment 2), which states: "Respond to alarms on color code priority *and plant conditions.*" (Emphasis added.) Staff argues that:

The plant conditions and indications specified in this question (*i.e.*, the refueling cavity filled and the transfer tube gate valve open with coincident SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms) *are mutually supportive and confirmatory*, and require entry into Off-Normal Operating Procedure 3-ONOP-033.2, "Refueling Cavity Seal Failure" (Hearing File #24⁴). [Emphasis added.]

Staff further argues that there is only one immediate action specified for a refueling cavity seal failure. That action, which the operator must be able to perform *from memory and before opening and reading the emergency procedures*, is to sound the containment evacuation alarm. Hearing File #24, 3-ONOP-033.2, at 5, § 4.1; Hearing File #25, 0-ADM-211, at 11, § 5.2.1; and Hearing File #25, 3-BD-EOP-E-O "BASIS DOCUMENT."

Staff stresses the importance of this immediate action. It states, in Staff Affidavit at 9, that:

Significantly, the need for such immediate action results from the fact that under the stated conditions, personnel located in the containment would quickly be exposed to high levels of radiation (due to loss of water which normally acts as a radiation shield) unless they are promptly notified by a containment alarm to evacuate the area.

Furthermore, Staff indicates that Off-Normal Operating Procedures have a high priority among plant operating procedures. Hearing Record #25, 0-ADM-211, at 25, § 5.13.1.

Staff also points out that the question explicitly asks for "the IMMEDIATE ACTION." Staff Affidavit at 10.

C. Mr. Tetrick's Position

Mr. Tetrick's answer was "*a. Verify alarms by checking containment sump level recorder and spent fuel level indication.*" He relies on the CONTROL ROOM ANNUNCIATOR RESPONSE procedure 3-ARP-097.CR to support his belief that, "The annunciators should be verified by additional supportive information to preclude the possibility of annunciator failure." Hearing File #20, discussion of Exam Question #63; *see also* Tetrick Request for Hearing, September 25, 1996.

⁴ Staff refers to "Item 24," which I have changed solely for the purpose of complying with the style used in this document.

D. Conclusion

The Staff has persuaded me that when two concurrent annunciators sound, indicating that there is an off-normal event that could cause harmful radiation within the containment, that the operator should take the required IMMEDIATE ACTION. Given the important safety problem that is being indicated by two different annunciators, that is not the time to verify that each of the annunciators is working properly. That they sound *together* is enough corroboration to act immediately to prevent injury to the health of plant employees.

Mr. Tetrick has had this Staff response available to him for some time and has never directly addressed it. In consequence, he continues to argue for an examination answer that could delay his action in preventing unnecessary exposure of his co-workers. I find that Mr. Tetrick's answer to this question was not correct.

I note, as well, that Mr. Tetrick is incorrect in stating that 3-ARP-097.CR states "that for all alarms the ARP shall be consulted." See the ARP at 8, "NOTES," at the bottom of the box. Step 2 in the notes requires that immediate corrective actions be taken as necessary. I interpret this to require that the immediate action of 3-ONOP-033.2 should be taken. The language quoted by Mr. Tetrick is from a bulleted paragraph that is part of paragraph "(3) Daily Annunciator Response Procedure Usage." I do not interpret that language to supersede or qualify in any way plant procedures that require immediate action.

II. QUESTION 84

A. The Question

Examination Question 84 stated as follows:

Which ONE of the following is the basis for step 1, "Verify Reactor Trip", of FR-S.1, Response to Nuclear Power Generation/ATWS?

- a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.*
- b. To ensure shutdown margin is within Technical Specifications limits for HOT STANDBY.*
- c. To alert the operator to take further corrective action if the reactor is NOT tripped.*
- d. To verify that all automatic reactor protective features have functioned as designed.*

B. Staff Position

Staff states that the correct answer is "a." Staff argues that the question requests the basis (or reason) for Step 1, Verify Reactor Trip, of FR-S.1,

Response to Nuclear Power Generation/ATWS. To determine the basis for Step 1, I first examine Step 1 in the following table:

Verify reactor trip:

- * Rod bottom lights — ON Manually trip reactor.
- * Reactor trip and bypass breakers — OPEN If reactor will *NOT* trip, *THEN* manually insert control rods.
- * Rod position indicators — AT ZERO
- * Neutron flux — DECREASING

Staff asserts that the reason or basis for this step (e.g., the reason the step is required) is: *“a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS [reactor coolant system].”* In support of this basis, Staff states that the reactor safeguard systems that protect the plant during an accident are designed on the basis that there are no additional sources of heat other than those mentioned in the correct answer, *a.* Staff Affidavit at 11-12, ¶¶26-27; Hearing File #20, “Page 9,” 3-BD-EOP-E-O, “Basis Document.”

C. Mr. Tetrick’s Position

Mr. Tetrick asserts that a correct answer to Question #84 is, “C. To alert the operator to take further corrective action if the reactor is not tripped.”

D. Conclusion

I conclude that the basis or “reason” for Step 1 has been correctly specified by the Staff as specified in File #20, 3-BD-EOP-E-O, “Basis Document.” Since the procedure correctly states the “basis,” a student could have answered correctly merely by learning what the procedure stated. The answer given by Mr. Tetrick is not the “basis” for Step 1. It is a followup action that might be taken after performing Step 1 but it is not the “basis” for that step.

III. QUESTION 90

A. The Question

Examination Question 90 stated as follows:

When draining the RCS using 3-OP-041.9, REDUCED INVENTORY OPERATIONS, the reactor vessel head and pressurizer are both vented to containment atmosphere.

Which one of the following describes the effects on reactor vessel indication if an adequate vent path is not provided? (Assume the reference leg remains full).

- a. A vacuum in the RCS loops will result in level indication being lower than actual levels.*
- b. A vacuum in the RCS loops will result in level indication being higher than actual levels.*
- c. A positive pressure in the RCS loops will result in level indication being lower than actual levels.*
- d. The level instruments automatically compensate for positive or negative pressure.*

B. Mr. Tetrick's Position

Mr. Tetrick's argument is simple. He states:

The assumption that the reference leg remains full makes this question invalid. At Turkey Point the drain down level indication has dry reference legs. This condition is verified by 0-PMI-041.110. Applicant requests that this question be deleted.

C. Staff Position

Staff states that the correct answer is:

- a. A vacuum in the RCS loops will result in level indication being lower than actual levels.*

Staff concedes that at Turkey Point the draindown level indication has a dry reference leg and that the assumption that the reference leg remains full is contrary to fact. Staff Affidavit at 15, ¶¶ 33, 35. Nevertheless, the Staff asserts that the question remains valid because "the fact that the reference leg is dry as opposed to filled with water is immaterial." Staff Affidavit at 17, ¶ 39.

The purpose of this question, according to the Staff, was to test an understanding of a basic hydraulic principle, that if a vacuum is drawn above the water level in the reactor pressure vessel, that will affect the instrument that measures water level because it will reduce the pressure exerted by the water in the pressure vessel.

The important leg to consider here is the variable leg of the water-level instrument. When there is a vacuum above the water in the pressure vessel, there will be less pressure on the variable leg than if the space above the water were filled by air at atmospheric pressure. The purpose of the "reference leg" of the pressure indicator is to measure the height of water that corresponds to

the pressure on the variable leg. Providing that there is no malfunction affecting the reference leg, it does not matter whether the design uses a wet or a dry reference leg. The answer will be the same: an accurate measurement of the height of the water in the variable leg. Staff Affidavit at 16-17, ¶¶ 37-39.

Staff states that:

38. This question tests applicants on their understanding of the hydraulic effects on level indication during mid-loop operations (*i.e.*, water level in the loop piping is less than full) and other draining operations if a vacuum is drawn while lowering water level. Numerous incidents have occurred within the nuclear industry which involved draining reactor coolant systems. A lack of understanding of the hydraulic effects on level indications by operators has been a prime contributor to many of these events. Therefore, it is important that applicants demonstrate an understanding of this problem, as examined on this question.

(Emphasis added.)

D. Conclusion

On this question, I agree with the Staff. The question is poorly worded, containing an assumption that is different from the plant configuration. This could have been somewhat confusing to Mr. Tetrick.

However, I have decided that if Mr. Tetrick had a basic knowledge of the principles that affect water-level indication, he should have realized that the entire purpose of the reference leg of the water-level indicator is to measure the height of water in the variable leg. Since the pressure exerted by the column of water in the variable leg would be reduced by the vacuum above the water in the reactor pressure vessel, *the water level indicated by the instrument would be lower than the water level in the reactor vessel.* Given the importance of this principle, I conclude that Mr. Tetrick should be able to understand it and answer the question correctly. There is no explanation for the answer he gave: that the water level indication would be higher than actual levels.

I conclude that, despite the contrary-to-fact predicate that makes this question more difficult than intended, Mr. Tetrick should have answered it correctly. The question is valid and Mr. Tetrick's answer is wrong.

IV. QUESTION 96

A. The Question

Examination Question 96 stated as follows:

Which ONE of the following is the lowest level position responsible for ensuring entries are made in the Technical Specification Related Equipment Out-of-Service Index?

- a. *Nuclear Plant Supervisor*
- b. *Assistant Nuclear Plant Supervisor*
- c. *Senior Nuclear Plant Operator*
- d. *Nuclear Watch Engineer*

B. Staff Position

Staff states that the correct answer is “*b. Assistant Nuclear Plant Supervisor.*” Staff states that

Procedure 0-ADM-213, “Technical Specification Related Equipment and Risk Significant S.C. Out-of-Service Logbook,” states that the ALPS is the lowest level position responsible for entering inoperable equipment in the subject index (Item 24). When the NWE [Nuclear Watch Engineer] relieves the ALPS, he then assumes the position of the ALPS. The NWE is not authorized to make entries in the subject index unless he is acting in the capacity of the ALPS, any more than he would be able to exercise any other functions of the ALPS unless he is acting in the ALPS capacity.

C. Mr. Tetrick’s Position

Mr. Tetrick states that “*d. Nuclear Watch Engineer*” is also correct because procedure 0-ADM-200 makes the Nuclear Watch Engineer (NWE) responsible “for routinely relieving the Assistant Nuclear Plant Supervisor (ALPS) of the control room command and control function to enable the ALPS to leave the control room.” [Emphasis added.] Staff does not question Mr. Tetrick’s statement that this substitution is authorized and routine.

D. Conclusion

I conclude that the question is ambiguous and should be struck.

Mr. Tetrick has reasonable ground to consider his answer to be correct. I do not think it necessary to address the following metaphysical question: Is the Nuclear Watch Engineer still at least in part a Nuclear Watch Engineer when he relieves the Assistant Nuclear Plant Supervisor? Staff apparently thinks that the Nuclear Watch Engineer completely loses his ordinary job identity when he acts as a substitute for the Assistant Nuclear Plant Supervisor. While that is a plausible way to view what happens, I do not think it fair to require Mr. Tetrick to adopt that view of the use of words in order to pass his examination. The question in its current form is ambiguous and invalid.

Mr. Tetrick has answered correctly 78 of 98 questions. His score, rounded to the nearest tenth of a percent is 79.6%.

I note that for the examination question to have the unambiguous meaning given to it by the Staff, it could have said: "*Which ONE of the following is the lowest level position that one must have (or be acting as) for ensuring entries are made in the Technical Specification Related Equipment Out-of-Service Index?*"

V. OVERALL CONCLUSION

I have determined that Mr. Tetrick was correct in 78 of 98 valid questions on his examination. Staff has not addressed the question of the number of digits in the examination score that should be considered significant. Because I have not been directed to any governing guidance or regulation, I have decided that it is appropriate to round up the answer to the nearest integer. These tests are not so precise that tenths of a percent have any meaning. Consequently, Mr. Tetrick's score is 80%, which is a passing score. He shall, therefore, be granted a license as a Senior Reactor Operator.

VI. ORDER

For all the foregoing reasons and upon consideration of the entire record in this matter, it is, this 28th day of February 1997, ORDERED that:

1. The Staff of the Nuclear Regulatory Commission may issue to Mr. Ralph L. Tetrick a Senior Reactor Operator License for Turkey Point Nuclear Generating Plant, Units 3 and 4.

2. Pursuant to 10 C.F.R. § 2.1251, this Initial Decision constitutes the final action of the Commission thirty (30) days after the date of issuance, unless any party petitions for Commission review in accordance with section 2.786 or the Commission takes review of the Decision sua sponte. If there is no petition for review, the date on which this Decision will become final is Monday, March 31, 1997.

3. Pursuant to 10 C.F.R. § 2.786, a petition for review must be filed within fifteen (15) days after service of this Decision, which is considered served on the date it is mailed, pursuant to 10 C.F.R. § 2.712(e). However, since service of this Decision is by mail, five days shall be added to the prescribed period of response, pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the petition for review must be served is Thursday, March 20. Service of the petition for review must, pursuant to this Order, be made by express mail.

4. A petition for review and a response to a petition for review must meet the requirements of 10 C.F.R. § 2.786.

5. If a petition for review is filed, the answer must be filed within 10 days. Since the petition for review shall be filed by express mail, two days shall be added to the period of response pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the answer must be served is Tuesday, March 16, 1997. Service of the answer must, pursuant to this Order, be made by express mail.

Peter B. Bloch, Presiding Officer
ADMINISTRATIVE JUDGE

Rockville, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Carl J. Paperiello, Director

In the Matter of

Docket No. 40-8989
(License No. SMC-1559)

ENVIROCARE OF UTAH, INC.

February 5, 1997

The Director, Office of Nuclear Material Safety and Safeguards, has denied a petition filed by Dr. Thomas B. Cochran on behalf of Natural Resources Defense Council (NRDC) requesting that the NRC take action regarding Envirocare of Utah, Inc. (Envirocare). The petition requested that the NRC immediately revoke any license or cause the State of Utah (Utah) to revoke any Agreement State license or licenses held by Envirocare, its President, Khosrow Semnani, or any entity controlled or managed by Mr. Semnani; prohibit the future issuance of any license by the NRC, Utah, or other NRC Agreement State to Mr. Semnani or any entity controlled or managed by him or with which he has a significant affiliation; and suspend Utah's Agreement State status until it can demonstrate that it can operate its Division of Radiation Control in a lawful manner. As a basis for the petition, the Petitioner asserted that an article in the *Salt Lake City Tribune* reported secret cash payments made by Mr. Semnani to the Director of the Utah Division of Radiation Control, and Utah's initiation of a criminal investigation into the matter. The reasons for the denial are set forth in the Decision.

**ATOMIC ENERGY ACT: ENFORCEMENT ACTION
(HEARING RIGHT)**

The Commission's regulations recognize that a licensee should be afforded under usual circumstances a prior opportunity to be heard before the agency suspends a license or takes other enforcement action, but that extraordinary circumstances may warrant summary action prior to hearing.

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING

Since the inception of the 10 C.F.R. § 2.206 process, the Commission has consistently stated that the purpose of 10 C.F.R. § 2.206 is to provide the public with the means for participating in the enforcement process.

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING

In accordance with the Commission's determination that the section 2.206 process should be focused on requests for enforcement action rather than an evaluation of safety concerns, petitions will be reviewed under 10 C.F.R. § 2.206 if the request is for enforcement action, and a request under section 2.206 should be distinguished from a request to deny a pending license application or amendment.

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

I. INTRODUCTION

In a letter dated January 8, 1997, Dr. Thomas B. Cochran, Director of Nuclear Programs, Natural Resources Defense Council (NRDC), requested, under 10 C.F.R. § 2.206 of the Commission's regulations, that NRC take action to revoke all licenses held by Envirocare of Utah, Inc. (Envirocare). Specifically, the petition requested that "NRC take the following actions:"

- 1) Immediately revoke the license or licenses, or cause the state of Utah to revoke its agreement state license or licenses, under which Envirocare is currently permitted to accept low-level radioactive waste and mixed waste for permanent disposal.
- 2) Immediately revoke the NRC 11.e(2) byproduct material license under which Envirocare is currently permitted to accept uranium mill tailings for disposal.
- 3) Immediately revoke any other NRC license, or agreement state license, if such license exists, held by Envirocare, Khosrow Semnani, or any entity controlled or managed by Khosrow Semnani.
- 4) Prohibit the future issuances of any license by the NRC, the State of Utah, or other NRC agreement state, to Khosrow Semnani or any company or entity which he owns, controls, manages, or [with which he] has a significant affiliation or relationship.
- 5) Suspend the agreement with the state of Utah under which regulatory authority has been transferred from the NRC to the Utah's [sic] Bureau of Radiation [Division of Radiation Control], until the state of Utah can

demonstrate that it can operate the Bureau of Radiation [Division of Radiation Control] in a lawful manner, and without the participation of licensees, or employees of licensees, in Bureau of Radiation [Division of Radiation Control] oversight roles.”

NRDC asserts, as a basis for the request, that a December 28, 1996 article in *The Salt Lake Tribune* reported that between 1987 and 1995, Mr. Semnani made secret cash payments to Mr. Larry F. Anderson, who served as Director of the Utah Division of Radiation Control (UDRC) from 1983 until 1993. The article also reported that the Utah Attorney General’s office has initiated a criminal investigation into the matter.

Although NRDC’s request that NRC suspend its agreement with the State of Utah, or cause Utah to revoke the license that it issued, does not squarely fall within the scope of matters ordinarily considered under section 2.206,¹ the Staff has evaluated the merits of those requests. This evaluation is contained in a separate “NRC Staff Evaluation of Natural Resources Defense Council Request to Suspend Section 274 Agreement with the State of Utah.” This Director’s Decision will address the NRDC requests that relate to the license to receive, store, and dispose of certain byproduct material issued to Envirocare by NRC, pursuant to section 11.e(2) of the Atomic Energy Act of 1954 (AEA), as amended.

II. BACKGROUND

Envirocare operates a radioactive waste disposal facility in Clive, Utah, 128 kilometers (80 miles) west of Salt Lake City in western Tooele County. Radioactive wastes are disposed of by modified shallow land burial techniques. Envirocare submitted its license application to the NRC in November 1989 for commercial disposal of 11.e(2) byproduct material, as defined in section 11.e(2) of the AEA. On November 19, 1993, NRC completed its licensing review and issued Envirocare an NRC license to receive, store, and dispose of uranium and thorium byproduct material. Envirocare began receiving 11.e(2) byproduct material in September 1994 and has been in continuous operation since.

To ensure that the facility is operated safely and in compliance with NRC requirements, the Staff conducts routine, announced inspections of the site. Areas examined during the inspections include management organization and controls, operations review, radiation protection, radioactive waste management, transportation, construction work, groundwater activities, and environmental

¹NRC Manual Directive 8.11, “Review Process for 10 CFR 2.206 Petitions,” issued September 23, 1994 (revised December 12, 1995), states that the scope of the section 2.206 process is limited to requests for enforcement action against licensees or entities engaging in NRC-licensed activities. *But see State of Utah* (Agreement Pursuant to Section 274 of the Atomic Energy Act of 1954, as Amended), DD-95-1, 41 NRC 43 (1995).

monitoring. The NRC has conducted five inspections of the Envirocare facilities and has cited the Licensee for three violations. All violations were categorized in accordance with the guidance in NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy) at a Severity Level IV.² The first violation, issued as a result of a July 1995 inspection, and the second violation, issued as a result of a July 1996 inspection, have been adequately resolved by Envirocare. The last inspection, conducted on November 18-22, 1996, resulted in the issuance of the third citation noted above. This violation involved a failure to develop and implement, in a timely manner: (1) site-specific standards for three constituents found in the groundwater that exceeded their baseline values, and (2) a Compliance Monitoring Plan for arsenic after it was found to exceed its baseline value. These results of the November 1996 inspection are documented in Inspection Report 40-8989/96-02 which was issued on January 28, 1997. The NRC is in the process of determining whether Envirocare has taken appropriate action to correct this violation.

In addition, the November 1996 inspection identified other areas of concern where the Staff determined that additional evaluation was necessary. As a result, a followup inspection was conducted the week of January 27, 1997. Areas that were examined during this inspection included: (1) the Licensee's quality assurance/quality control program; (2) the Licensee's review of changes made to the facility; and (3) contractor laboratory certification. The results of the January 27, 1997 inspection are currently being evaluated. Once this evaluation is complete, the NRC will document the results in an inspection report. Based on a preliminary review of the inspection results, no significant violations were identified.

III. DISCUSSION

In December 1996, the *Salt Lake Tribune* published a series of articles that questioned the relationship between Larry F. Anderson, former Director of UDRC, and Khosrow Semnani, President of Envirocare, during the licensing of the low-level radioactive waste (LLW) disposal facility. Subsequently, the NRC Staff learned that on May 16, 1996, Larry F. Anderson filed a complaint against Khosrow B. Semnani in the Third Judicial District Court of Salt Lake County, State of Utah, to obtain compensation for alleged consulting services in the sum of 5 million dollars. The complaint alleges that, while Director of UDRC, Mr. Anderson recognized the need for an LLW site in Utah; incorporated a consulting

² As explained in section IV of the Enforcement Policy, violations are normally categorized in terms of four levels of severity. A Severity Level IV violation is defined as a violation of more than minor concern which, if left uncorrected, could lead to a more serious concern.

firm, Lavicka, Inc., for the express purpose of developing a plan for siting the facility; and entered into a business arrangement to provide Mr. Semnani with a license application and consulting services. Mr. Anderson alleges that Mr. Semnani, President of Envirocare, agreed to pay a consulting fee of 100,000 dollars and an ongoing remuneration of 5% of all direct and indirect revenues that Mr. Semnani would realize from such a facility, if the site were successful. The complaint contends that Mr. Semnani owes Mr. Anderson unpaid compensation for consulting services in the sum of 5 million dollars.

In October 1996, Mr. Semnani filed a counterclaim in the court, denying Mr. Anderson's claim and alleging that, in fact, Mr. Anderson used his position as the Director of UDRC to extort money in the sum of 600,000 dollars. Mr. Semnani contends that all the money he paid was based on the belief that if he did not pay, Mr. Anderson would use his official position and capacity as an officer and employee of the State of Utah to deny Mr. Semnani fair consideration, review, hearing, and determination on his license application and, thereby, cause the license not to be granted, or, if Envirocare was granted a license, Mr. Anderson would use his position to subject the facility to unfair and biased oversight and supervision of the operation of the facility under the license. As a result of these allegations, the Utah Attorney General's office is investigating the relationship between Mr. Semnani and Mr. Anderson.

The NRDC petition is based on the events described above. The NRC has evaluated the NRDC's requests and found no basis to take the requested actions.

As an initial matter, NRDC requests that the NRC immediately revoke the NRC 11.e(2) byproduct material license under which Envirocare is currently permitted to accept uranium mill tailings for disposal. In addition, NRDC also asks that the NRC immediately revoke any other NRC license, or agreement state license, if such license exists, held by Envirocare, Khosrow Semnani, or any entity controlled or managed by Khosrow Semnani.

The NRC's Enforcement Policy describes the various enforcement sanctions available to the Commission once it determines that a violation of its requirements has occurred. In accordance with the guidance in section VI.C.3 of the Enforcement Policy, Revocation Orders may be used: (a) when a licensee is unable or unwilling to comply with NRC requirements; (b) when a licensee refuses to correct a violation; (c) when a licensee does not respond to a Notice of Violation where a response was required; (d) when a licensee refuses to pay an applicable fee under the Commission's regulations; or (e) for any other reason for which revocation is authorized under section 186 of the Atomic Energy Act (e.g., any condition that would warrant refusal of a license on an original application). Pursuant to 10 C.F.R. § 2.202(a)(5), the Commission may issue an immediately effective order to modify, suspend, or revoke a license if the Commission finds that the public health, safety, or interest so requires or that the violation or conduct causing the violation was willful. The Commission's regu-

lations recognize that a licensee should be afforded under usual circumstances a prior opportunity to be heard before the agency suspends a license or takes other enforcement action, but that extraordinary circumstances may warrant summary action prior to hearing. *See Advanced Medical Systems, Inc.* (One Factory Row, Geneva, Ohio 44041), CLI-94-6, 39 NRC 285, 299 (1994).

In this case the NRDC has not provided the NRC with specific information establishing that a violation of NRC requirements has occurred, nor provided the NRC with any other information that would provide a basis for immediate suspension of the Envirocare license. As NRDC notes in its request, the Utah State Attorney General has initiated a criminal investigation into the matter of the relationship between Mr. Anderson and Mr. Semnani. Absent specific information supporting the existence of such extraordinary circumstances as would warrant such action, NRC believes that it would be premature to initiate immediate action pending completion of this investigation. We recognize that this matter involves potential issues of integrity, which, if proven, may raise questions as to whether the NRC should have the requisite reasonable assurance that Envirocare will comply with Commission requirements. NRC intends to follow the investigation of the State Attorney General closely. If NRC receives information of public health and safety concerns during the investigation or on its completion, or receives such information from other sources, including NRC's ongoing Agreement State oversight activities, it will evaluate that information and take such appropriate action at that time as may be warranted.

Furthermore, the NRC Staff has reviewed the bases for its licensing actions involving Envirocare, and confirmed that NRC did not rely on technical evaluations performed by the State to reach a decision regarding the evaluation of Envirocare's 11.e(2) byproduct material license. The Staff conducted an independent technical evaluation of Envirocare's license application and subsequent amendment requests, and concluded that Envirocare had adequately demonstrated compliance with all applicable health and safety standards and regulations. In addition, as noted above, NRC inspections of Envirocare have not revealed significant violations that would warrant immediate action.

Moreover, with regard to NRDC's request that the NRC immediately revoke any other license, the NRC has issued no other license to Envirocare, Khosrow Semnani, or any entity controlled or managed by Khosrow Semnani. For these reasons, this request is denied.

NRDC also requests that the NRC prohibit the future issuances of any license by the NRC, the State of Utah, or other NRC agreement state, to Khosrow Semnani or any company or entity that he owns, controls, manages, or with which he has a significant affiliation or relationship.

With regard to this request, we have already noted that there is no basis for NRC to take immediate action. In any event, section 2.206 is not a venue for presenting licensing contentions of the sort raised by this aspect of NRDC's

petition. Section 2.206 provides for requests for action under that portion of the NRC's regulations governing enforcement actions, namely 10 C.F.R. Part 2, Subpart B. Subpart B is entitled "Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties." Since the inception of the section 2.206 process, the Commission has consistently stated that the purpose of section 2.206 is to provide the public with the means for *participating in the enforcement process*.³ The Commission has determined that the section 2.206 process should be focused on requests for enforcement action rather than evaluations of safety concerns. In accordance with this determination, the Commission's Management Directive 8.11, "Review Process for 10 C.F.R. 2.206 Petitions," Part III, section A, states that petitions will be reviewed under section 2.206 if the request is for enforcement action, and that a request under section 2.206 should be distinguished from a request to deny a pending license application or amendment.

Because this request by the NRDC concerns licensing-type action, not enforcement-type action, the Staff has determined that, consistent with the guidance of Management Directive 8.11, this request is not within the scope of section 2.206.⁴ To the extent that further facts may be developed that may warrant consideration of this request, the matter may be raised in an individual licensing proceeding; however, no such proceeding is presently pending, as there is no application pending for the issuance of a license to Envirocare.

IV. CONCLUSION

On the basis of the above assessment, I have concluded that no substantial health and safety issues have been raised regarding Envirocare that would require initiation of the immediate action requested by the NRDC, and the petition is therefore denied. As explained above, the NRDC has not provided any information in support of its requests of which the NRC was not already aware. Moreover, NRC inspections of the Envirocare facility have not revealed the existence of extraordinary circumstances that would warrant immediate suspension of the Envirocare license. In addition, the Staff's review of the technical basis for its issuance of the license and subsequent amendments found no evidence of the existence of any substantial health or safety issue that would justify the actions requested by the NRDC. NRC will monitor the investigations

³"Requests to Impose Requirements by Order on a Licensee, or to Modify, Suspend or Revoke a License," 39 Fed. Reg. 12,353 (April 5, 1974); "LeBoeuf, Lamb, Leiby & Macrae," 41 Fed. Reg. 3359 (Jan. 22, 1976); "Petitions for Review of Director's Denial of Enforcement Requests," 42 Fed. Reg. 36,239 (July 14, 1977).

⁴Even if this request were interpreted as a request that the NRC issue an enforcement order prohibiting Mr. Semnani from engaging in licensed activities, and thus constitute a request for enforcement action within the scope of section 2.206, NRDC has not provided the NRC with specific information such as would warrant the requested action, as explained above.

and actions being conducted by the State of Utah. If NRC receives any specific information that there is a public health or safety concern as a result of these actions or from any other source, including the NRC ongoing Agreement State oversight activities, NRC will evaluate that information and take such action as it deems is warranted at that time.

**FOR THE NUCLEAR
REGULATORY COMMISSION**

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

**Dated at Rockville, Maryland,
this 5th day of February 1997.**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Carl J. Paperiello, Director

In the Matter of

Docket Nos. 50-346
72-1004TOLEDO EDISON COMPANY, *et al.*
(Davis-Besse Independent Spent Fuel
Storage Installation)

February 5, 1997

The Director of the Office of Nuclear Material Safety and Safeguards grants, in part, and denies, in part, a petition filed pursuant to 10 C.F.R. § 2.206 on behalf of the Toledo Coalition for Safe Energy, Alice Hirt, Charlene Johnston, Dini Schut, and William Hoops. The petition is granted to the extent that the NRC has initiated a rulemaking to modify the Certificate of Compliance for the VECTRA Technologies NUHOMS-24P dry-shielded canisters (DSCs) in order to require fabrication inspection. The Petitioners' request that the NRC require the unloading of DSCs pending completion of the rulemaking is denied. The Director also finds no basis for taking any further enforcement action against VECTRA or to require the halting of the ISFSI operation at Davis-Besse.

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

By a petition dated December 5, 1995, filed on behalf of the Toledo Coalition for Safe Energy, Alice Hirt, Charlene Johnston, Dini Schut, and William Hoops (Petitioners),¹ the U.S. Nuclear Regulatory Commission (NRC) was asked

¹ According to the petition, the Toledo Coalition is a grassroots antinuclear organization with members who reside within a 35-mile radius of the Davis-Besse Nuclear Power Station. The petition indicates that it is also offering the positions of the Maryland Safe Energy Coalition, an organization represented to have members near the Calvert Cliffs nuclear plant, another site where NUHOMS-24P dry storage canisters are being used.

immediately to issue an order to prevent the loading of spent nuclear fuel into the VECTRA Technologies, Inc. (VECTRA), NUHOMS-24P dry-shielded canisters (DSCs) at the Davis-Besse Nuclear Power Station until the NRC conducts a rulemaking and/or license modification hearing on all safety-related changes that have been made to the DSCs, as described in the Safety Analysis Report (SAR). Also, the NRC was requested not to authorize any loading of the DSCs until a written procedure for unloading them, in both urgent and nonurgent circumstances, was written, approved, and field tested.

Petitioners contend that the safety of the DSCs has been compromised because of certain reductions that were made by VECTRA in the thickness of the welds in the DSC metal walls. In addition, Petitioners question the legal validity of the administrative and regulatory processes used by NRC after discovery of the DSC wall-thickness issue. They assert that an agency rulemaking or other public process is required for the DSCs at the Davis-Besse site.

The petition was referred to me pursuant to NRC regulations in 10 C.F.R. § 2.206.² Because the petition requested immediate relief (i.e., a halt to any loading of the DSCs at Davis-Besse), it was necessary for me to give an immediate response to that portion of the Petitioners' request. By letter dated December 18, 1995, I denied the Petitioners' request for immediate action on the petition on the basis of my judgment that there was (and continues to be) no imminent risk to health, safety, or environment such as to warrant the emergency relief sought by the Petitioners.³

By letter dated January 23, 1996, to Mr. Lodge, on behalf of the Petitioners, I formally acknowledged receipt of the petition. Notice of receipt was published in the *Federal Register* on January 30, 1996 (61 Fed. Reg. 3060).

Based on the NRC Staff's evaluation of the issues and for the reasons given below, I have now concluded that the Petitioners' request should be granted in part and denied in part.

BACKGROUND

NRC regulations contain a general license that authorizes nuclear power plants licensed by NRC, such as Davis-Besse, to store spent nuclear fuel at a reactor site in storage casks approved by NRC. *See* 10 C.F.R. §§ 72.210 and 72.212. Among

² Section 2.206 provides that "[a]ny person may file a request to institute a proceeding . . . to modify, suspend, or revoke a license, or for such other action as may be proper." The Director of the NRC office with responsibility for the subject matter of the request — in this case, the Office of Nuclear Material Safety and Safeguards — is to decide whether to institute the requested proceeding and, if no proceeding is instituted, will provide the reasons for the Decision.

³ My December 18 letter also notified Petitioners of my intention to treat their December 5 request as a petition under 10 C.F.R. § 2.206 and indicated that NRC would respond to the legal and technical issues they raised within a reasonable time.

other things, the Licensee is required to conform to certain NRC conditions for ensuring safe storage and to notify NRC at least 90 days prior to the first storage of spent fuel under the general license. By letter dated June 30, 1995, Toledo Edison Company (Licensee) informed NRC that it planned to use the VECTRA Standardized NUHOMS-24P dry spent fuel storage system (NUHOMS) under the general license at the independent spent fuel storage installation (ISFSI) facility at the Davis-Besse Nuclear Power Station. VECTRA's NUHOMS had previously been approved by NRC in December 1994 (59 Fed. Reg. 65,898) and as further reflected by the issuance of NRC Certificate of Compliance No. 1004 (COC) to VECTRA, the cask vendor. This NRC approval was granted after notice-and-comment rulemaking, to allow use of the NUHOMS system (subject to conditions specified in the COC) to store dry spent fuel at a nuclear power reactor site under the terms and conditions of the general license in 10 C.F.R. Part 72.

NRC regulations require cask vendors, such as VECTRA, to permit NRC to inspect the premises and facilities at which NRC-approved storage casks are fabricated and tested. *See* 10 C.F.R. §72.232. On June 20-23, 1995, NRC conducted an inspection of VECTRA's contractor, Ranor, Inc., at Westminister, MA. At that time, Ranor was fabricating the three NUHOMS DSCs and the transfer cask (TC) for VECTRA that were destined for Davis-Besse. The objective of the NRC inspection was to confirm that activities associated with the fabrication of the DSCs and TC had been executed in accordance with the requirements of the NRC COC and commitments made by VECTRA in the "Safety Analysis Report for the Standardized NUHOMS Horizontal Modular Storage System for Irradiated Nuclear Fuel" (SAR).⁴ VECTRA/Ranor was fabricating the DSCs and the TC for Toledo Edison (Davis-Besse site).

The NRC inspection identified three items of concern that required further action by VECTRA: (1) there was inadequate documentation to demonstrate that changes made by VECTRA/Ranor to the storage cask design described in the SAR had been reviewed and evaluated by the cask vendor in accordance with Condition 9 of the COC;⁵ (2) cask wall-thickness measurements had not been taken by VECTRA/Ranor after welding and grinding operations were performed

⁴ Under NRC regulations, a cask vendor who requests NRC approval of a spent fuel storage cask must submit an application that includes a SAR describing the proposed cask design and how the cask should be used to store spent fuel safely. *See* 10 C.F.R. §72.230.

⁵ The NRC report to VECTRA (*see* July 7, 1995 CAL) inaccurately described the corrective action as follows: "VECTRA will provide to the NRC written notification that the safety evaluations consistent with 10 CFR 72.48 have been completed and no unresolved safety issues were identified prior to shipping the DSCs and the TC." In fact, VECTRA was required to provide (and ultimately did provide) safety evaluations "consistent with Condition 9 of the COC." Condition 9 and 10 C.F.R. §72.48 are substantively similar in that each permits changes to the cask design described in the SAR, without prior NRC approval, if certain specified conditions are met and documented by a written safety evaluation. However, Condition 9 applies to changes by the cask vendor (i.e., VECTRA), whereas section 72.48 applies to changes by the licensee (i.e., Toledo Edison).

on the DSCs;⁶ and (3) leak testing was performed on the DSCs in lieu of pressure testing.⁷ On July 7, 1995, NRC issued a Confirmatory Action Letter (CAL) to VECTRA, confirming VECTRA's commitment to take actions to resolve the above three items of concern. Among those actions, as listed in the CAL, the following actions are related to Davis-Besse's ISFSI operation.

1. Regarding the finding of inadequate documentation of design changes, VECTRA was to review evaluations for adequacy and complete the documentation packages. VECTRA was to provide to the NRC written notification that the safety evaluations were completed and that no unresolved safety issues were identified prior to shipping the three DSCs and TC to Davis-Besse.
2. Regarding the finding on the lack of wall-thickness measurements after welding and grinding operations, VECTRA was to inspect welded areas in the DSCs to determine actual wall thickness and prepare an engineering document providing an evaluation of the safety significance of any wall thinning below design specifications. VECTRA was not to ship the three DSCs affected by wall thinning until this issue was resolved with NRC.⁸
3. Regarding the finding on performing leak testing instead of pressure testing, VECTRA was to provide to NRC an engineering evaluation justifying the use of a leak test in lieu of a pressure test. VECTRA was not to ship DSCs until this issue was resolved with NRC.

It is item 2 above — the absence of DSC wall-thickness measurements by VECTRA — that relates to the major issue of this petition.

As to item 2 of the CAL, on September 5, 1995, VECTRA informed NRC that the maximum thickness measured in the three DSCs prepared for Davis-Besse was 0.682 inch and occurred off the weld seam and in the base metal. VECTRA said that the minimum thickness measured in the three DSCs was 0.581 inch

⁶ VECTRA's NUHOMS design described in the SAR uses a nominal DSC shell thickness of 0.625 inch. However, VECTRA/Ranor had not measured the actual thickness of the fabricated DSC shells after welding and grinding operations to verify that it conformed to the description in the SAR.

⁷ As indicated in the SAR, the DSCs are designed, with one exception, as pressure vessels in accordance with the applicable sections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. The ASME B&PV Code calls for proof-pressure testing of the vessel. The one exception is the DSC top and bottom closure welds to which the ASME B&PV Code cannot practicably be applied.

⁸ The CAL also required VECTRA to evaluate the potential safety impact of the lack of wall-thickness measurements on previously fabricated DSCs which were shipped to sites other than Davis-Besse and to provide an engineering analysis and any recommended actions resulting from that analysis to NRC. By letter dated August 7, 1995, VECTRA submitted an action plan to address the issue related to those previously fabricated DSCs, such as those at the Calvert Cliffs site. Subsequently, VECTRA submitted information for Staff review in letters dated October 2, 1995, March 8, 1996, and April 25, 1996. The Staff evaluated the submitted information, and by letter dated January 3, 1997, informed VECTRA that the CAL issues were resolved and, therefore, closed. Given the followup activities of VECTRA and NRC already under way pursuant to the CAL and the absence of any additional information or claims in the petition relating specifically to Calvert Cliffs, I see no basis to take any further action at this time with regard to Calvert Cliffs.

and occurred in the weld seam of one of the DSCs. VECTRA also performed calculations that demonstrated that a DSC of 0.500-inch uniform wall thickness still met all ASME Code stress allowables, although the original design shell thickness in the SAR is 0.625 inch. In essence then, when it performed the required measurements of the three DSCs fabricated for Davis-Besse, VECTRA found actual, minimum wall thicknesses in each of the DSCs that were less than the 0.625-inch nominal thickness described in the SAR and a minimum thickness in one DSC of 0.581 inch. VECTRA thereafter went on to analyze whether a thinner wall design of 0.500 inch would satisfy NRC design criteria. The results of VECTRA's analysis submitted to NRC on September 5, 1995, showed that it would.

On October 12, 1995, NRC responded to the VECTRA actions taken in response to the CAL. Regarding item 2 of the CAL (the lack of wall-thickness measurements and VECTRA's subsequent September 5, 1995 reevaluation), NRC accepted VECTRA's 0.500-inch uniform wall-thickness calculation as meeting the ASME Code stress allowables, the original structural design criteria for the three DSCs. NRC said the structural capability of the DSCs would not be compromised if wall thinning from weld grinding were limited to local spots along weld seams and if the remaining shell thickness was 0.500 inch or more. However, NRC said that, because of the limited experience in performing weld-thickness measurements, "it is prudent to require a minimum weld inspection threshold thickness of 0.563 inches," to maintain a 0.063-inch fabrication margin over the 0.500-inch minimum. The NRC Staff prepared a safety evaluation dated October 5, 1995, documenting the basis for its acceptance of VECTRA's response to item 2.

NRC's October 12, 1995 response also found that VECTRA had acceptably addressed items 1 and 3 in the CAL. Thus, based on the actions taken by VECTRA and NRC's independent evaluation of the technical issues and review of the supplementary documentation provided, NRC found that VECTRA had acceptably completed the actions specified in the CAL and could, therefore, ship the three DSCs and the TC to the Davis-Besse site. VECTRA shipped the DSCs and TC to Davis-Besse shortly thereafter.

On November 14, 1995, one of the Petitioners (Ms. Charlene F. Johnston) wrote NRC asking for clarification on certain questions relating to the following issues: (1) whether an amendment process is required for the change in the wall thickness of the DSCs at Davis-Besse, and (2) whether the legality of a vendor's changes to a cask design can be questioned because the vendor is not a utility licensee and, therefore, cannot use the provisions of section 72.48 in making changes. Since the petition covers issues that are related to the two issues in the November 14 letter — and adds a third issue on cask unloading procedures — I have decided to include my response to the November 14 letter in this Decision.

DISCUSSION

The petition and associated November 14 letter raise three issues involving the DSCs at Davis-Besse. First, Petitioners contend that the reduction in the DSC shell thickness to less than 0.625 inch compromises the safety of the DSCs. Second, Petitioners question the legal validity of the administrative and regulatory processes used by NRC after discovery of the DSC shell-thickness issue and assert that an agency rulemaking or other public process is required for the DSCs at the Davis-Besse site. Finally, Petitioners contend that NRC should have reviewed and approved and field tested the procedure for unloading the DSCs both in urgent and nonurgent circumstances prior to the operation at the Davis-Besse site. In the following discussion, I will address each of these issues in turn.

A. Reduction of Shell Thickness Does Not Compromise the Safety of the DSCs

Petitioners claim that “the reduction in the thickness of the DSC metal walls to less than 0.625 inch compromises the safety of the DSCs.” Petition at 1. For the reasons that follow, I conclude that the change will not compromise safety. I begin by discussing the safety function of the DSC.

The DSC shell provides a key confinement barrier for the spent fuel stored inside the NUHOMS dry cask. Thus, the DSC shell helps to ensure safety for dry cask storage and protection of public health and safety by maintaining safe confinement of the stored fuel despite the forces, pressures, and stresses that are constantly acting on the cask (including the DSC shell) during normal handling, as well as during anticipated occurrences or potential cask accidents.

It is logical for Petitioners to conclude that, by reducing the thickness of the DSC shell, VECTRA could adversely impact the DSC’s capability as a safe confinement barrier. Indeed, it may seem obvious that a DSC having a shell thickness of 0.625 inch would have *more* capability to withstand cask bumps, drops, and pressure extremes than a DSC shell of reduced weld seam thickness no matter how small or limited the areas of thinning might be. Thus, at the core of Petitioners’ claim is the intuitive assertion that VECTRA’s change in the DSC shell thickness lessened the DSC’s capability as a confinement barrier to some extent. The question raised, but not answered, by the petition is whether this reduction in capability is sufficiently great to compromise safety. I conclude that it is not.

In NRC’s original evaluation, when it certified the NUHOMS and accepted VECTRA’s SAR in 1994, the NRC Staff reviewed a variety of potential cask accidents (e.g., a cask drop or tipover, vent blockage leading to cask heatup, low temperatures, earthquakes and tornadoes, explosions, lightning, floods) that

were thought to cover the range of cask accidents that might reasonably be assumed to occur. In the NRC review, the accident was assumed to occur (i.e., probability of occurrence was assumed to be one), and the consequences were evaluated. For each accident, the NRC Staff review found that the DSC would maintain confinement of the spent fuel without any breach or rupture of the DSC. Therefore, there could be no adverse impact on the public. As noted, the original NRC evaluation was based on a DSC nominal shell thickness of 0.625 inch.

In NRC's evaluation of the VECTRA September 5, 1995 submittal, which used a minimum DSC wall thickness of 0.500 inch to demonstrate a bounding case, the NRC Staff review assumed the occurrence of essentially the same range of accidents. Again, the NRC Staff found that the DSC would maintain confinement of the spent fuel without any breach or rupture of the DSC.

When VECTRA initially sought the NRC's 1994 approval of the NUHOMS, it provided design criteria for the DSC in the SAR as a basis for NRC approval of the NUHOMS system. VECTRA's proposed design criteria for the DSC were certain portions of the ASME BP&V Code.⁹ Materials (such as the materials that make up the DSC) have known stress values at which they will bend or break. During an accident, if the stresses acting on a vessel such as the DSC exceed those values, then it can be assumed that the material will fail. To facilitate the design process, the Code prescribes design criteria in the form of "allowable stresses" and requires that vessels such as the DSC must be analyzed under accident conditions to ensure that the stresses resulting from the accident do not exceed the allowable stresses of the materials used in the vessel. Depending on the likelihood of given design loading conditions, the Code builds into the design criteria and allowable stress values for each material a safety margin by setting generally the allowable stress at a fraction of the stress at which the material is known to bend or break.

⁹NRC approved the NUHOMS based on VECTRA's application and a supporting SAR which, in turn, pursuant to applicable NRC regulations, included appropriate design criteria for the storage cask. See 10 C.F.R. § 72.236(b). A vendor's design criteria in the SAR are important because they are to be used to analyze the acceptability of the vendor's proposed cask design against potential stresses on the cask after it is loaded with spent nuclear fuel. The stresses to be analyzed cover a variety of conditions that the cask may encounter during use, including those attributable to the dead weight or temperature of the spent fuel in the cask, internal pressures placed on the cask after it is loaded and sealed, normal handling of the cask during onsite transport or transfer, a potential handling accident such as a jammed canister when it is being placed in or retrieved from the storage module or a dropped cask during transport, seismic loads that arise from ground accelerations during an earthquake, postulated flood events, and stresses from certain load combinations.

The design criteria in the SAR submitted by VECTRA to NRC covered each of the cask conditions applicable to the proposed NUHOMS design (including the DSC). VECTRA also analyzed the NUHOMS design against these design criteria in the SAR, using a nominal DSC wall thickness of 0.625 inch, prior to NRC cask approval in December 1994. Further, and as detailed in the NRC Staff's Safety Evaluation Report (SER) which supported the rulemaking and ultimate approval of the VECTRA NUHOMS, the NRC Staff evaluated and accepted VECTRA's design criteria and analyses before issuing VECTRA the COC as part of the NRC's December 1994 approval.

VECTRA used the same ASME Code provisions for evaluating the DSC designs¹⁰ and demonstrated that the Code provisions were met by a DSC shell thickness of 0.500 inch.¹¹ Thus, even with the reduction in shell thickness, VECTRA demonstrated that the ASME Code provisions will be met by the DSC shell thickness of 0.500 inch.¹²

Therefore, I conclude that the reduction in shell thickness does not compromise the safety of the three DSCs at Davis-Besse. VECTRA has demonstrated that a DSC with a minimum shell thickness of 0.500 inch will provide safe confinement of spent fuel in the event of an accident.

VECTRA's revised structural analysis assumed that the entire DSC shell thickness, including all shell plating and weld lengths, had been reduced from 0.625 to 0.500 inch. This assumption by VECTRA resulted in a calculation that underestimated the strengths of the actual DSCs at Davis-Besse that were measured by VECTRA and found to have the specified 0.625-inch material thicknesses for nearly all of the shell weld lengths. Thus, the actual DSCs at Davis-Besse, with nonconforming weld thicknesses on only a portion of their weld lengths, should readily perform as well as VECTRA's revised structural

¹⁰ The design criteria used in VECTRA's reevaluation remained the same but the counting of the dead load effects differed in one respect from the SAR. In a request for additional information (RAI) dated August 17, 1995, the Staff commented that, "the deduction of dead weight (DW) from normal handling stress (Ln) load condition is a change in the design criteria used in the SAR." Later, in the October 5, 1995 Staff safety evaluation of VECTRA's revised calculation package and response to the RAI, the Staff noted that, "the calculation package considers the same design bases and criteria as those in the SAR." In the SAR, in analyzing certain load combinations, VECTRA had counted some dead weight stresses two times, whereas in the reevaluation of the DSC, it did not. The Staff agreed that the double counting in the SAR was unnecessary and, therefore, accepted the removal of the double counting in the revised analysis.

¹¹ As discussed previously, after the June 1995 discovery by NRC inspectors of the DSC wall-thickness issue, VECTRA was asked in the July 7, 1995 NRC CAL to provide an engineering analysis addressing the potential safety impact of the lack of wall-thickness measurements that covered casks in fabrication, most particularly the three DSCs destined for Davis-Besse. VECTRA chose to submit a revision to the structural analysis previously provided to NRC in the SAR, using a minimum DSC shell thickness of 0.500 inch, while considering the *same* design criteria as those in the SAR, which had been found acceptable by NRC for meeting NRC requirements including section 72.236(b). The Staff notes that, during the design process for components such as the DSCs, vendors commonly use conservative assumptions in their calculations to simplify the calculation process. (See NRC's SER §3.2.3.) Therefore, it can and should be expected that it may be possible to use an alternative method to perform design calculation (e.g., a more refined calculation that eliminates some of the conservative assumptions) to demonstrate that a different DSC shell design (e.g., a design that uses a thinner wall thickness) will also satisfy the design criteria embodied in the ASME Code. As discussed above, this is exactly what VECTRA did. That is, to resolve the wall-thickness measurement issue raised in the July 7, 1995 CAL, VECTRA performed a structural reanalysis of the NUHOMS. VECTRA reanalyzed the DSC with a uniform wall thickness of 0.500 inch, which is thinner than the nominal wall thickness of 0.625 inch used in the analysis originally provided by VECTRA in the SAR. Further, the structural adequacy of the DSC was demonstrated by comparing the calculated stress intensities for the 0.500-inch DSC shell to the same design criteria used for the 0.625-inch shell (i.e., ASME Code § III stress allowables).

¹² When the NRC Staff reviewed VECTRA's revised structural analysis submitted in September 1995 (i.e., the analysis demonstrating the structural acceptability of the DSC using 0.500-inch DSC shell thickness), the NRC Staff also relied on compliance with the same ASME Code provisions to establish the relevant design criteria for determining whether the 0.500-inch DSC shell design would provide the required safety. Specifically, in its safety verification of the VECTRA calculation package (NUH004.0213, "Standardized NUHOMS-24P DSC Shell Minimum Acceptable Uniform Thickness"), the NRC Staff concurred that all calculated stresses for the 0.500-inch DSC shell thickness are acceptable. (See NRC Letter to VECTRA, dated October 12, 1995.)

analysis predicted. In either case, the affected casks will perform in accordance with the pertinent ASME Code requirements, the operative design standard inherent in the NRC Staff's approval. This level of performance provides reasonable assurance that public health and safety will be protected.

Thus, while VECTRA failed to comply with its SAR commitment of 0.625 inch, its failure resulted in no compromise of safety. Nonetheless, the failure raised an issue of poor control during the fabrication process. This deficiency was identified by NRC during the June 1995 inspection; and VECTRA was cited for it in the NRC Notice of Nonconformance issued to VECTRA in August 1995.¹³

B. Rulemaking Should Be Conducted to Propose Changes to the NUHOMS Certificate in Light of the Weld-Thinning Issue, and Petitioners' Claims Can Be Made in That Rulemaking

Petitioners question the legal validity of the administrative and regulatory processes used by NRC after discovery of the DSC wall-thickness issue. Petition at 1. Specifically, Petitioners believe an NRC rulemaking (or other public proceeding) should have been held.

As set forth above, my conclusion is that the DSCs at Davis-Besse are safe. However, as I will explain below, I believe an issue remains as to whether NRC should take some additional action with respect to VECTRA's COC for the NUHOMS cask.

I have already referenced the NRC's action with respect to VECTRA's failure to conform to NRC requirements. In particular, the fabrication process for the DSCs did not ensure that acceptable DSC wall thickness was maintained as required by NRC. The process included an instruction that the operator manually flush-grind the welds, after welding the DSC shell seams. However, there was no procedure that provided an adequate level of control in maintaining minimum acceptable wall thickness. Moreover, under the procedures, the operator did not measure the final wall thickness of the DSC in the area of the welds after grinding. Further, measurements were not taken in any subsequent steps in the fabrication process to ensure that minimum wall thickness was

¹³ Petitioners' November 14 letter asserts that VECTRA violated NRC regulations when it failed to do measurements on DSC wall thickness and weld seams during fabrication. NRC's June 1995 inspection of VECTRA/Ranor and NRC's August 1995 Notice of Nonconformance to VECTRA have already indicated that VECTRA failed to conform to NRC regulations. The Petitioners' November 14 letter also questions whether VECTRA may have willfully failed to report a nonconformance or deviation in wall thicknesses for the DSCs. The NRC inspection did not identify any indications of a willful failure to report. Rather, the failure on the part of VECTRA/Ranor was its failure to have adequate quality control measures in place during the fabrication process to measure DSC welds after grinding. It appears that VECTRA/Ranor did not anticipate that grinding the weld could result in going below the specified plate thickness. Therefore, the Petitioners' concern about a possible willful failure to report a nonconformance cannot be substantiated.

maintained. VECTRA thus failed to ensure conformance to NRC's requirement that activities affecting quality must be prescribed by appropriate, documented instructions, procedures, or drawings that include criteria for determining that important activities have been satisfactorily accomplished. See 10 C.F.R. § 72.150, "Instructions, procedures, and drawings." As a consequence, NRC issued VECTRA a Notice of Nonconformance on August 29, 1995, citing VECTRA for its failure.¹⁴

Petitioners, however, seek additional action. Specifically, in their December 5, 1995 petition, Petitioners state that they believe that an NRC rulemaking (or other public proceeding) was required to permit use of the three DSCs with wall thinning at Davis-Besse. Further, in their related November 14, 1995 letter, Petitioners question whether an NRC rulemaking was required because VECTRA's change of the three DSCs to a wall thickness of less than 0.625 inch involved a reduction in "the margin of safety" that must be approved by an NRC amendment process.

Petitioners' November 14 questions appear to be aimed at VECTRA's implementation of Condition 9 in the NRC COC issued to VECTRA in December 1994.¹⁵ Condition 9 permits VECTRA to make changes in the DSC design without NRC approval provided, among other things, the change does not involve "an unreviewed safety question." Condition 9 states that a change shall be deemed to involve an unreviewed safety question "[i]f the margin of safety as defined in the basis for any technical specification or limit is reduced." After evaluation, VECTRA concluded that the wall thinning of the three DSCs at Davis-Besse did not involve a reduction in the "margin of safety" or "an unre-

¹⁴ NRC's Notice of Nonconformance cited VECTRA for several other nonconformances with NRC requirements unrelated to DSC wall thickness or the petition.

¹⁵ In their November 14, 1995 letter, Petitioners questioned VECTRA's legal authority to make changes to the DSC. In 1990, to fulfill the mandate of the Nuclear Waste Policy Act of 1982, the NRC amended 10 C.F.R. Part 72 so as to put in place the regulatory procedures that authorize a nuclear power reactor licensee to store spent fuel on site under a general license without the need for an additional site-specific Commission approval. 55 Fed. Reg. 29,181 (1990). To use the general license, the reactor licensee must store the spent fuel in a cask that has been certified under the provisions of 10 C.F.R. Part 72, Subpart K. See 10 C.F.R. § 72.212(a)(2). A vendor who meets the Subpart K requirements will be issued a COC by the NRC, and, after a public rulemaking proceeding, the cask will be added to the list of approved spent fuel storage casks at 10 C.F.R. § 72.214.

These regulatory procedures were used with respect to NRC's approval of VECTRA's NUHOMS system. On June 2, 1994, the NRC published a proposed rule adding the NUHOMS system to the approved list and gave notice that the draft COC was available for inspection and comment at the NRC Public Document Room. 59 Fed. Reg. 28,496 (1994). As Petitioners are aware, Condition 9 of the COC provides to the holder of the COC the same type of authority to make changes as is provided to licensees under section 72.48. Condition 9 provides to VECTRA, among other things, the authority to make changes in the cask design described in the SAR without prior Commission approval unless the proposed change involves a change in the COC, an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact. The Commission received a number of positive and negative comments, including comments from some of the present Petitioners, on its proposal to incorporate section 72.48 type language into Condition 9 of the COC but determined to retain this language in the final rule. See 59 Fed. Reg. 65,898, 65,914-15 (1994). Condition 9 was adopted through notice-and-comment rulemaking, and VECTRA was entitled to utilize its provisions in considering changes to the cask design as described in the SAR.

viewed safety question.” By asserting that an NRC rulemaking was required, Petitioners may be effectively arguing that I should find these VECTRA conclusions to be wrong.¹⁶ However, it is not necessary to evaluate VECTRA’s conclusions in order to decide that Petitioners’ request for NRC rulemaking should be granted with respect to the wall-thinning issue. As I explain below, I believe that rulemaking should be undertaken for different reasons.

In this regard, I note that the NRC Staff’s October 5, 1995 SER (issued when Staff accepted VECTRA’s analysis of a minimum DSC shell wall-thickness of 0.500 inch) includes the conclusion that “it is prudent to require” a minimum weld inspection threshold thickness. As part of its response to the CAL and to address the nonconformance with NRC requirements described above, VECTRA had proposed an inspection procedure to ensure that DSC weld-grinding operations do not result in wall thinning below acceptable levels. Staff viewed (and continues to view) VECTRA’s proposed inspection procedure, which invokes enhanced actions if grinding operations exceed a 0.563-inch threshold, as an acceptable quality control practice. Further, it was Staff’s intent in the SER to reflect VECTRA’s inspection plan as an important consideration in Staff’s acceptance of VECTRA’s response to the CAL.

However, although VECTRA implemented the inspection procedure as to the three Davis-Besse DSCs and committed to use it in fabricating future DSCs and although the NRC Staff’s SER expressly relied on the VECTRA inspection procedure as a consideration in accepting VECTRA’s response, nothing in the VECTRA COC explicitly requires VECTRA to conduct inspections during fabrication of the DSC. Thus, one purpose of rulemaking would be to consider whether these (and possibly other) circumstances of the NUHOMS wall-thinning issue justify the step of putting a fabrication inspection requirement in the VECTRA COC. Specifically, rulemaking could propose to amend the VECTRA COC to require that, in the fabrication of the DSC, the shell and basket assembly must be inspected to ensure that structural design margins, associated with the ASME Code § III stress allowables, are not compromised. Such a requirement would serve the purpose of helping to ensure that the DSC fabrication process, including weld-grinding operations, produces DSC components that conform to the design criteria and safety margins approved by NRC.

¹⁶ An NRC inspection team found VECTRA’s safety analysis for the wall-thinning issue to be in administrative compliance with Condition 9. The technical aspects of VECTRA’s safety analysis were not reviewed by the team. See NRC Inspection Report No. 72-1004/96-207. I should note that NRC policy in this area might undergo clarification. The regulatory language in Condition 9 is similar to language in 10 C.F.R. § 50.59, a separate and unrelated provision involving nuclear reactors. NRC is conducting an internal review of its policy guidance on identifying “unreviewed safety questions” in the context of section 50.59. I intend to monitor that review and, when it is complete, consider whether there is a need to develop clarifying guidance for Condition 9, as well as section 72.48 which governs changes by Part 72 licensees.

At this point, I am inclined to believe that VECTRA's COC should be modified in light of the weld-thinning issue.¹⁷ As discussed above, I believe that changes to VECTRA's COC merit consideration as possible additional actions to ensure the quality of VECTRA's NUHOMS components in light of the history of this matter. Further, rulemaking would allow us to receive and consider comments of the Petitioners and other members of the public who are interested in the weld-thinning issue. As part of the rulemaking, NRC could include in the record the entire NRC Staff safety evaluation of VECTRA's wall-thickness reevaluation and the VECTRA reevaluation itself submitted in response to the NRC July 7, 1995 CAL. As noted, the Staff's safety evaluation and the acceptability of VECTRA's reevaluation both depended, in part, on a VECTRA inspection that the rulemaking would propose to require in VECTRA's COC.

In the rulemaking, as I envision it, Petitioners, as well as any other interested member of the public, would be given the opportunity to comment on any aspect of the NRC safety evaluation associated with this issue. At the conclusion of the comment period, NRC would consider all comments and provide a response. Further, if NRC determined, after considering the comments, that it should modify VECTRA's COC or change the Staff's previous determination to accept VECTRA's 0.500-inch uniform wall-thickness calculation, the rulemaking would provide a vehicle for it to do so.

This course of action, which I intend to pursue, would provide Petitioners the agency rulemaking they seek on the reduction in the thickness of the DSC metal walls to less than 0.625 inch, and it will provide them the opportunity to examine and comment on NRC's determination that the safety of the DSCs has not been compromised and to submit such other information as they wish on any aspect of the wall-thickness issue. Therefore, to this extent, I am determining that the petition should be granted.

I have also considered whether NRC should take some additional action, pending completion of the rulemaking, with respect to the three DSCs now in service at the Davis-Besse site. In Part A of this discussion, I set forth the basis for my conclusion that the reduction in the shell thickness of the DSCs at Davis-Besse does not compromise their safety. Therefore, I believe that continued storage of spent fuel in the DSCs, pending completion of the rulemaking, would not pose an unreasonable risk to public health and safety and that there is no technical basis to require their unloading. Further, as I have previously summarized in this Part B, NRC already cited VECTRA for its failure to comply with NRC requirements in August 1995. Accordingly, to the extent Petitioners seek additional action, pending completion of the rulemaking, their request is denied.

¹⁷ Under NRC internal procedures, the Staff must request and obtain Commission approval before undertaking rulemaking. Therefore, I intend to seek Commission approval to do so.

C. There Is No Basis to Grant Petitioners' Request That NRC Review, Approve, and Field-Test Procedures for Unloading DSCs Prior to Operation

Petitioners also present claims concerning the unloading of the casks at Davis-Besse. Specifically in this regard, they demand that "no loading of the canisters be authorized until there is in place a written, approved, and field-tested procedure for unloading the DSCs both in urgent and nonurgent circumstances." Petition at 1-2.

There is no regulatory requirement for NRC to review, approve, and field test a licensee's operating procedures, including unloading of spent fuel casks under urgent and nonurgent circumstances. Rather, NRC's approach is to require that licensees have a formal process for procedure development and control. Generally, in the analogous case of a power reactor, this process is part of the facility license. NRC oversees the implementation of that process and the product (i.e., the procedures and their use) through its inspection program. This approach to overseeing licensee operations has been effectively demonstrated by successful startup of power reactors following construction and the continued safe operation of existing facilities. NRC expects that a general licensee under Part 72 will prepare ISFSI procedures in accordance with its established procedure development process and as required by its quality assurance program.

As a general licensee, Toledo Edison is required to comply with the terms and conditions of the COC issued for the VECTRA NUHOMS-24P. *See* 10 C.F.R. § 72.212(b). The applicable conditions in the COC can be found in section 1.1.2 which requires that "[w]ritten operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance." This condition is broadly written and interpreted by NRC to require the licensee to have detailed written procedures for loading and unloading a DSC. Another related condition in the COC appears in Section 1.1.6 which requires "[p]re-operational testing" that includes, but is not limited to, a dry run of loading and unloading a DSC. Thus, it is the licensee's responsibility to prepare, review, approve, and test written procedures for cask loading and unloading. Further, NRC requires a licensee to conduct activities related to ISFSI operation, including cask loading and unloading, in accordance with those written procedures once the licensee has approved, tested, and put procedures in place. *See* 10 C.F.R. § 72.212(b)(9). The NRC also conducts periodic audits of these activities through its inspections program.

It is not NRC's practice to review and approve a licensee's operating procedures. It is important to understand that, just with respect to dry cask storage activities, which are a very small fraction of the daily activities conducted at an operating nuclear power plant, the applicable written procedures of a general licensee are likely to be voluminous. Moreover, the written procedures

prepared by a licensee typically are site-specific in nature and thus reflect the licensee's special knowledge of its plant and how dry cask storage activities interconnect with plant personnel, as well as other plant activities and procedures. The written procedures are prepared according to the formal procedure development process and exercised during the dry run. In my judgment, there would be very little additional value to be gained from a requirement of NRC review and approval of a licensee's written operating procedures, particularly given our existing inspection activities illustrated by the Davis-Besse example, below.

In particular, with regard to Davis-Besse's ISFSI operation, the Licensee developed written operating procedures for dry cask handling including loading and unloading procedures. These procedures were used by the Licensee for the preoperational dry-run testing at the Davis-Besse plant during November 30 through December 11, 1995. The NRC Staff inspectors were present at the plant throughout the testing, conducted an onsite observation of the Licensee's dry-run loading and unloading activities, and also inspected the detailed written procedures used by the Licensee for cask loading and unloading. NRC Inspection Report 50-346/95-09 documents the extensive NRC inspection activities, as well as the inspection finding that the dry-run activities were conducted satisfactorily and in a safe manner. Therefore, based on the circumstances reflected in the foregoing discussion, I conclude that there is in place at Davis-Besse an adequate written procedure, approved and field tested by the Licensee, for unloading the DSCs if needed, and that the Petitioners' request — to the extent it seeks further NRC review and approval — should be denied.

CONCLUSION

As discussed above, VECTRA's change to the wall thickness of certain weld seams does not compromise the safety of the three DSCs at Davis-Besse. However, the NRC COC for VECTRA's Standardized NUHOMS-24P should be modified to require a fabrication inspection of the DSC. An agency rulemaking is, therefore, needed and should be conducted to accomplish this modification. In rulemaking, Petitioners would have the opportunity to comment on any aspect of the DSC wall-thickness issue. However, because the continued storage of spent fuel at the DSCs at Davis-Besse does not pose an unreasonable risk to public health and safety, I find no technical basis to require the DSCs to be unloaded pending completion of this rulemaking. Further, VECTRA has already been cited for a nonconformance with NRC regulations, and I find no basis in the petition to take other action in this regard.

Toledo Edison has developed loading and unloading procedures for handling spent fuels. These procedures have been applied for the dry-run testing with

NRC's oversight. Therefore, I find no basis in the petition for requiring halting of the ISFSI operation at Davis-Besse.

Accordingly, the petition from Toledo Coalition for Safe Energy is granted to the extent that it requests an agency rulemaking and is denied in all other respects.

**FOR THE NUCLEAR
REGULATORY COMMISSION**

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

Dated at Rockville, Maryland,
this 5th day of February 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of

Docket No. 50-245
(License No. DPR-21)

NORTHEAST NUCLEAR ENERGY
COMPANY
(Millstone Nuclear Power Station,
Unit 1)

February 11, 1997

The Acting Director, Office of Nuclear Reactor Regulation, has granted in part and denied in part a petition filed by Anthony J. Ross requesting action regarding Millstone Nuclear Power Station, Unit 1. The Petitioner requested that the Commission take escalated enforcement action against the Licensee and certain individuals based upon the deliberate failure to comply with procedures involving sign-out of measuring and test equipment, and conduct an investigation into alleged procedural violations and audit the Millstone Unit 1 maintenance department Measuring and Test Equipment folders for widespread problems regarding procedural noncompliance. To the extent that the Petitioner requested escalated enforcement action be taken, the petition has been denied; to the extent that the Petitioner requested an investigation into the procedural violations and an audit, the petition has been granted.

ENFORCEMENT POLICY: SEVERITY OF VIOLATIONS

Minor violations, as described in the current enforcement policy, are not the subject of formal enforcement action and are usually not cited in inspection reports. To the extent that such violations are described, they are now noted as noncited violations.

RULES OF PRACTICE: INSTITUTION OF PROCEEDINGS UNDER 10 C.F.R. § 2.206

The institution of a proceeding pursuant to 10 C.F.R. § 2.206 is appropriate only if substantial health and safety issues have been raised.

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

I. INTRODUCTION

On January 5, 1995, Mr. Anthony J. Ross (Petitioner) filed a petition with the Executive Director for Operations of the Nuclear Regulatory Commission (NRC) pursuant to section 2.206 of Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206). In the petition, the Petitioner raised concerns regarding noncompliance with Procedure WC-8, "Control and Calibration of Measuring and Test Equipment," at Millstone Nuclear Power Station, Unit 1, and requested that escalated enforcement action be taken. Specifically, the Petitioner provided several examples of what he alleged were violations of Procedure WC-8, which he stated required that measuring and test equipment (M&TE) be signed out from, and returned to, a custodian upon completion of work. The Petitioner requested that the NRC institute sanctions against his department manager, his first-line supervisor, and "two coworkers"¹ for engaging in deliberate misconduct in violation of 10 C.F.R. § 50.5 in failing to comply with Procedure WC-8. The Petitioner also asserted that the NRC should conduct an investigation into violations of this procedure and audit the Millstone Unit 1 maintenance department M&TE folders for widespread problems regarding noncompliance with this procedure.

On February 23, 1995, the NRC informed the Petitioner that the petition had been referred to the Office of Nuclear Reactor Regulation pursuant to section 2.206 of the Commission's regulations. The NRC also informed the Petitioner that the Staff would take appropriate action within a reasonable time regarding the specific concerns raised in the petition. On the basis of a review of the issues raised by the Petitioner, as discussed below, I have concluded, for the reasons explained below, that the petition is denied with regard to the request for escalated enforcement action and instituting sanctions against the department manager, first-line supervisor, and two co-workers, but granted with regard to

¹The "two coworkers" are understood to be an individual the Petitioner alleges willfully falsified (backdated) an entry on the form to indicate that the meter was returned on October 13, 1994, and an individual the Petitioner alleges willfully violated Procedure WC-8 on November 17, 1994, by signing out his own M&TE.

the requests for an "investigation into the above mentioned procedure violations" and for the NRC to "audit the Unit 1 maintenance department M&TE folders."

II. DISCUSSION

In the petition, the Petitioner raises concerns regarding numerous noncompliances with Procedure WC-8, Revision 0, at Millstone Unit 1. Specifically, the Petitioner states that (1) quality assurance (QA)² test meter 1587 was signed out on October 13, 1994, for weekly battery readings, and as of October 19, 1994, the user had not returned the meter or signed it in. The Petitioner states that this practice was in violation of Procedure WC-8, which stated "return M&TE to custodian upon completion of work";³ (2) although he identified a problem with Procedure WC-8 (specifically, who was responsible for the actual signing in and out of M&TE) to his first-line supervisor on November 7, 1994, as of December 1994, the procedure still had not been changed (in accordance with Procedure DC-4, "Procedural Compliance," which requires that if a procedure conflict or interpretation problem exists, a change or revision should be made); (3) on November 10, 1994, he noticed on a station form that someone signed in the QA meter with the return date of October 13, 1994, and that this was a willful falsification (backdating) of a nuclear record; (4) on November 17, 1994, an electrician co-worker was directed by their first-line supervisor to willfully violate Procedure WC-8 by signing out his own M&TE, and signed out his own M&TE although both the supervisor and co-worker knew they were to have the custodian sign out the equipment; (5) on November 21, 1994, his department manager instructed the custodian to give a spare key for the QA locker to the Millstone Unit 1 control room so the control room could sign out equipment at night; and (6) on November 25, 1994, a mechanic signed out M&TE without a custodian.

In addition, the Petitioner states that he believes that his department manager was directly responsible for sharing the effects of a new, revised, or rewritten procedure with the employees of his department if the procedure directly affected day-to-day operations. The Petitioner asserts that this individual's "lack of

²Quality Assurance comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system that provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

³This procedure had become effective on June 20, 1994. It required that a "designated custodian" enter the date of issue and date of return on the custody and usage record, and that the user of the equipment return it to the custodian upon completion of work. In Attachment 1 to the procedure, "custodian" was defined as the individual designated by the department head to store, track, and issue the department's M&TE.

communications” regarding the procedure has caused a “widespread problem of procedure noncompliance.”⁴

In letters to Northeast Nuclear Energy Company (NNECO), Licensee for Millstone Units 1, 2, and 3, dated December 5 and 28, 1994, and February 14, 1995, the NRC Staff raised a number of maintenance-related issues. In those letters, the NRC Staff requested NNECO to review these issues and submit a written response. Among these issues, the NRC requested NNECO to review two issues associated with Procedure WC-8 that are now presently being raised by the Petitioner. These were that: (1) the Millstone Unit 1 QA test meter 1587 was signed out on October 13, 1994, to perform weekly battery readings, but as of October 19, 1994, the user had not returned the meter or signed in the meter; and (2) many members of the Millstone Unit 1 Maintenance Department never received training on Procedure WC-8, Rev. 0, within 60 days of the effective date of June 20, 1994, as required by the documentation of training requirements form of NNECO Procedure DC-1.

In a letter dated March 6, 1995, NNECO responded to the issue regarding failure to return the QA meter signed out on October 13, 1994. In its letter, NNECO stated that on October 13, 1994, a maintenance electrician signed out QA test meter 1587 to perform weekly battery surveillances and signed it back in on the M&TE log on the same day. On October 19, 1994, a different maintenance electrician signed out and returned QA test meter 1587. Sometime later that day, QA test meter 1587 was signed out again and subsequently returned the same day. NNECO stated that it was unable to determine, based on interviews with the parties involved and a review of the custody and usage record, the exact circumstances surrounding QA test meter 1587. However, what was known was that QA test meter 1587 had been signed out once on October 13 and twice on October 19, 1994. NNECO’s review further concluded that strict compliance with Procedure WC-8 was not being observed at all three Millstone units in that a custodian was not being used to ensure that certain actions (i.e., signing in and out M&TE on the M&TE log) were being accomplished. However, NNECO stated that it believed it met the “intent of the procedure” in that the user of the M&TE stored, tracked, and issued the equipment as required by the procedure, except that the custodian was not involved. As a result of its review, NNECO undertook certain corrective actions. Specifically, NNECO held a site-wide meeting for all departments responsible for use or issuance of QA M&TE on February 21, 1995, to determine corrective actions necessary to ensure procedural compliance. Subsequently, NNECO revised Procedure WC-8

⁴ NNECO Procedure DC-1 requires that the Licensee select the training requirements to be used in training employees whenever procedures are revised, and indicate the type of training that would be performed on Attachment 5 to Procedure DC-1. For Procedure WC-8, Revision 0, the training required was marked as “training to be done by Department or Nuclear Training Department within 60 days of the effective date and prior to performance of procedure.”

on April 27, 1995, to specifically allow the user of M&TE to sign QA test equipment in and out. The custodian is still responsible for storing and tracking M&TE. In addition, Millstone Unit 1 control room personnel responsible for accessing QA M&TE were made aware of the logging requirements.

The NRC conducted a special safety inspection from May 15 through June 23, 1995, at the Millstone station. During this inspection, the Staff reviewed a number of the concerns, including the concerns about QA test meter 1587 and the other examples of noncompliance with Procedure WC-8 alleged by the Petitioner, and issued its findings in Inspection Report (IR) 50-245/95-22, 50-336/95-22, 50-423/95-22 (95-22), dated July 21, 1995.

During the inspection, the NRC Staff reviewed the custody and usage record sheets for QA test meter 1587 from September 27 to November 11, 1994. Based on this review, the Staff was unable to determine whether QA test meter 1587 was properly logged in and out in October 1994 or if the custody and usage record sheet was backdated. The NRC Staff discussed this issue with the workers involved who indicated that they had no recollection of the exact circumstances surrounding QA test meter 1587 and that, to the best of their knowledge, QA test meter 1587 was logged in and out properly. Therefore, the Staff was unable to determine whether QA test meter 1587 was controlled improperly and whether the Petitioner's co-worker willfully falsified (by backdating) a nuclear record (M&TE log).

The Staff also reviewed the original procedure and determined that although Procedure WC-8, Rev. 0, was not clear in specifying who was responsible for the actual signing in and out of equipment, NNECO was meeting the intent of the procedure in that M&TE was stored, tracked, and issued in a controlled manner. The NRC Staff further concluded that NNECO's additional corrective actions (i.e., modifying the procedure) were adequate in clarifying the procedure and should prevent interpretation problems in the future.

Notwithstanding the findings of the inspection report, however, the NRC has reconsidered this matter and determined that NNECO was not in compliance with Procedure WC-8, Rev. 0. This determination is supported by the fact that NNECO admitted in its March 6, 1995 letter that it was not in compliance with Procedure WC-8. In addition, the NRC has reviewed the custody and usage records for signing in and out M&TE on November 17 and 25, 1994, and determined that an electrician and mechanic had signed out their own M&TE, respectively, on those dates. Accordingly, the Petitioner's assertions that the procedure was violated when a co-worker electrician signed out his own M&TE on November 17, 1994, and a mechanic signed out M&TE on November 25, 1994, is substantiated. However, the NRC has been unable to confirm that either of these individuals had been "directed" by supervision to sign out the equipment.

In addition, NNECO's review, as described in its letter dated March 6, 1995, and verified by the Staff in IR 95-22, determined that keys had been available during this time frame in all Millstone control rooms and were in the possession of security personnel to allow access to QA M&TE storage locations. These groups required access to these areas in order to properly execute their duties. Therefore, since the custodian did not sign in and out the equipment, the Petitioner's additional assertion that the procedure was violated because security personnel and personnel in the Millstone Unit 1 control room could sign out M&TE at night is substantiated. However, the NRC has been unable to confirm that the department manager had instructed the custodian to give a spare key to the control room so the control room could sign out M&TE at night.

Furthermore, the Staff has determined that, since there were no safety consequences as a result of these events, the noncompliances with Procedure WC-8 did not constitute a violation that could reasonably be expected to have been prevented by the Licensee's corrective action for a previous violation or a previous Licensee finding that occurred within the past 2 years of the inspection at issue, adequate corrective actions were implemented regarding Procedure WC-8, and the violation was not willful, the violation would have been categorized in accordance with the enforcement policy in effect at the time of the inspection as a noncited Severity Level V violation and would not have been the subject of formal enforcement action.⁵

In addition, since the procedure was not clear in describing specific responsibilities and NNECO believed it was meeting the intent of the procedure, the NRC has concluded that the Petitioner's department manager, his first-line supervisor, and two co-workers did not deliberately violate NRC regulations or the Millstone Unit 1 operating license and, therefore, did not violate the provisions of 10 C.F.R. § 50.5. Moreover, NNECO revised Procedure WC-8 on April 27, 1995, and the procedure now more clearly allows the user of the M&TE to sign in and out QA test equipment. The custodian still is responsible for storing and tracking M&TE. Therefore, the Staff has determined that, although the Petitioner is correct in that the procedure was not revised as of December 1994, the procedure was subsequently revised, so that Procedure DC-4 was not violated.

⁵The staff has reconsidered this violation in accordance with the current enforcement policy (NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Action") and has concluded that the violation is below the level of significance of Severity Level IV violations. This determination is based on the fact that NNECO was meeting intent of the procedure; there was negligible impact on safety; NNECO's interpretation of the M&TE custodian's responsibilities does not indicate a programmatic problem that could have safety or regulatory impact; if the violation recurred, it would not be considered a significant concern; and the violation was not willful. Therefore, if considered under the new enforcement policy, this violation would be classified as a minor violation. Minor violations, as described in the current enforcement policy, are not the subject of formal enforcement action and are usually not cited in inspection reports. To the extent that such violations are described, they are now noted as noncited violations.

By letter dated April 26, 1995, NNECO provided its review of whether members of the Maintenance Department received training within 60 days of Revision 0 of Procedure WC-8 (June 20, 1994). In its letter, NNECO stated that no documentation indicating that training was conducted for Procedure WC-8, Rev. 0, had been found. While no training records were located, NNECO stated that the Millstone Unit 1 Maintenance Manager recalled that the procedure was discussed at a Maintenance Department meeting within 60 days of its effective date.

The NRC Staff reviewed Procedure DC-1 and determined that since NNECO could not locate the training records for Procedure WC-8, Rev. 0, and that training by the Maintenance Department or the Nuclear Training Department was not conducted within 60 days of the effective date for Procedure WC-8, Rev. 0, NNECO was in violation of Procedure DC-1.

The Staff's review of NNECO's April 26, 1995 response to the NRC letter dated February 14, 1995, was documented in IR 95-22. The Staff has reviewed NNECO's corrective actions that included NNECO management reemphasizing the importance of training on new or revised procedures and following procedures, the revising of Procedure WC-8, and training on the revised procedure. Based on that review, the Staff has determined that the corrective actions the Licensee has taken are acceptable. The Staff has further determined that since there were no safety consequences as a result of this event, it was not a violation that could reasonably be expected to have been prevented by the Licensee's corrective action for a previous violation or a previous Licensee finding that occurred within the past 2 years of the inspection at issue, adequate corrective actions were implemented, and the violation was not willful, the violation would have been categorized in accordance with the enforcement policy in effect at the time of the inspection as a noncited Severity Level V violation and would not have been the subject of formal enforcement action.⁶

III. CONCLUSION

The institution of a proceeding pursuant to section 2.206 is appropriate only if substantial health and safety issues have been raised. *See Consolidated Edison*

⁶The Staff has reconsidered this violation in accordance with the guidance in the current enforcement policy and has concluded that the violation is below the level of significance of Severity Level IV violations. This determination is based on the fact that there was negligible impact on safety; the violation does not indicate a programmatic problem that could have safety or regulatory impact; if the violation recurred, it would not be considered a significant concern; and the violation was not willful. Therefore this violation is classified as a minor violation and, as previously discussed, minor violations are not normally the subject of formal enforcement action and are usually not cited in inspection reports. To the extent that such violations are described, they are characterized as noncited violations.

Co. of New York (Indian Point, Units 1, 2, and 3), CLI-75-8, 2 NRC 173, 175 (1975), and *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), DD-84-7, 19 NRC 899, 924 (1984). This is the standard that has been applied to the concerns raised by the Petitioner to determine whether the action requested by the Petitioner, or other enforcement action, is warranted.

On the basis of the above assessment, I have concluded that, although certain minor procedural violations occurred, no substantial health and safety issues have been raised by the petition regarding Millstone Unit 1 that would require initiation of enforcement action. Therefore, to the extent that the Petitioner requests that escalated enforcement action be taken against individuals and NU for violations of Procedure WC-8 or failure to train employees on the procedure, the petition has been denied. However, as described above, the NRC conducted an inspection into the alleged violations of Procedure WC-8 from May 15 through June 23, 1995, and conducted an audit of the custody and usage record sheets. Therefore, to the extent that the Petitioner has requested an NRC "investigation into the above mentioned procedure violations" and for the NRC to "audit the Unit 1 maintenance department, M&TE folders," the petition has been granted.

As provided 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 a review of the Decision in that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

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

Dated at Rockville, Maryland,
this 11th day of February 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 70-3070-ML

**LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center)**

March 21, 1997

The Commission grants Nuclear Energy Institute's motion for leave to file an *amicus curiae* brief in the appeal of the Atomic Safety and Licensing Board's second Partial Initial Decision, LBP-96-25, 44 NRC 331 (1996), and adjusts the briefing schedule and page limits for responsive and reply briefs. The Commission also grants Louisiana Energy Services' motion for the Commission to defer filing of petitions for review of the third Partial Initial Decision, LBP-97-3, 45 NRC 99 (1997).

RULES OF PRACTICE: *AMICUS CURIAE*

"[A]n *amicus curiae* necessarily takes the proceeding as it finds it. An *amicus curiae* can neither inject new issues into a proceeding nor alter the content of the record developed by the parties." *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-862, 25 NRC 144, 150 (1987) (footnote omitted).

ORDER

The Commission has before it two contested motions in the proceeding on Louisiana Energy Services' (LES's) application for construction and operation

of the Claiborne Enrichment Center near Homer, Louisiana. The first is the Nuclear Energy Institute's (NEI's) motion for leave to file an *amicus curiae* brief in the appeal of the Atomic Safety and Licensing Board's second Partial Initial Decision, LBP-96-25, 44 NRC 331 (1996). The second is LES's motion for deferral of the schedule for seeking Commission review of the Board's third Partial Initial Decision, LBP-97-3, 45 NRC 99 (1997). We have decided to grant NEI's motion, and LES's motion in part, and to make appropriate adjustments in the briefing schedule and page limits.

1. Attached to NEI's motion is the *amicus* brief itself. NEI seeks leave to file its brief because it believes that LBP-96-25 rests on "significant legal error which, if allowed to stand, could severely affect the interests of the nuclear energy industry." The Intervenor, Citizens Against Nuclear Trash (CANT), opposes NEI's motion and requests that the Commission deny it. According to CANT, it would be "unduly burdensome" to require CANT, with its "extremely limited resources," to respond to yet another entity's arguments, when the license applicant is "adequately represented by two large law firms with significant resources." In the alternative, CANT requests that it be given sufficient time to respond to the NEI brief.

NEI's motion for leave to file the *amicus* brief is granted. CANT will suffer no substantive prejudice from the *amicus* filing: "[A]n *amicus curiae* necessarily takes the proceeding as it finds it. An *amicus curiae* can neither inject new issues into a proceeding nor alter the content of the record developed by the parties." *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-862, 25 NRC 144, 150 (1987) (footnote omitted). We adjust CANT's briefing deadline and page limits as indicated below so that CANT's brief can take account of the NEI filing.

2. LES's motion requests that the Commission defer the filing of petitions for review of the third Partial Initial Decision, LBP-97-3, until after a fourth Partial Initial Decision is issued sometime in the near future. LES states that "this approach will allow LES, and indeed all parties, to evaluate whether to file a petition for review based upon both partial decisions, and would allow the two partial decisions to be addressed simultaneously and therefore most efficiently."

CANT opposes the motion. According to CANT, LES's approach would be "unduly burdensome and unfair" because it might require CANT to simultaneously address both LBP-97-3 and the Board's forthcoming decision. However, a proposed filing schedule submitted by CANT indicates that CANT does not object to delaying the filing of a petition for review of LBP-97-3.

We have decided against mandating simultaneous petitions, because the two decisions likely will address quite separate issues: decommissioning funding (LBP-97-3) and "environmental justice" (the anticipated fourth Partial Initial Decision). However, to the extent that LES's motion requests a delay in filing a petition for review, we grant the motion. We anticipate that the parties can

better evaluate the need for and scope of further petitions after they have the opportunity to review the Board's fourth Partial Initial Decision, which we expect to be issued by May 1, 1997.

3. To accommodate the NEI *amicus* brief, we amend the briefing schedule and page limits with respect to LBP-96-25 as follows:

- (1) CANT shall file a single responsive brief on or before May 1, 1997. Its brief shall not exceed 55 pages.
- (2) The reply briefs shall be filed on or before May 15, 1997.

To accommodate the delay in filing petitions for review of LBP-97-3, we establish the following schedule:

- (1) Petitions for Review of LBP-97-3 shall be filed within 7 days after the date of issuance of the fourth Partial Initial Decision.
- (2) Responses to any petition for review of LBP-97-3 shall be filed in accordance with 10 C.F.R. § 2.786(b)(3).

Finally, the deadline for filing petitions for review of the fourth Partial Initial Decision is extended by 7 days beyond the deadline established by 10 C.F.R. § 2.786(b)(1).¹ In all other respects, all petitions and responses shall be filed in accordance with 10 C.F.R. § 2.786.

IT IS SO ORDERED.

For the Commission

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 21st day of March 1997.

¹ We have attempted to devise a schedule that avoids simultaneous filings. However, we recognize that depending on the date of issuance of the fourth Partial Initial Decision this schedule may need to be readjusted. The parties remain free to request an adjustment in this schedule if they believe that circumstances warrant it.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Thomas S. Moore, Chairman
Richard F. Cole
Frederick J. Shon

In the Matter of

Docket No. 70-3070-ML
(ASLBP No. 91-641-02-ML)
(Special Nuclear Material License)

LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center)

March 7, 1997

In this Partial Initial Decision in the combined construction permit–operating license proceeding for the Claiborne Enrichment Center, the Licensing Board resolves in favor of the Intervenor a portion of decommissioning funding contention B.1 and environmental contention J.3 concerning the conversion component of the estimated cost of tails disposal.

RULES OF PRACTICE: BURDEN OF PROOF

The Commission's rules of practice for the conduct of formal adjudicatory hearings provide in 10 C.F.R. § 2.732 that the applicant has the burden of proof in the proceeding. Thus, in order for the applicant to prevail on each contested factual issue, the applicant's position must be supported by a preponderance of the evidence. *Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 720 (1985); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-763, 19 NRC 571, 577 (1984). See 1 Charles H. Koch, Jr., *Administrative Law and Practice* § 6.44 (1985).

USEC PRIVATIZATION ACT: DEPLETED URANIUM TAILS

The USEC Privatization Act, 42 U.S.C. § 2297h-11(a)(1)(B) now makes the Department of Energy, at the request of an NRC-licensed enricher, responsible for the disposal of depleted uranium tails at DOE's disposal costs, including a pro rata share of any of DOE's capital costs.

PARTIAL INITIAL DECISION (Resolving Contentions B and J.3)

This Partial Initial Decision addresses contentions B and J.3 dealing with decommissioning funding filed by the Intervenor, Citizens Against Nuclear Trash ("CANT"), in this combined construction permit–operating license proceeding. The Applicant, Louisiana Energy Services, L.P. ("LES"), seeks a 30-year materials license to possess and use byproduct, source, and special nuclear material to enrich uranium using a gas centrifuge process at the Claiborne Enrichment Center ("CEC"). The Applicant intends to build the CEC on a site in Claiborne Parish, Louisiana, adjacent to and between the two unincorporated African-American communities of Center Springs and Forest Grove some 5 miles northeast of the town of Homer, Louisiana. The history of this licensing proceeding may be found in our earlier Partial Initial Decisions, LBP-96-7, 43 NRC 142 (1996), resolving contentions H, L, and M that challenged the Applicant's emergency plan and safeguards measures, and LBP-96-25, 44 NRC 331 (1996), resolving contentions J.4, K, and Q that challenged the need for the facility, the treatment of the no-action alternative in the final environmental impact statement ("FEIS"), and the Applicant's financial qualifications.

I. DECOMMISSIONING FUNDING CONTENTIONS

A. Contentions B and J.3

CANT's contention B, titled "Decommissioning Plan Deficiencies," asserts that "[t]he LES decommissioning [funding] plan does not provide reasonable assurance that the CEC site can be cleaned up and adequately restored upon cessation of operations." Although the Intervenor proffered a number of supporting bases for this contention, the Licensing Board, as then constituted, found three bases supported the contention. In basis B.1, CANT asserts that there is no realistic basis for LES' then estimate (of \$9.5 million per year) for the cost of depleted UF₆ tails ("DUF₆") disposal because the Applicant does not have a plan for the offsite disposal of tails. The Intervenor claims in basis B.4 that LES provides no details on how CEC decommissioning costs

were determined. Finally, in basis B.5, CANT declares that the Applicant's summary of decommissioning costs fails to indicate the facilities that will be decontaminated and the extent to which they will be decontaminated. LBP-91-41, 34 NRC 332, 337 (1991). On the strength of these three bases, the Licensing Board admitted contention B "insofar as it challenges the reasonableness of LES' decommissioning funding plan." *Id.*

In admitting contention B, the Board noted that the Commission's hearing notice for the licensing proceeding directed that the Applicant must have a "plausible strategy" for the disposition of DUF₆ tails. 56 Fed. Reg. 23,310, 23,313 (1991). Additionally, the Board stated that the Commission's regulations, 10 C.F.R. § 70.25(a), (e), require that the Applicant submit a decommissioning funding plan containing a cost estimate for decommissioning and the means for adjusting cost estimates and funding levels periodically over the life of the facility. *See also* 10 C.F.R. § 40.36(a), (c)(1), (d), (e)(3). In light of these factors, the Board ruled that, although there was no regulatory requirement that the Applicant have a "concrete plan" for the disposal of depleted uranium tails, LES must have a plausible strategy for tails disposition and, in order for the regulations to have any meaning, the Applicant's "cost estimate should contain reasonable estimates for an adequately described decommissioning strategy." 34 NRC at 338. Thus, the Board ruled that CANT's contention B supported by bases B.1, B.4, and B.5 had satisfied the Commission's contention pleading requirements by alleging that "the decommissioning funding plan does not contain reasonable estimates for decommissioning nor does it adequately describe the underlying decommissioning strategy." *Id.*

CANT's contention J, titled "Inadequate Assessment of Costs Under NEPA," alleges that the Applicant's environmental report ("ER") for the CEC does not adequately describe or weigh the environmental, social, and economic impacts and costs of operating the facility and that the costs of the project far outweigh the benefits of the proposed action. In basis J.3, the Intervenor asserts that LES has not provided a sufficient foundation for its decommissioning cost estimates and incorporates the bases it proffered in support of contention B. The Licensing Board found that bases B.4 and B.5 also supported contention J and admitted the contention. *Id.* at 350. Although CANT contention J.3 is phrased only in terms of a challenge to the Applicant's ER, the contention necessarily encompasses the Staff's later-filed environmental impact statement as well. *See* 44 NRC at 337-38. Further, because the Intervenor's contention J.3 challenges the same decommissioning costs (albeit in the context of the Applicant's ER and the Staff's EIS) that are the subject of contention B, all parties addressed the contentions together in their testimony. Similarly, we do not separately address CANT's contention J.3 and our findings and conclusions on contention B also encompass contention J.3.

B. Witnesses and Exhibits

In support of its position on contentions B and J.3, the Applicant presented the testimony of a panel of witnesses consisting of Peter G. LeRoy, Bernard G. Dekker, Richard W. Dubiel, and John M. A. Donelson. Due to a pretrial procedural ruling the prefiled direct testimony of this panel of witnesses appears in the record in two parts, i.e., that of Mr. LeRoy and Mr. Dekker (LeRoy-Dekker fol. Tr. 1016) and that of Mr. Dubiel, Mr. Donelson, and Mr. LeRoy (Dubiel-Donelson fol. Tr. 1026).

Mr. LeRoy, the Licensing Manager of the CEC, was responsible for compiling the information on decommissioning planning and funding in the LES Decommissioning Funding Plan, the LES Safety Analysis Report, and the Applicant's ER. (LeRoy-Dekker at 2 fol. Tr. 1016.) Mr. Dekker is the Manager of Safety, Safeguards, and Licensing for Urenco Nederland B.V., which operates uranium enrichment facilities at Almelo in the Netherlands. He has held that position since 1984 and, in his over 18 years working for Urenco Nederland, B.V., he has gained extensive experience in the operation, decontamination, and decommissioning of gas centrifuge uranium enrichment facilities. Mr. Dekker was retained by the Applicant to advise LES on various matters with respect to planning and funding for decontamination and decommissioning of the CEC, including the development of the LES Decommissioning Funding Plan. (*Id.*)

Mr. Dubiel holds a bachelor of science degree in physics and a master of science degree in nuclear engineering and he currently is the Director of Special Programs at Applied Radiological Control, Inc. In that capacity he is responsible for overseeing specialty health physics and radiological decontamination services provided to the United States Departments of Energy and Defense and various NRC licensees. He has over 20 years of experience handling NRC-licensed materials, including classifying, packaging, and shipping radioactive waste for disposal. (Dubiel-Donelson at 2 & Attach. 2 fol. Tr. 1026.) Like Mr. Dubiel, Mr. Donelson also has earned a bachelor of science degree in physics and a master of science degree in nuclear engineering. He is an engineer in the Fuel Management Section of the Nuclear Engineering Division of Duke Power Company and his specific area of responsibility is uranium enrichment. Mr. Donelson is knowledgeable about the characteristics and properties of uranium in various physical and chemical forms. (*Id.* at 3.)

The prefiled direct testimony of these witnesses on contentions B and J.3 was admitted into evidence pursuant to a pretrial stipulation of the parties and without further objection at the hearing. (Tr. 1016, 1026.) Because the Applicant did not offer these witnesses as experts and, in light of the parties' admissibility stipulation, the Board did not rule at the hearing on the qualifications of these witnesses as experts. Obviously, however, as the LES official responsible for

compiling the information on decommissioning in the LES license application, Mr. LeRoy is qualified to testify on that information and related submittals. Further, we find that Mr. Dekker is qualified by knowledge and experience and that Mr. Dubiel and Mr. Donelson are qualified by education, knowledge, and experience to testify as expert witnesses on the issues involved in contentions B and J.3.¹

In support of its contentions B and J.3, the Intervenor presented the testimony of Dr. Arjun Makhijani, President of the Institute for Energy and Environmental Research. (Makhijani at 1 fol. Tr. 1081.) Dr. Makhijani earned his Ph.D. in engineering from the University of California, Berkeley, where his dissertation subject involved controlled nuclear fusion. He currently serves as a consultant to the United States Environmental Protection Agency ("EPA") Science Advisory Board, Radiation Advisory Committee, and he is a member of the Subcommittee on Radiation Cleanup Standards of the EPA National Advisory Council for Environmental Policy and Technology. He has also been a consultant to numerous other institutions such as the Congressional Office of Technology Assessment, Lawrence Berkeley Laboratory, Tennessee Valley Authority, Ford Foundation, and Edison Electric Institute. Dr. Makhijani has extensive experience in the area of nuclear waste classification and disposal and he has published numerous books and reports on these topics, including co-authoring *High-Level Dollars Low-Level Sense: A Critique of Present Policy for the Management of Long-Lived Radioactive Waste and Discussion of an Alternative Approach*, Apex Press, New York (1992). (*Id.* at 1 & Attach.) The prefiled direct testimony of Dr. Makhijani was admitted pursuant to a pretrial stipulation of the parties and the Intervenor offered his testimony as that of an expert in the field of nuclear engineering. (Tr. 1081.) We find that Dr. Makhijani is qualified

¹ Pursuant to a stipulation of the parties, the following Applicant exhibits were admitted into evidence relating to contentions B and J.3: Applicant's Exhibit 3, SECY-91-019, "Disposition of Depleted Uranium Tails from Enrichment Plants," Jan. 25, 1991 (App. Exh. 3); Applicant's Exhibit 4, correspondence (with attachments) between NRC and LES re decommissioning designated 4(a)-(q) (App. Exh. 4(a)-(q)); Applicant's Exhibit 5, Letter from Frank A. Shallo, Vice President, Market Development, COGEMA, Inc., to W. Howard Arnold, President, LES (Oct. 16, 1991) (App. Exh. 5); Applicant's Exhibit 6, Letter from Frank A. Shallo, Vice President, Market Development, COGEMA, Inc., to W. Howard Arnold, President, LES, Feb. 22, 1995 (App. Exh. 6); Applicant's Exhibit 7, Uranium Enrichment Organization (Oak Ridge, Tenn.), Martin Marietta Energy Systems, Inc., "The Ultimate Disposition of Depleted Uranium," Dec. 1990 (report prepared for U.S. Dep't of Energy [hereinafter Martin Marietta Report] (App. Exh. 7); Applicant's Exhibit 8, Waste Management Technology Division, Science Applications International Corp., "Depleted Uranium Disposal Options Evaluation," May 1994 (report prepared for EG&G Idaho, Inc., and U.S. Dep't of Energy) [hereinafter EG&G Report] (App. Exh. 8); Applicant's Exhibit 9, Bureau of Mines, U.S. Dep't of the Interior, *Minerals Yearbook*, 1992, at 183-89, 194, 202, 208 (App. Exh. 9). Previously, Applicant's Exhibits 1, the CEC License Application; 1(a) the CEC Safety Analysis Report; 1(e), the CEC Proposed License Conditions; and 1(h), the CEC Environmental Report, which are also relevant to these contentions, were previously admitted into evidence. (Tr. 31.)

by education, knowledge, and experience to testify as an expert on the issues involved in contentions B and J.3.²

The NRC Staff presented the testimony of a panel of witnesses consisting of Yawar H. Faraz, John W. N. Hickey, and Dr. Joseph D. Price, although only Mr. Faraz and Mr. Hickey presented the Staff's prefiled direct testimony. (Faraz-Hickey fol. Tr. 1106.) Mr. Faraz holds a bachelor of science degree in nuclear and mechanical engineering and he is a nuclear process engineer in the Certification Section, Enrichment Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards ("NMSS"). Since April 1994, he has served as the NRC Licensing Project Manager for the CEC. (*Id.* at 1.) Mr. Hickey earned a bachelor of science degree in mechanical engineering and a master of science degree in environmental health. He is the Chief of the Enrichment Branch, Division of Fuel Cycle Safety and Safeguards, NMSS, which has responsibility for all regulatory matters related to uranium enrichment. (*Id.* and Attach. 2.) Dr. Price earned his Ph.D. in chemical engineering and currently he is a senior chemical engineer with Science Applications International Corporation ("SAIC"). As task manager, he directed SAIC's effort to develop under contract to the NRC the Safety Evaluation Report for the CEC and, in over 16 years with SAIC, Dr. Price has had extensive experience in safety, transport, and environmental analyses of nuclear waste facilities as well as chemical process modeling and analysis. (Staff Exh. 4.)³ Pursuant to the pretrial stipulation of the parties and without further objection at the hearing, the prefiled direct testimony of Mr. Faraz and Mr. Hickey on these contentions was admitted. (Tr. 1104.) We find that Mr. Faraz, Mr. Hickey, and Dr. Price are qualified by education, knowledge, and experience to testify as experts on the issues involved in contentions B and J.3.

As in the case of the other contentions adjudicated in this proceeding, the Commission's rules of practice for the conduct of formal hearings provide in 10 C.F.R. § 2.732 that the Applicant has the burden of proof in the proceeding. Therefore, in order for LES to prevail on each contested factual issue, the Applicant's position must be supported by a preponderance of the evidence. *Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 720 (1985); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-763, 19 NRC 571, 577 (1984). See 1 Charles H. Koch, Jr., *Administrative Law and Practice* § 6.44 (1985). In

² Without objection, Intervenor's Exhibit I-AM-70, Sandia National Laboratories, "Performance Assessment of the Proposed Disposal of Depleted Uranium as Class A Low-Level Waste," Dec. 1992 (I-AM-70), was offered into evidence by CANT on these contentions and admitted. (Tr. 1081.)

³ Without objection, Staff's Exhibit 4 (Staff Exh. 4), a statement of Dr. Price's professional qualifications, was offered into evidence by the Staff and admitted. (Tr. 1106.) Previously, the Staff's Safety Evaluation Report ("SER"), Staff Exh. 1, and the Staff's FEIS, Staff Exh. 2, which are also relevant to these contentions, were admitted into evidence in the proceeding. (Tr. 154, 501.)

accordance with the Commission's burden of proof rule and pursuant to the stipulation of the parties, the Applicant presented its case on these contentions first, followed by the Intervenor, and then the NRC Staff.

II. BOARD FINDINGS ON PARTIES' POSITIONS

Before turning to contention B, a further brief explanation of the applicable standard for judging the Intervenor's challenge to the Applicant's funding plan is helpful. As previously mentioned, the Licensing Board admitted CANT's contention B to the extent that it challenged the reasonableness of the LES Decommissioning Funding Plan. In so ruling, the Board noted that the Commission's hearing notice required the Applicant to have a plausible strategy for the disposal of DUF_6 tails as part of its funding plan and that the Commission's regulations required the funding plan to contain reasonable cost estimates for the components of the plan. Although in its hearing notice the Commission listed a number of possible generic tails disposal strategies such as storage of tails at the plant site as a possible future resource or conversion of tails to uranium oxide for disposal, the Commission did not specifically define what constitutes a plausible strategy. The plain meaning of these terms, however, provides the answer. The dictionary defines "plausible" as "reasonable" or "credible," *Webster's Third New International Dictionary* 1736 (1971), and "strategy" as a "plan." *Id.* at 2256. Thus, in assessing the plausible tails disposal strategy adopted by the Applicant as part of its decommissioning funding plan, we first must determine whether the funding plan contains a reasonable or credible plan to dispose of the DUF_6 tails generated at the CEC and then determine whether the Applicant's cost estimates for the components of the plan are reasonable.

A. LES Tails Disposal Strategy

The Applicant's tails disposal strategy is capsulized in the LES Decommissioning Funding Plan that appears as Exhibit 1 to the LES License Application. In pertinent part, the Applicant's funding plan states:

The annual tails disposal cost is estimated to be \$16.175 million. This is multiplied by 30 years to arrive at the \$485.3 million figure. Costs are based on converting UF_6 to U_3O_8 with subsequent disposal in a facility under cognizance of the NRC. U_3O_8 conversion costs are based on estimates by a vendor which could make this service available to LES. Disposal costs are based on NRC recommendations and a study by Martin Marietta. The conversion and disposal costs are added and escalated to 1996 dollars.

(App. Exh. 1, at Exh. 1, at I-4.) Further, the LES funding plan states that the Applicant intends to set aside the annual tails disposal cost component of its overall decommissioning costs in an external trust that meets the requirements of the Commission's funding regulations. (*Id.* at I-2, I-5, I-8 to -9.) Finally, the LES plan states that the Applicant will update its decommissioning cost estimate at least once every 5 years. (*Id.* at I-6.)

At the hearing, the Applicant's witnesses, Mr. LeRoy and Mr. Dekker, provided additional details of the LES tails disposal plan. Their testimony recognizes that there currently are no facilities in the United States to convert DUF_6 to U_3O_8 , but they stated that COGEMA, Inc., the American affiliate of a French nuclear fuel company, "has indicated to LES in writing its willingness to consider providing, in the United States, conversion services for DUF_6 ," (LeRoy-Dekker at 24 fol. Tr. 1016; App. Exhs. 5 & 6.) These LES witnesses asserted that, in its letters to LES, COGEMA indicated that the experience gained by its parent company in successfully operating a commercial-size defluorination facility in France could be used as the basis for employing technology in the United States to convert DUF_6 to U_3O_8 . As the COGEMA letter states, the "prudent management of depleted UF_6 should consider conversion to U_3O_8 powder, which is insoluble in water, does not react with external chemical agents, is free of fluorine and is the most compact form for storage." (App. Exh. 5.) Additionally, Mr. LeRoy and Mr. Dekker testified that, in 1991, COGEMA estimated its charge for deconversion services to be in the range of \$3-5 per kilogram of uranium and its 1995 updated estimate indicated a range of \$4-6. These witnesses stated that these estimates assume the construction and operation of a deconversion facility in the United States under NRC standards. (LeRoy-Dekker at 24 fol. Tr. 1016; App. Exhs. 5 & 6)

After conversion of the DUF_6 tails to U_3O_8 , the LES disposal strategy provides for the U_3O_8 , as waste, to be shipped to a final disposal site for deep land burial such as in a deep mine. Again, the LES disposal plan recognizes that currently there are no operating deep disposal sites, but Mr. LeRoy testified that it is reasonable to assume such a site will be available in the future because in the United States there are dozens of underground uranium mines and other underground mines. (LeRoy-Dekker at 34 fol. Tr. 1016.)

Although the Applicant's tails disposal strategy calls for LES to convert the CEC tails to U_3O_8 and then ship the U_3O_8 for deep burial as waste, Mr. LeRoy candidly admitted in his testimony that, "[a]s a practical matter, LES is holding open its options for disposition of UF_6 ." (*Id.* at 19.) He testified that "for purposes of this licensing proceeding, in order to satisfy the Commission's requirement that the CEC license application contain a 'plausible strategy' for disposition of depleted uranium, LES has assessed, and factored into its funding plan the costs of conversion of DUF_6 to DU_3O_8 and land disposal (deep burial) of DU_3O_8 as if it were a waste." (*Id.*) The Applicant's witnesses stated, however,

that LES did not necessarily plan on disposing of the depleted uranium from the CEC by burying it as waste and that there were other potential options for the future disposition of DUF_6 . They noted that the Department of Energy ("DOE") is currently analyzing the tails disposition issue and that European enrichers consider depleted uranium tails a resource rather than a waste product. Further, they testified that Urenco's long-term plan for the disposition of depleted uranium is being studied and that, at present, the plan calls only for the offsite conversion of tails to U_3O_8 . (*Id.* at 20.) Finally, Mr. LeRoy readily conceded that, as a practical matter, LES will follow the same tails disposition option that DOE selects for its stockpile of tails. (Tr. 1076-77, 1069-70.)

The NRC Staff witnesses, Mr. Faraz and Mr. Hickey, stated in their direct testimony that they found the Applicant's tails disposition plan calling for conversion of DUF_6 to U_3O_8 , with subsequent deep subsurface burial, an acceptable plausible strategy. In this regard, the Staff's review of the LES decommissioning plan in the SER states:

Currently there are no facilities designed and equipped for the disposition of large volumes of depleted uranium originating from enrichment facilities. The Department of Energy (DOE) currently possesses essentially the entire depleted UF_6 inventory in the United States. In July 1993, the United States Enrichment Corporation (USEC) took over from DOE low enriched uranium production activities conducted at the two operating gaseous diffusion plants (GDP) located in Portsmouth, Ohio and Paducah, Kentucky. Currently neither DOE nor USEC has in place a plan concerning final disposition of the DUF_6 . The Energy Policy Act of 1992 requires DOE to address this issue. The NRC staff believes that it is premature to require a prescriptive resolution prior to DOE's determination on disposition of DUF_6 , which will, to a large extent, determine the disposition options for LES' DUF_6 . For the purpose of estimating funding requirements related to the disposition of DUF_6 , the NRC staff finds acceptable the applicant's estimates based on conversion of DUF_6 to U_3O_8 , which is much more environmentally stable than UF_6 or uranium tetrafluoride (UF_4), and disposition in a deeper than shallow land burial facility (for example, an abandoned mine cavity).

(Staff Exh. 1, at 15-12.)

Additionally, in the FEIS, the Staff modeled the respective doses for both near-surface and deep burial disposal because there currently are no disposal facilities for large quantities of depleted uranium tails. Because the projected drinking water and agricultural doses from a modeled near-surface burial site consisting of an earth-mounded bunker subject to the environmental characteristics of the humid southeastern United States exceeded the 10 C.F.R. Part 61 limits, the Staff concluded that a deep disposal site is most likely to be selected for tails disposal. (Staff Exh. 2, at 4-66 to -67 & Appendix A, at A-9.) The Staff also modeled a hypothetical deep disposal site. It assumed the site would be an existing cavity, such as an abandoned mine, located in the United States and would have geologic characteristics similar to those of two representative sites that previously have been characterized for disposal of radioactive waste,

i.e., a granite formation overlain by a thin layer of glacial till or a sequence of interbedded sandstone and basalt layers. (Staff Exh. 2, at 4-66 to -67 & Appendix A, at A-10.) The Staff's FEIS analysis concluded that all estimated dose impacts for a deep disposal site are less than those set forth in 10 C.F.R. Part 61. (Staff Exh. 2, at 4-67 to -68 & Appendix A, at A-10 to -15.)

The purpose of the Applicant's tails disposal strategy is to enable the computation of reasonable cost estimates for the various essential elements of the decommissioning plan, thereby ensuring compliance with the Commission's regulatory requirement that during the CEC's life LES escrows sufficient funds to cover, *inter alia*, the cost of tails disposal. With this in mind, we find that the Applicant has presented a plausible disposal strategy. The Applicant's plan to convert DUF_6 to U_3O_8 at an offsite facility in the United States and then ship that material as waste to a final site for deeper than surface burial is a reasonable and credible plan for tails disposal. Although no conversion facilities currently exist in the United States, the LES materials license will give the Applicant 15 years before it first must move the accumulated DUF_6 offsite. (App. Exh. 1(e), at 1-2.) The conversion of DUF_6 to U_3O_8 , as the COGEMA experience in France demonstrates, is a commercially feasible process using known chemical processes that could be readily employed in the United States by COGEMA or another entity without first having to overcome difficult technical hurdles. (App. Exh. 7, at 18; Staff Exh. 2, at Appendix A, at A-1 to -4.) Thus, contrary to the Intervenor's assertion,⁴ the fact that there is no currently operating defluorination facility in the United States or a firm commitment by COGEMA or some other entity to build such a facility does not somehow make it unlikely, or unreasonable to assume, that one will be built here in the future to convert DUF_6 tails to U_3O_8 . Similarly, in light of the numerous existing uranium and other mines in the United States, it is reasonable to assume an appropriate site for deep burial of U_3O_8 will be available in the future. Indeed, the reasonableness and credibility of the LES disposal strategy is enhanced by the Department of Energy's clear need to address the disposal options for its huge inventory of DUF_6 that, as of mid-1992, amounted to some 534,000 metric tonnes (App. Exh. 8, at 3) — an amount of depleted uranium tails five times the amount of tails the CEC will produce under its 30-year license.

Further, CANT's legal challenge to that element of the Applicant's disposal strategy calling for deep burial of U_3O_8 is without merit. It argues that pursuant to the Commission's regulations, 10 C.F.R. § 61.55(a)(2)(iv), deeper than surface burial is unavailable for DUF_6 disposal. According to the Intervenor, DUF_6 waste, which CANT claims is closely comparable in radiological properties to transuranic waste, must be disposed of in a geologic repository (with a

⁴ Citizens Against Nuclear Trash's Proposed Reply Findings of Fact and Conclusions of Law Regarding Contentions B and J.3 (June 26, 1995) at 21 [hereinafter CANT RF].

consequent order of magnitude increase in cost) unless the Commission first approves and licenses a specific disposal site. The Intervenor claims, therefore, that LES does not have the option of establishing, based on a generic analysis like that in the FEIS, that the tails can be disposed of in some intermediate waste disposal facility.⁵ The Intervenor's assertions, however, merely repeat the same arguments CANT made to us in its pretrial "Petition for Waiver of 10 C.F.R. § 61.55(a)(3) and 10 C.F.R. § 61.55(a)(6) and for Classification of Depleted Uranium Tails as Greater Than Class C Radioactive Waste" (Jan. 17, 1995). In a pretrial Memorandum and Order (Mar. 2, 1995) we rejected these same arguments and denied the Intervenor's waiver petition. Our earlier ruling is the law of the case on these issues and forecloses any reexamination here. Thus, in accordance with our earlier ruling, we find that the Applicant's tails disposal strategy is not deficient for failure to treat the CEC tails as greater than class C waste with mandatory disposal in a geologic repository licensed under 10 C.F.R. Part 60.

Although we find that the Applicant's tails disposal plan is a plausible strategy for purposes of estimating LES' tails disposal costs, we note that a recent change in the law by the enactment of the USEC Privatization Act, Pub. L. No. 104-134, 100 Stat. 1321 (1996), will most likely dictate the actual LES disposal strategy.⁶ That Act now makes DOE, at the request of an NRC-licensed enricher, responsible for the disposal of depleted uranium tails at DOE's disposal costs,

⁵ Citizens Against Nuclear Trash's Proposed Findings of Fact and Conclusions of Law Regarding Contentions B and J.3 (May 26, 1995) at 28-30, 39-47 [hereinafter CANT PF].

⁶ In its entirety, 42 U.S.C. § 2297h-11 provides as follows:

(a) Responsibility of DOE

(1) The Secretary, at the request of the generator, shall accept for disposal low-level radioactive waste, including depleted uranium if it were ultimately determined to be low-level radioactive waste, generated by—

(A) the Corporation as a result of the operations of the gaseous diffusion plants or as a result of the treatment of such wastes at a location other than the gaseous diffusion plants, or

(B) any person licensed by the Nuclear Regulatory Commission to operate a uranium enrichment facility under sections 2073, 2093, and 2243 of this title.

(2) Except as provided in paragraph (3), the generator shall reimburse the Secretary for the disposal of low-level radioactive waste pursuant to paragraph (1) in an amount equal to the Secretary's costs, including a pro rata share of any capital costs, but in no event more than an amount equal to that which would be charged by commercial, State, regional, or interstate compact entities for disposal of such waste.

(3) In the event depleted uranium were ultimately determined to be low-level radioactive waste, the generator shall reimburse the Secretary for the disposal of depleted uranium pursuant to paragraph (1) in an amount equal to the Secretary's costs, including a pro rata share of any capital costs.

(b) Agreements with other persons

The generator may also enter into agreements for the disposal of low-level radioactive waste subject to subsection (a) of this section with any person other than the Secretary that is authorized by applicable laws and regulations to dispose of such wastes.

(c) State or interstate compacts

Notwithstanding any other provision of law, no State or interstate compact shall be liable for the treatment, storage, or disposal of any low-level radioactive waste (including mixed waste) attributable to the operation, decontamination, and decommissioning of any uranium enrichment facility.

including a pro rata share of any of DOE's capital costs. 42 U.S.C. § 2297h-11(a)(1)(B),(a)(3). As previously indicated, the Applicant's Licensing Manager, Mr. LeRoy, testified that, as a practical matter, LES will follow the same disposal option selected by DOE for the government's DUF₆ stockpile. Similarly, the Staff's witness, Mr. Hickey, testified that the NRC anticipates that LES will use the same tails disposal method that DOE selects. (Tr. 1156-57.) The Intervenor also apparently agrees, for in its proposed findings CANT states that "LES intends to rely on DOE's disposition strategy." CANT PF at 50. Thus, even though the USEC Privatization Act, 42 U.S.C. § 2297h-11(b), provides LES with the option of using other authorized persons for tails disposal, we think it is clear, and all parties apparently agree, that the Applicant's actual disposal method will be to transfer the CEC tails to DOE and pay DOE's disposal charges.⁷

B. Cost Estimates for Tails Disposal

While we recognize that DOE's future charges for tails disposal will ultimately determine the Applicant's tails disposal costs, the Commission's regulations require that the Applicant provide reasonable cost estimates for its tails disposal plan at this time in order to ensure that LES sets aside sufficient funds during the life of the CEC to cover its disposal costs. Accordingly, we must determine whether the Applicant's cost estimates for the components of its chosen plan are reasonable on the basis of the record before us. We turn now to those cost estimates, noting that, because DOE's disposal scheme is likely to be the same as the Applicant's plausible strategy, the current hearing record still is relevant to the issue of whether the Applicant's ultimate tails disposal cost estimate is reasonable.

As earlier indicated, the Applicant's Decommissioning Funding Plan provides that the annual tails disposal cost for the CEC is \$16.175 million, totaling \$485.3 million for 30 years of operation. (App. Exh. 1, at Exh. 1, at I-4.) At the hearing, Mr. LeRoy's direct testimony stated that the annual tails disposal figure includes \$0.8 million for shipment costs, \$12.0 million for conversion costs of

⁷ As a practical matter, the enactment of 42 U.S.C. § 2297h-11(a) and (c) making DOE responsible for depleted uranium tails upon the request of the enricher and insulating any state or interstate compact from liability for such wastes, also moots the Intervenor's legal argument that the LES tails disposal strategy is implausible because it fails to provide that the tails from the CEC must be disposed of in Louisiana, or within the states of the Central Interstate Compact of which Louisiana is a member, under the provisions of the Low Level Radioactive Waste Policy Act ("LLRWPA"), 42 U.S.C. § 2021b *et seq.*, and the practical workings of that law. CANT PF at 7-10, 30-34; CANT RF at 15-17. The Applicant already has indicated that its actual disposal method will be to transfer the CEC tails to DOE — a view shared by the Staff and the Intervenor. Therefore, in light of the new federal option available to the Applicant, it is a virtual certainty, for many of the reasons urged by the Intervenor in its argument, that no State or interstate compact will undertake the time-consuming, expensive, and politically difficult task of licensing a facility for depleted uranium tails, thereby further ensuring that the Applicant will request DOE to dispose of the CEC tails. Thus, the Intervenor's elaborate argument under the LLRWPA has been overtaken by the passage of the USEC Privatization Act.

DUF₆ to U₃O₈, and \$3.375 million for disposal of U₃O₈. (LeRoy-Dekker at 23 fol. Tr. 1016.) In the SER, the Staff found the Applicant's estimated facility decommissioning funding, which includes the Applicant's annual tails disposal cost of \$16.175 million, adequate. (Staff Exh. 1, at 15-21, 15-23.) At the hearing, Mr. Faraz and Mr. Hickey stated in their direct testimony that the LES tails disposal estimates were reasonable and, more specifically, that the Applicant's estimated cost for U₃O₈ burial was reasonable. (Faraz-Hickey at 7, 9 fol. Tr. 1106.) The Intervenor challenges each of the Applicant's component cost estimates.

1. Transportation Costs

In his prefiled direct testimony, Mr. LeRoy stated, without elaboration, that the LES estimate of \$800,000 per year transportation costs for depleted uranium tails "is based on conversations with shippers of UF₆ and U₃O₈." (LeRoy-Dekker at 25 fol. Tr. 1016.) The Intervenor's witness, Dr. Makhijani, challenged the validity of the LES estimate, asserting that it implicitly assumes that the conversion facility will be located very close to the disposal site. He opined that, because the location of the disposal site is unknown, such an assumption is rash and that it was unlikely any community would accept both a conversion facility and a disposal site. Dr. Makhijani testified that the Applicant's transportation costs should have provided for the cost of the shipment of U₃O₈ from the conversion facility to the disposal site as well as for packaging the U₃O₈ for shipment. (Tr. 1200.)

The Applicant's testimony setting out the basis for its annual tails disposal cost estimate is sparse, at best. Nevertheless, contrary to Dr. Makhijani's assertion, the reasonable inference from Mr. LeRoy's bare-bones testimony that the LES estimate is based upon information from shippers of UF₆ and U₃O₈ is that the Applicant's estimated shipping costs are based upon the shipment of DUF₆ tails to the converter as well as the shipment of U₃O₈ from the converter to a disposal facility. And, in the end, any weakness in the Applicant's testimony about its transportation costs is rectified by the transportation cost data contained in the 1990 Martin Marietta Report, "The Ultimate Disposition of Depleted Uranium," prepared at Oak Ridge for DOE (App. Exh. 7, at 17-18) and the 1994 EG&G Report, "Depleted Uranium Disposal Options Evaluation," prepared at Idaho Falls also for DOE. (App. Exh. 8, at 48-50.)

The Martin Marietta Report estimated that the rail transportation cost of shipping DUF₆ from Paducah, Kentucky, the location of one of the gaseous diffusion plants then owned by DOE, to an unspecified West Coast location for conversion and disposal was approximately \$0.15/kgU. The EG&G Report estimated that the truck transportation cost of shipping U₃O₈ from Piketon, Ohio, the location of another DOE facility, to the Nevada Test Site ("NTS") in Nevada

was approximately \$0.18/kgU in 1993 dollars. In addition, the EG&G Report estimated that 55-gallon drum container costs added another \$0.11/kgU to the estimate. Obviously, precise transportation cost estimates cannot be obtained at this time because such costs are dependent on the location of the conversion facility and the ultimate disposal site. But the application of this same rail rate from Paducah to the same West Coast location for the CEC UF₆ tails yields transportation costs of less than half the amount to be set aside by LES for annual transportation costs. Even escalating that cost to 1996 dollars yields an amount that is a little over half the LES estimate. Similarly, the application of this same truck and container rate from Piketon to the NTS for the CEC U₃O₈ yields total transportation costs that are about 90% of the amount to be set aside by LES for annual transportation costs. Even escalating that cost to 1996 dollars yields an amount that is approximately the same as the LES estimate. Although Paducah, Kentucky, and Piketon, Ohio, obviously are not Homer, Louisiana, this comparison serves to illustrate the dimensions of the rail transportation costs of UF₆ and the truck transportation costs of U₃O₈ from east of the Mississippi River to the West Coast and the NTS, respectively. Accordingly, we find that the Intervenor's challenge to the Applicant's annual tails disposal transportation cost estimate is without merit and that the LES estimate of the transportation component of its tails disposal estimate is a reasonable one.

2. Disposal of U₃O₈

The Applicant's annual tails disposal estimate also includes \$3.375 million for the deep disposal by burial of U₃O₈. Mr. LeRoy testified that the LES estimate is based upon a June 18, 1993 letter from the NRC to LES. (LeRoy-Dekker at 25 fol. Tr. 1016.) In part, the NRC letter states that "[u]ntil the specific disposal site and method are identified, the estimated cost is uncertain. However, for financial planning purposes, we believe that it is reasonable to assume a disposal cost of approximately \$1.00 per kilogram of U₃O₈." (App. Exh. 4h, at 1-2.) In turn, the Staff's basis for the \$1.00/kg U₃O₈ relies upon the 1990 Martin Marietta Report and the Staff's tracking of low-level waste burial charges. (LeRoy-Dekker at 26 fol. Tr. 1016; Faraz-Hickey at 9 fol. Tr. 1106.) The Martin Marietta Report estimates the permanent disposal costs of U₃O₈ utilizing the waste disposal fees for shallow burial at the federal NTS and Hanford, Washington disposal sites. It states that, with efficient packaging, low-density U₃O₈ would cost about \$0.25/kgU for NTS disposal and \$1.00/kgU at Hanford. The Report concludes that the higher-cost disposal estimate of \$1.00/kgU represents the prudent basis for current estimates. (App. Exh. 7, at 17.)

Mr. LeRoy explained that the LES estimate stated in kilograms of U₃O₈ is about 15% higher than the estimates from the Martin Marietta Report stated in

kilograms of uranium because U_3O_8 is about 85% uranium by weight. (LeRoy-Dekker at 27 fol. Tr. 1016.) Additionally, he testified that a 1994 EG&G Report indicates that the LES burial estimate of \$1.00/kg U_3O_8 remains valid. (*Id.* at 26.) The EG&G Report estimates the cost of nonretrievable burial of DU_3O_8 by DOE at the NTS to be \$0.15/kgU in 1994 dollars and about 19% more, or \$0.18, for a non-DOE generator. Further, the EG&G Report estimates the cost of U_3O_8 burial at the Hanford Site at \$1.81/kgU. (App. Exh. 8, at 51; LeRoy-Dekker at 26-27 fol. Tr. 1016.) Thus, Mr. LeRoy concluded that the LES estimate of \$1.00/kg U_3O_8 in 1993 dollars, which translates to \$1.27/kgU in 1994 dollars, falls squarely within the range of estimates in the EG&G Report of \$0.18 to \$1.81/kgU in 1994 dollars and remains reasonable today. (LeRoy-Dekker at 27 fol. Tr. 1016.)

Dr. Makhijani challenges the reasonableness of the Applicant's U_3O_8 burial cost estimate asserting that the estimate of the Applicant and the NRC Staff is not based on the Applicant's own plausible strategy for tails disposal. Rather, he asserts that while the LES disposal plan calls for deeper than surface burial, the two studies on which the LES and Staff estimates are based deal only with near-surface burial costs, not deep burial. (Makhijani at 4, 20 fol. Tr. 1081.)

While acknowledging that the disposal cost estimates in the Martin Marietta and the EG&G Reports are based on near-surface disposal, Mr. LeRoy testified that deep disposal should be no more costly than near-surface disposal because deep burial of U_3O_8 does not require expenses for engineered barriers and containers that are usually required for near-surface disposal. He stated that lower costs for deep disposal also would result from reduced security expenses based on the decreased likelihood of an intruder entering a deep burial site. (LeRoy-Dekker at 31-32 fol. Tr. 1016.) Similarly, the Staff witnesses, Mr. Faraz and Mr. Hickey, indicated that several factors will tend to decrease the cost of disposal for depleted uranium including the large volume and uniformity of tails; the economies of scale that will be possible if the CEC tails are buried with those from DOE; and savings in construction costs if the tails are disposed of in an existing underground cavity. (Faraz-Hickey at 10 fol. Tr. 1106.)

Based on this Applicant and Staff testimony, we find that it was not unreasonable for the Applicant to base its cost estimate for deep disposal on the near-surface cost estimates in the Martin Marietta and the EG&G Reports. Accordingly, we find that the LES cost estimate for burial of the CEC depleted uranium tails is a reasonable one.⁸

⁸ In its proposed findings, the Intervenor argues that the LES estimate for burial of U_3O_8 is also unreasonable because it fails to take into account the costs of siting, characterizing, and licensing a disposal facility. CANT PF at 36, CANT RF at 19-20. But the argument the Intervenor now makes in its proposed findings is not one it sought to support at the hearing with evidence. In making its evidentiary presentation, the Intervenor sought to demonstrate that neither LES nor the Staff had proposed or provided for the contingency that there

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3. DUF₆ Conversion Costs

The Applicant's estimate of \$12 million annually, or \$360 million over 30 years of operation, for the conversion of DUF₆ to U₃O₈ comprises the largest component of the LES tails disposal cost estimate of \$16.175 million per year or \$485.3 million over 30 years of operation. In their prefiled direct testimony, Mr. LeRoy and Mr. Dekker stated that "[t]he cost of conversion of DUF₆ to depleted U₃O₈ is based on an estimated conversion cost of \$4.86 per kilogram of uranium (\$1996), which was provided to LES by COGEMA, Inc., the U.S. affiliate of a French nuclear service company." (LeRoy-Dekker at 23-24 fol. Tr. 1016.) The Applicant's witnesses then stated that COGEMA has indicated to LES in writing its willingness to consider providing conversion services for UF₆ in the United States and that, in a 1991 letter (App. Exh. 5), COGEMA estimated that its charge for such services in 1991 dollars would be in the range of \$3 to \$5/kgU. (*Id.* at 24.) Referring to Applicant's Exhibit 6, they then stated that "[i]n its more recent letter, COGEMA provided an updated estimate of \$4 to \$6/kgU (\$1995), which is in line with LES' conversion cost estimate of \$4.86 (\$1996) (\$4.67 in \$1995)."⁹ (*Id.*) They also declared that these estimates assume construction and operation of a conversion facility in the United States and are based on COGEMA's actual experience in construction and operation of a commercial facility in France. (*Id.*) Mr. LeRoy and Mr. Dekker asserted that these cost estimates also are comparable to actual costs incurred by Urenco for conversion of UF₆ to U₃O₈ in Europe. Further, they testified that "[t]he estimate provided by COGEMA includes the understanding that COGEMA would assume responsibility [for the] handling of any non DU₃O₈ material produced during conversion (e.g., hydrofluoric acid-HF) [and] LES is responsible for dispositioning the DU₃O₈ only." (*Id.*) Finally, the LES witnesses declared that this practice is consistent with Urenco's actual conversion experience in Europe, where HF remained with the converter. (*Id.* at 25.)

At the hearing, the Intervenor's witness, Dr. Makhijani, challenged the validity of the Applicant's conversion cost estimate of \$4.86/kgU asserting, in

may be no disposal site available in 15 years or even 30 years at the end of the CEC license term. The Intervenor thus argued that a careful analysis of the safety and durability of UF₆ storage cylinders was necessary. (Makhijani at 22-23 fol. Tr. 1081.) There is ample evidence in the record about the safe useful life of UF₆ cylinders that addresses the Intervenor's concern about cylinders. (App. Exh. 7, at 9-11.) The Applicant and the Staff are not required to counter an evidentiary case that the Intervenor never made.

⁹The Applicant did not explain further the derivation of the LES conversion cost "estimate" provided to LES by COGEMA of a rather exact \$4.86/kgU in 1996 dollars or \$4.67 in 1995 dollars when the 1991 COGEMA letter (App. Exh. 5) and the subsequent 1995 letter (App. Exh. 6) referred, respectively, to a charge in the range of \$3-5/kgU and \$4-6. In responding to a June 18, 1993 Staff request for revised tails disposal cost estimates (App. Exh. 4h), however, the Applicant informed the Staff that "[t]he cost of conversion of DUF₆ to depleted uranium oxide (DU₃O₈) is based upon an estimate of \$4.00 per kilogram uranium. This estimate was provided to LES by COGEMA." (App. Exh. 4o.) It appears that the \$4.00 is merely the mid-range of COGEMA's 1991 estimate of \$3-5 escalated from 1991 to 1995 and 1996 dollars using the Applicant's standard 4% per year escalator yielding \$4.67 in 1995 dollars and \$4.86 in 1996 dollars.

effect, that the Applicant's failure to break the estimate into constituent parts precludes any evaluation of the estimate or its reasonableness. (Tr. 1205-06.) Specifically, he testified that the Applicant's \$4.86 figure understates the cost of conversion because it fails to include the considerable cost of approximately \$1.50/kgU for neutralizing to calcium fluoride ("CaF₂") the hydrofluoric acid ("HF") byproduct that is produced during the conversion of UF₆ to U₃O₈. (Tr. 1206-09.) Such neutralization costs were necessary he asserted because his past evaluation of the demand in the United States for hydrofluoric acid showed that it was a declining market. According to Dr. Makhijani, a very large use of HF is in the production of ozone-depleting chlorofluorocarbons ("CFCs") that now are being phased out pursuant to federal law and international agreements. Although recognizing that HF is used in the initial production of UF₆, Dr. Makhijani testified that large purchases of Russian high-enriched uranium for reactor fuel and the additional release of American stockpiles of high-enriched uranium will further drive down the domestic demand for HF by limiting the need for enrichment services. He further stated that a 1990 Oak Ridge report, "The Ultimate Disposition of Depleted Uranium" (DE 91-006414), that was published before the establishment of any firm deadlines for phasing out CFCs or the American purchase of Russian high-enriched uranium, concluded that there may be no market for contaminated hydrofluoric acid in the United States. Finally, Dr. Makhijani testified that converting high-enriched uranium in the form of uranium metal to reactor fuel can be done using conversion methods that either use or do not use HF and that the process for conversion in this country has yet to be selected.

On the basis of the evidentiary record in this proceeding, we cannot find that the Applicant's estimated cost of \$4.86/kgU (totaling \$12 million annually and \$360 million over 30 years of operation) is a reasonable estimate for converting DUF₆ to U₃O₈. The LES estimate is deficient because it fails to include the significant cost of neutralizing the hydrofluoric acid byproduct of the conversion process. The evidentiary record is clear that the Applicant's cost estimate for converting DUF₆ to U₃O₈ does not include any provision for incurring the additional substantial cost of neutralizing the byproduct HF from the primary conversion process. (LeRoy Tr. 1055, 1049. *See also* App. Exh. 7 at 17.)¹⁰ Instead the Applicant's position assumes that the COGEMA operation

¹⁰ The EG&G Report establishes that the conversion costs of neutralizing HF to CaF₂ are significant and contribute about \$1.50/kgU to the total conversion cost of \$8.40 in 1992 dollars. (App. Exh. 8, at 47; Hickey Tr. 1133-35.) This HF neutralization cost estimate in 1992 dollars is derived from the EG&G Report and excludes any construction or other miscellaneous fees. It also assumes that the disposal cost for CaF₂ is minimal due to its slight contamination and likelihood of disposal as ordinary waste. (Hickey Tr. 1134-35.) The Staff's witness, Mr. Hickey, agreed that the estimate of \$1.50/kgU for the neutralization of byproduct HF to CaF₂ was reasonable and that he had no other estimate to offer. (Tr. 1135.) Adding the conservative \$1.50 cost of HF neutralization to the Applicant's estimated costs for converting DUF₆ to U₃O₈ results in a more than 30% increase to the LES

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in France, in which HF is recycled as part of COGEMA's extensive nuclear fuel cycle manufacturing activities or otherwise marketed, will be replicated in the United States. It has not, however, provided any supporting evidence that there will be a sufficient market in the United States for the byproduct HF allowing it to be economically recycled or otherwise sold. Without evidence to show that there will be a sufficient market for the byproduct HF in the United States, we can only conclude that a domestic conversion facility, regardless of whether it is ultimately built and operated by COGEMA or some other entity, will have to neutralize the HF as an additional step in the conversion process and that the additional cost must be included in the cost of conversion. Thus, contrary to the assertions of the Applicant's witness that the conversion of HF to CaF_2 is not the Applicant's concern because COGEMA's cost estimate for UF_6 conversion includes the understanding that COGEMA would assume responsibility for all conversion byproducts except U_3O_8 (LeRoy at Tr. 1050), the reasonableness of the LES conversion cost estimate component is not "converter specific" and is not dependent upon COGEMA performing the service.¹¹

In making this finding, we are aware that the Applicant's witness, Mr. LeRoy, testified that in "the conversations we have had with COGEMA and in the SECY paper [SECY-91-019 (App. Ex. 3)], it is stated that COGEMA, after converting the DUF_6 to U_3O_8 uses the HF that is produced either for the forward process of converting natural U_3O_8 to natural UF_6 or the HF is sold on the industrial market." (Tr. 1049. *See also* Tr. 1050, LeRoy-Dekker at 29 fol. Tr. 1016.) But this proffer of the COGEMA model in France, with its extensive nuclear fuel reprocessing, manufacturing, and waste disposal activities under one government umbrella, is not sufficient to establish, without significant additional evidence, the feasibility or likelihood that a conversion facility in the United States could economically recycle or otherwise market the byproduct HF from the conversion of the CEC tails.

This failure of proof is especially significant in the circumstance where the domestic chemical market also will be faced with the byproduct HF from the conversion of the huge DOE stockpile of tails as well as the ever-increasing

conversion costs, increasing the Applicant's annual conversion costs from \$12 million to \$15.7 million and, over 30 years of operation, from \$360 million to \$471 million. The addition of this increase in conversion costs to the LES total tails disposal cost estimate increases it from \$485.3 million over 30 years of operation to almost \$600 million.

¹¹ Indeed, for this same reason we rejected the Intervenor's assertion in considering the Applicant's transportation cost estimate that the Applicant's disposal strategy was not plausible because LES did not have a firm commitment from COGEMA, Inc., to build a conversion facility in the United States. The Applicant offered no evidence that COGEMA, Inc., actually would build and operate a conversion facility in the United States. Rather, it only offered an expression of interest letter stating "COGEMA Inc.'s willingness to consider the possibility of providing, in the United States, conversion services." (App. Ex. 6.) Because the Applicant had no such commitment, the Intervenor asserted that the LES transportation estimate would have to include the costs of shipping the DUF_6 to France and returning the U_3O_8 to the United States. CANT RF at 21-22. The record indicates those costs would add some \$4-5 million a year to the LES transportation costs. (App. Ex. 4(l) at Appendix E, at E-2; LeRoy Tr. 1059-60.)

accumulation of tails from the United States Enrichment Corporation. Indeed, Mr. LeRoy indicated that the Applicant's cost projections for disposal did not include any analysis of the future market for conversion byproducts and he acknowledged that there could be a glut of such byproducts on the market in the future from tails conversion. (Tr. 1051.) He further conceded that the question of the cost of neutralization of HF is not irrelevant to the LES cost estimate. (Tr. 1055-56.) He thus provided nothing to counter effectively the testimony of the Intervenor's witness, Dr. Makhijani, that his past analysis showed the domestic market for HF was shrinking due to the phase out of CFCs and the decrease in demand for enrichment services from the introduction of Russian and American high-enriched uranium, *see* LBP-96-25, 44 NRC at 352-60, a conclusion he further buttressed with the 1990 Oak Ridge report indicating that there may be no market in the United States for byproduct HF.

Further, we note that in assessing the environmental impacts from the conversion of UF_6 to U_3O_8 , the Staff's FEIS assumes that the byproduct HF will be neutralized to CaF_2 . (Staff Exh. 2, at A-2 to -4.) More important, however, is the Staff's response in the FEIS to public comments on the draft environmental impact statement concerning the decline in the American market for HF. The Staff described the sale of HF as merely a "possibility" (Staff Exh. 2, Vol. 2, at 1-198) and went on to state in responding to comments about the impacts of transporting HF that "[c]onversion operations would likely result in production of calcium fluoride." (*Id.* at 1-199.) Similarly, the 1994 EG&G Report introduced by the Applicant that evaluates the disposal options and costs for DOE's depleted uranium and estimates \$8.40/kgU as the cost of conversion assumes that all byproduct HF from the conversion of UF_6 to U_3O_8 is neutralized by converting it to CaF_2 and disposing of it in that form. (App. Exh. 8, at 43, 47.) Accordingly, on the basis of this evidentiary record, we cannot find that the Applicant has met its burden of proof and demonstrated by a preponderance of the evidence that the LES cost estimate for the conversion of DUF_6 to U_3O_8 is a reasonable one because it fails to include the substantial costs for neutralizing the byproduct HF from the conversion process.¹²

¹²In this regard, we note that the Staff's witness, Mr. Hickey, testified that the Applicant's "estimate of \$4.86 per kilogram [of uranium] for conversion, we believe, includes the possibility that the option of converting to calcium fluoride will be exercised." (Tr. 1130-31.) Besides being contradicted by the Applicant's testimony (LeRoy Tr. 1055), Mr. Hickey's assertion to the effect that the LES conversion cost estimate covers both the conversion of DUF_6 to U_3O_8 and the conversion of HF to CaF_2 is not supported by the record as a whole.

Further, Mr. Hickey opined that the Applicant's conversion cost estimate of \$4.86 nevertheless was adequate to cover the additional cost of converting byproduct HF to CaF_2 , stating

[t]he prices that were quoted to us from LES that came from COGEMA, we believe were over-inflated and included a lot of profit on the part of COGEMA. And, in fact, a conversion facility could be built in the United States, and they could dispose of the hydrogen fluoride in the form of calcium for less than \$5 a kilogram.

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Finally, we note that, in contrast to the detailed final decommissioning plan that LES must submit near the end of the license term, the Applicant's Decommissioning Funding Plan is required only to provide a reasonable cost estimate to ensure that the Applicant sets aside adequate funds to cover, *inter alia*, the cost of tails disposal. The reasonableness of the Applicant's cost estimate is necessarily dependent upon all the circumstances and the Commission has indicated that "the plan must contain essential elements sufficient to ensure that a reasonable estimate of decommissioning costs can be made." *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-88-10, 28 NRC 573, 587 (1988). Here, the largest component of the Applicant's estimate for tails disposal is that for the conversion of DUF_6 to U_3O_8 . As we have found, however, the Applicant's estimate has not properly accounted for neutralizing the byproduct HF as part of its estimate. This additional cost is substantial and it is not the type of expense, like an increase for inflation or the development of a new technology (*see* 50 Fed. Reg. 5600, 5604 (1985)), that merely should be added sometime in the future after one of the Applicant's periodic decommissioning funding reviews that the Applicant is committed to performing at least once every 5 years. (App. Exh. 1(e), at 7-1.) Rather, the neutralization of the byproduct HF produced as part of the conversion of DUF_6 to U_3O_8 is clearly an essential element of the conversion cost (and hence the tails disposal cost) that reasonably can be estimated at this time.

Further, because the depleted uranium tails are created as the Applicant performs enrichment services, the Applicant's tails disposal funds must come from a portion of the price charged by LES for the separate work units ("SWUs") it performs. (Arnold Tr. 672-73; App. Exh. 4n, at 4; App. Exh. 1(a), at 11.8-15; Staff Exh. 1, at 15-21.) In order to provide reasonable assurance that there are adequate funds set aside to cover tails disposal, the Applicant must factor the realistic reasonable cost estimate of tails disposal into its market price for SWUs from the initiation of operations. (App. Exh. 4n at 4.) This is especially important in light of the nature of the enrichment market and the Applicant's financial structure. As we found in LBP-96-25, 44 NRC at 355-56, 359-60, 361,

(Tr. 1131.) Mr. Hickey then used the conversion cost estimate in the EG&G Report of \$8.40 that includes byproduct HF neutralization to illustrate his assertion. (Tr. 1131, 1135-36; App. Exh. 8, at 47.) According to Mr. Hickey, after 5 years of operation of the hypothetical conversion facility in the EG&G Report, the initial plant costs would have been recovered and, thereafter, the cost per kilogram for conversion would amount to about \$4.80. (Tr. 1136.) But Mr. Hickey attempts to prove too much. He not only failed to escalate his estimate from the 1992 dollars of the EG&G Report to the 1996 dollars of the LES estimate — a step that raises his estimate considerably — but his assumptions about the EG&G Report estimate (assumptions that are not explicit in the EG&G Report) raise more questions than are answered regarding such things as return of capital, depreciation, carrying costs, taxes, decontamination costs, and profit margins. Because the record provides no corroborating support for the proposition that a future domestic conversion facility is to be built and operated without a healthy regard for profits, we are unable to accept Mr. Hickey's assertions regarding the cost of conversion of depleted uranium tails, including the neutralization of byproduct HF. In so concluding we are not unmindful of Mr. Hickey's candid appraisal that the Staff's forecasting accuracy of disposal costs has been "very poor." (Tr. 1153.)

the enrichment market is a fiercely competitive, international one in which the supply of enrichment production capacity and the supply of enriched uranium far exceeds demand and this situation will prevail for the foreseeable future. In such a competitive market, a significant shortfall in the funds set aside to pay for tails disposal cannot simply be remedied by a price increase without harming the Applicant's competitive position and future market prospects. Moreover, unlike a utility reactor operator that can rely upon a public utility commission to set a rate structure adequate to recover all decommissioning costs even after the shutdown of a facility (*see* 53 Fed. Reg. 24,018, 24,031 (1988)), the Applicant's tails disposal funds can only be collected from its charges for enrichment services on an ongoing basis.

In other words, LES must be totally self-reliant in paying for tails disposal. As we detailed in LBP-96-25, 44 NRC at 378-80, LES is a newly formed entity created to build and operate the CEC. It is structured as a limited partnership and LES has no significant independent assets. *Id.* at 398-99. Similarly, none of the LES general or limited partners are corporations of worth. *Id.* Further, under the LES Partnership Agreement, as well as general principles of corporate and partnership law, the corporate parents and other affiliates of the LES general and limited partners have no liability for the obligations of the partnership. *Id.* at 402 n.30. In these circumstances, we cannot conclude that the Applicant's tails disposal estimate need only be a rough approximation that can be adjusted in the future upon periodic reviews by the Applicant. Rather, for the LES tails disposal estimate to be a reasonable one, it must include the substantial cost of neutralizing the HF from the conversion of DUF_6 to U_3O_8 . Our finding in this regard is without prejudice to the Applicant acting to amend the LES Decommissioning Funding Plan consistent with this Decision and the Commission's regulations.

C. Intervenor's Other Challenges

In addition to its direct challenge to the Applicant's tails disposal cost estimate,¹³ the Intervenor also challenges the Staff's FEIS alleging that a number of technical deficiencies and other shortcomings undermine its validity, thereby discrediting the LES tails disposal estimate for deep burial of the CEC tails. According to the Intervenor, these various deficiencies so eviscerate the Staff's analysis that the FEIS cannot support the conclusion that deep burial of the

¹³ At the hearing, the Intervenor did not pursue the specific assertions set forth in CANT's original bases B.4 and B.5 and the Intervenor did not include findings on these bases in filing its proposed findings. Hence, the Intervenor has waived these claims and, pursuant to 10 C.F.R. § 2.754(b), is in default as to these claims. In any event, the Applicant and the Staff presented testimony and other evidence on these matters. (LeRoy-Dekker at 15-18, 43-47 fol. Tr. 1016; App. Exh. 1(a), at 11.8-10 to -16; Faraz-Hickey at 11-12 fol. Tr. 1106) and the Applicant has met its burden of proof on these claims. Hence, the claims in Intervenor's bases B.4 and B.5 cannot be sustained.

CEC depleted uranium tails will provide adequate protection to the public and the environment. Consequently, the Intervenor asserts that the CEC tails must be disposed of in a licensed geologic repository at a cost likely to be no less than \$10/kg U_3O_8 and perhaps much more. (Makhijani at 4-7, 16-17, 20-21 fol. Tr. 1081.)¹⁴ We summarily address below the deficiencies in the FEIS alleged by the Intervenor and find them without merit.

1. Use of Inappropriate pH, Retardation Factor, and Redox Potential Values

Dr. Makhijani asserts that the values chosen by the Staff for groundwater regarding pH, retardation factor, and redox potential for use in its FEIS analysis of the environmental impacts of deep disposal of depleted uranium tails at two representative sites (*see supra* pp. 107-08) could result in a serious underestimation of the doses to the public. (Makhijani at 8-13 fol. Tr. 1081.) Specifically, Dr. Makhijani claims that the pH value — an important factor governing uranium solubility and subsequent uranium transport — of 7.8 that was used by the Staff came from near-surface water data from a location in New York. (*See* Staff Exh. 2, Appendix A, at A-12.) According to Dr. Makhijani, the pH of groundwater in the basalt rock formations for repository locations has been found to be greater than 9. (Makhijani at 9-10 fol. Tr. 1081.) Contrary to Dr. Makhijani's assertion, however, we find that the Staff's use of a pH value of 7.8 based on New York data was not unreasonable in light of the reference literature for groundwater showing a pH range of 7.2 to 8.5. (Price Tr. 1115; LeRoy Tr. 1164-65.) Thus, the Staff's use of a pH value falling within the reference range was appropriate and reasonable.

Dr. Makhijani also argues that a retardation factor of 1200 should not have been used by the Staff in the FEIS (Staff Exh. 2, Appendix A, at A-13) because it is considerably higher than the retardation factors for granite and basalt rock formations recommended in a report of the National Academy of Sciences. (Makhijani at 10 fol. Tr. 1081.) The retardation factor is determined by dividing the ratio of water velocity by the radionuclide transportation velocity. Radionuclides dissolved in groundwater are adsorbed and exchanged through contact with the surrounding solid phase and thus travel at a lower velocity than

¹⁴The Applicant argues that its tails disposal cost estimate was developed before the environmental impact statement was prepared and was not based on any information in the FEIS. It implies, therefore, that the Intervenor's challenges to the technical underpinnings of the FEIS are irrelevant to the LES disposal cost estimate for deep burial. (Tr. 1066; Applicant's Proposed Findings of Fact and Conclusions of Law (May 26, 1995) at 402-03.) But the Applicant's position ignores the thrust of the Intervenor's argument that because of numerous deficiencies in the Staff's analysis deep burial of U_3O_8 has not been shown to protect the environment thereby mandating disposal in a geologic repository at a much higher cost. The Applicant's witnesses also testified, however, that on the basis of their review of the Staff's analysis of deep disposal in the FEIS they found the analysis satisfactory. (Dubiel-Donelson at 11-15 fol. Tr. 1026.)

the groundwater. (Staff Exh. 2, Appendix A, at A-13.) The Staff's witness, Dr. Price, as well as the Applicant's witness, Mr. Dubiel, both testified that the value used by the Staff, which was based on a Swedish study, was appropriate because the data were from actual experimental observation for a comparable medium and were corroborated by a second study using such data. (Price Tr. 1115-17, 1235; Dubiel Tr. 1164-65.) Based on this testimony, we cannot find that the Staff's use of a retardation factor of 1200 drawn from actual experimental data, in contrast to theoretical evaluations, was unreasonable.

Dr. Makhijani next claims that the redox potential value ("eH") of minus 100 millivolts used by the Staff in its FEIS analysis (Staff Exh. 2, Appendix A, at A-12) is outside the range of values that the Staff otherwise lists in the FEIS for uranium mines and the FEIS contains no other comparative groundwater eH values. He asserts that the solubility of uranium is critical to the determination of the amount of uranium in groundwater and that the Staff has made arbitrary assumptions that tend to minimize the amount of uranium in solution. (Makhijani at 10-12 fol. Tr. 1081.)

Redox potential, measured in volts or millivolts ("mV"), is a measure of the potential of groundwater to oxidize or reduce (i.e., to change chemically materials disposed of in groundwater). An increased redox potential increases the potential for uranium to dissolve in water. (*Id.* at 11; Price Tr. 1118.) Although the Staff's comparative table of eH values in the FEIS and the Staff's choice of an eH value of minus 100 mV certainly could have been more clearly explained in the FEIS (Price Tr. 1148-49), we find Dr. Makhijani's criticism without merit. As Dr. Price testified, the Staff chose an eH value of minus 100 mV because it was representative of deep groundwater from experimental observations showing redox potentials of minus 26 mV to minus 210 millivolts, with some reference data going even lower. (Tr. 1118-19.) He stated that the data set forth in the FEIS for uranium mines are not fully representative of deep groundwater and the conditions that will be chosen and prevail for the deep burial of depleted uranium tails will be a reducing environment. (Tr. 1145-49.) The Applicant's witness, Mr. Dubiel, also testified that the reference literature supported the Staff choice of eH value for the groundwater depths involved in the FEIS evaluation. (Tr. 1165-66.) Based on this testimony, we find that the eH value used by the Staff in its analysis is a reasonable one.

2. Failure to Perform Uncertainty Analysis, Consider Range of Geologic Settings, and Fully Analyze Appropriate Chemical Form of Tails for Disposal

Dr. Makhijani next asserts that, contrary to sound scientific practice, the Staff failed to perform an uncertainty analysis of deep burial as part of its environmental impact analysis so that upper and lower bounds for estimated

doses could be obtained. Because of this failure, he asserts that the resulting Staff analysis fails to meet the minimal test of sound science. (Makhijani at 13-16 fol. Tr. 1081.)

In response to this criticism, Dr. Price testified that an uncertainty analysis was impractical and unnecessary here because an actual deep burial site was not being characterized. Rather, he stated that the objective of the Staff's analysis in the FEIS was not to support a licensing position on a disposal site but merely to determine the plausibility of deep burial of depleted uranium as a disposal strategy. Indeed, Dr. Price noted that the analogous NRC branch technical position for low-level waste facilities requires significant site-specific data for the performance of an uncertainty analysis. (Tr. 1120-21.) In these circumstances, we cannot find that an uncertainty analysis was necessary for the Staff's evaluation of the impacts from two representative hypothetical disposal sites.

Further, the Intervenor's witness claimed that the FEIS analysis is deficient for considering only two geologic settings, a granite formation and a basalt formation, instead of considering a wide range of potential geologic settings. Dr. Makhijani indicated that the Staff first should have performed a preliminary screening of all potential geologic settings for their respective advantages and disadvantages and only then selected particular rock types for study. (Makhijani at 9 fol. Tr. 1081.) The Staff witnesses, Dr. Price and Mr. Faraz, both testified that the use of two representative geologic settings was appropriate because the objective of the FEIS analysis was to determine whether deep burial of depleted uranium tails was plausible. (Tr. 1112-13.) All of the Applicant's witnesses concurred in this same view. (Tr. 1163.) Contrary to Dr. Makhijani's charge, we find that the Staff's use of two representative geologic settings was reasonable in light of the purpose of the FEIS evaluation.

Finally, Dr. Makhijani asserts that the Staff's analysis is deficient for failing to consider the appropriateness of converting UF_6 to UO_2 instead of U_3O_8 for disposal. Although he concedes that both uranium oxide forms are insoluble in water, Dr. Makhijani asserts that the complexes they form with other chemicals in specific geologic environments could be different, depending on the particular conditions. Therefore, he claims the Staff should have considered UO_2 in addition to U_3O_8 and presented a comparative analysis showing the legitimacy of its choice of U_3O_8 . (Makhijani at 7-8 fol. Tr. 1081.)

Dr. Makhijani's assertion is without merit. The record evidence overwhelmingly demonstrates that U_3O_8 is the preferred form of uranium oxide for disposal. (App. Exh. 41, at 18-19 & Appendix D, at D-1; App. Exh. 7, at 14-15; App. Exh. 8, at 11-13; LeRoy-Dekker at 30 fol. Tr. 1016.) Further, as Dr. Price testified, it is also necessary to consider how to manage and handle the uranium oxide as it is produced, stored, and transported for burial, and U_3O_8 is more stable upon exposure to the atmosphere than UO_2 . (Tr. 1111.) Indeed, as Applicant's

Exhibit 7 states "UO₂ will ignite spontaneously in heated air and burn brilliantly." (App. Exh. 7, at 36.)

Finally, in addition to the foregoing findings, we have carefully considered all of the other arguments, claims, and proposed findings of the parties relative to contentions B and J.3 and find that they are either without merit, immaterial, or unnecessary to this Decision.

D. Concerns of the State of Louisiana

Pursuant to 10 C.F.R. § 2.715(c) of the Commission's Rules of Practice, the State of Louisiana has participated in this proceeding as an interested State. In its proposed findings, the State has requested that we condition any LES license for the CEC to ensure that Louisiana does not have to take responsibility for any radioactive waste from the CEC. Additionally, the State requests a number of corollary conditions designed to ensure that no financial obligations fall on Louisiana from any of the CEC radioactive waste.¹⁵

The State's concern that any LES license authorization be conditioned so that the State cannot be held responsible for any radioactive waste from the CEC has now been resolved by the recent enactment of the USEC Privatization Act. The Act specifically provides that "[n]otwithstanding any other provision of law, no State or interstate compact shall be liable for the treatment, storage, or disposal of any low-level waste . . . attributable to the operation, decontamination, or decommissioning of any uranium enrichment facility." 42 U.S.C. § 2297h-11(c). With the enactment of this federal statute, no further consideration of the State's request for license conditions is necessary.

III. CONCLUSION

For the reasons detailed in Part II.B.3, we conclude that the Applicant's cost estimate of \$12 million annually for the conversion of DUF₆ to U₃O₈ is not a reasonable one given its failure to include the substantial costs of neutralizing the conversion process byproduct hydrofluoric acid. Thus, to this extent, the Intervenor's contention B.1 is sustained. For the same reason and to the same extent, the Intervenor's contention J.3 is sustained and, pursuant to 10 C.F.R. § 51.102, the FEIS is hereby supplemented by the discussion of the economic costs of tails disposal in this Decision and the underlying adjudicatory record. See *Philadelphia Electric Co. (Limerick Generating Station, Units 1 and 2)*, ALAB-819, 22 NRC 681, 706 (1985).

¹⁵ Louisiana's Proposed Findings of Fact and Conclusions of Law in the Form of an Initial Decision Relative to DUF₆ Waste Generated at the Proposed LES Facility (June 23, 1995) at 3-5.

Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this Partial Initial Decision will constitute the final Decision of the Commission on these contentions forty (40) days from the date of its issuance unless a petition for review is filed in accordance with 10 C.F.R. § 2.786, or the Commission directs otherwise. Within fifteen (15) days after service of this Partial Initial Decision, any party may file a petition for review with the Commission on the grounds specified in 10 C.F.R. § 2.786(b)(4). The filing of a petition for review is mandatory in order for a party to have exhausted its administrative remedies before seeking judicial review at the appropriate time. Within ten (10) days after service of a petition for review, any party to the proceeding may file an answer supporting or opposing Commission review. The petition for review and any answers shall conform to the requirements of 10 C.F.R. § 2.786(b)(2)-(3).

It is so ORDERED.

**THE ATOMIC SAFETY AND
LICENSING BOARD**

**Thomas S. Moore, Chairman
ADMINISTRATIVE JUDGE**

**Richard F. Cole
ADMINISTRATIVE JUDGE**

**Frederick J. Shon
ADMINISTRATIVE JUDGE**

**March 7, 1997
Rockville, Maryland**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

**G. Paul Bollwerk, III, Chairman
Peter B. Bloch
Thomas D. Murphy**

In the Matter of

**Docket No. 50-461-OLA
(ASLBP No. 97-725-01-OLA)**

**ILLINOIS POWER COMPANY and
SOYLAND POWER COOPERATIVE
(Clinton Power Station, Unit 1)**

March 11, 1997

In this proceeding regarding the proposed transfer of the ownership share of Clinton Power Station minority owner Soyland Power Cooperative to majority owner Illinois Power Company, the Licensing Board grants the unopposed request of Petitioner Southwestern Electric Cooperative, Inc., to dismiss its protective intervention petition and terminate the proceeding.

RULES OF PRACTICE: INTERVENTION PETITION

Simply because a filing is labeled a petition to intervene does not prevent the presiding officer from treating it as a request to initiate a hearing if this, in fact, is what the petitioner is seeking. *See Yankee Atomic Electric Co. (Yankee Nuclear Power Station), CLI-96-1, 43 NRC 1, 5 (1996).*

MEMORANDUM AND ORDER

(Terminating Proceeding)

Responding to a January 23, 1997 notice of opportunity for hearing, *see* 62 Fed. Reg. 4437 (1997), in a February 28, 1997 filing entitled "Petition for Leave to Intervene," Petitioner Southwestern Electric Cooperative, Inc. (Southwestern), sought leave to participate in any adjudicatory proceeding convened in connection with an October 17, 1996 application (as supplemented and modified by letter dated December 31, 1996) for agency approval of an operating license amendment for the Clinton Power Station, Unit No. 1 (CPS). The proposed license revision would permit the transfer of Soyland Power Cooperative's (Soyland) minority ownership in CPS to Illinois Power Company (Illinois Power), the facility's majority owner and operator. On March 7, 1997, this Licensing Board was established to rule on Southwestern's petition. *See* 62 Fed. Reg. 11,933 (1997).

Subsequently, on March 11, 1997, Petitioner Southwestern filed a letter addressed to the Licensing Board requesting that this proceeding be terminated. In support of its motion, Southwestern asserts that its original petition was intended only to preserve its interests in the event that Illinois Power, Soyland, or some other party sought and was granted a hearing.¹ No other party apparently having filed a timely hearing request, Southwestern now wishes to have this proceeding terminated.²

Under the circumstances, we grant Southwestern's request.

For the foregoing reasons, it is, this 11th day of March 1997, ORDERED that:

1. The March 11, 1997 motion of Southwestern to terminate this proceeding is *granted*; and

¹ Although Southwestern's February 28 filing was labeled as a "petition to intervene," this would not prevent us from treating it as a request to initiate a hearing if this, in fact, was what Southwestern was seeking. *See Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-1, 43 NRC 1, 5 (1996).

² By telephone this date, we were advised by counsel for Southwestern that neither Illinois Power nor Soyland objects to the termination of this proceeding. Also, upon inquiry, counsel for the NRC Staff advised the Board that the Staff has no objection to termination of this proceeding.

2. Southwestern's February 28, 1997 petition for leave to intervene is *dismissed* and this proceeding is *terminated*.

THE ATOMIC SAFETY AND
LICENSING BOARD³

G. Paul Bollwerk, III, Chairman
ADMINISTRATIVE JUDGE

Thomas D. Murphy
ADMINISTRATIVE JUDGE

Rockville, Maryland
March 11, 1997

³ Administrative Judge Bloch was not available to sign this Memorandum and Order. He was, however, advised of its contents and approved its issuance.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

G. Paul Bollwerk, III, Presiding Officer
Jerry R. Kline, Special Assistant

In the Matter of

Docket No. 30-02764-MLA
(ASLBP No. 97-722-01-MLA)

UNIVERSITY OF CINCINNATI
(Denial of License Amendment)

March 27, 1997

MEMORANDUM AND ORDER
(Dismissing Proceeding)

In this proceeding, Licensee University of Cincinnati (University) has challenged the December 12, 1996 action of the NRC Staff denying the University's January 5, 1996 application for an amendment to its 10 C.F.R. Part 30 byproduct materials license. The requested amendment would allow specified visitors of radiation therapy patients to receive a dose of up to 500 millirem (mrem) total effective dose equivalent (TEDE) per year instead of the current public dose limit of 100 mrem per year provided for in 10 C.F.R. § 20.1301(a)(1).

Now pending before me is the March 13, 1997 motion of the University requesting that I dismiss this proceeding. In its motion, the University declares that on February 14, 1997, the NRC Staff issued Amendment No. 80 to the University's license (NRC License No. 34-06903-05), a copy of which was provided on March 20, 1997. *See* Presiding Officer Memorandum (Mar. 26, 1997), attachs. 1-2. Under License Condition 27 provided for by that amendment, an individual visiting a patient is permitted to receive 500 mrem during the patient's confinement period provided:

- (1) the visitor has been determined by a physician to be necessary for the emotional and/or physical support of the patient;
- (2) the visitor is 18 years of age or older and, if female, is not pregnant;
- (3) the visitor (a) is instructed to maintain exposures as low as is reasonably achievable (ALARA), emphasizing the basic radiation safety precautions of time, distance, and shielding, and (b) is advised (i) that the exposures received may exceed the general public's regulatory limit, and (ii) of the risks of radiation exposure; and
- (4) a visitor's exposures received under the license condition are estimated by appropriate means to ensure the 500 mrem dose limit is not exceeded, with records documenting compliance maintained for three years.

The University's motion also states that the Staff has no objection to the University's dismissal request.

The controversy in this proceeding has been mooted by the issuance of the February 14, 1997 license amendment. Accordingly, the University's dismissal request is granted and this proceeding is terminated.

For the foregoing reasons, it is, this 27th day of March 1997, ORDERED that:

1. The March 13, 1997 motion of the University to dismiss this proceeding is *granted*.
2. This proceeding is *dismissed*.

G. Paul Bollwerk, III, Presiding Officer
ADMINISTRATIVE JUDGE

Rockville, Maryland
March 27, 1997

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

**Peter B. Bloch, Presiding Officer
Peter Lam, Special Assistant**

In the Matter of

**Docket No. 55-20726-SP
(ASLBP No. 96-721-01-SP)
(Re: Operator License)**

**RALPH L. TETRICK
(Denial of Application for Reactor
Operator License)**

March 27, 1997

The Presiding Officer denied the Staff's motion for reconsideration. He ruled that the Staff should reasonably have foreseen the importance of whether or not to round up applicant's examination score. Consequently, Staff should have raised this question earlier and it was untimely to do so in a Motion for Reconsideration. Since the Presiding Officer also concluded that there was no important safety issue involved, he used his discretion to deny the untimely motion.

**CORRECTED COPY OF MEMORANDUM AND ORDER
(Denial of Reconsideration, Stay)**

On March 10, 1997, the Staff of the Nuclear Regulatory Commission filed a motion, "NRC Staff's Request for Issuance of an Order Staying the Effectiveness of the Presiding Officer's Initial Decision (LBP-97-2)" (Motion for a Stay). The Staff asked that the Presiding Officer issue an order staying the effectiveness

of his Initial Decision in this proceeding,¹ pending the Presiding Officer's review and consideration of the Staff's Motion for Reconsideration (Motion for Reconsideration), filed simultaneously. Ralph L. Tetrick filed his response to the Staff motions on March 17, 1997.

Because the Motion for Reconsideration has been filed, we retain jurisdiction over this case. See 10 C.F.R. § 2.771; *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-235, 8 AEC 645 (1974).

I have decided that the Motion for Reconsideration shall be denied because it improperly raises an argument based on evidence that should have been incorporated in the record earlier in this case. The Motion for a Stay also shall be denied. The Motion for a Stay stated, in part, that it was pending "the Presiding Officer's review and consideration of the Staff's Motion for Reconsideration." Upon denial of the Motion for Reconsideration, I no longer have jurisdiction of this case, so it would be inappropriate to grant a stay "pending consideration by the Commission," as the Staff also requests.

With respect to the Motion for Reconsideration, I note that:

A motion for reconsideration should not include new arguments or evidence unless a party demonstrates that its new material relates to a Board concern that could not reasonably have been anticipated.

Texas Utilities Electric Co. (Comanche Peak Steam Electric Station, Units 1 and 2), LBP-84-10, 19 NRC 509, 517-18 (1984). (Emphasis added to the quoted paragraph by the Staff. See "NRC Staff's Response to Memorandum and Order of March 21, 1997," March 25, 1997 [Staff Response] at 2.) In this case, Staff opposed Mr. Tetrick's challenges to three questions on its Senior Reactor Operator's examination. It now argues that it could not anticipate that one of these three questions might be struck, forcing the Presiding Officer to decide whether or not a score of 79.59% should be considered passing or failing.² NRC Response at 3-4. We reject Staff's argument that it "did not yet have any reason to anticipate that the Presiding Officer would strike Question 96. . . ." (NRC Response at 4.) The key question being litigated was the validity of each of the challenged questions and whether or not Mr. Tetrick would pass the examination. I conclude that the Staff should have anticipated this contingency and presented arguments about how it should be resolved. In the interest of finality in decision making, I do not consider it appropriate to permit the Staff to raise this argument at this stage of the proceeding.

¹ *Ralph L. Tetrick* (Denial of Application for Reactor Operator License), LBP-97-2, 45 NRC 51 (1997) (Initial Decision).

² It was necessary to the decision in this case for the Presiding Officer to determine whether or not to round off the examination score. The Staff suggestion that this decision was "*sua sponte*" is frivolous.

In making this ruling, I recognize that Mr. Tetrick will be granted a license while other candidates, with scores between 79.5% and 80.0%, were denied a license. NUREG-1021, "Operator Licensing Examiner Standards," sets forth that "80% of the questions must be correctly answered." Motion for Reconsideration at 5. Only recently, the Staff has amended its NUREG to require a passing score of "80.00" percent, changing the number of significant digits in the NUREG itself from a whole percentage to 1/100th of a percentage point. Motion for Reconsideration, attached Supplemental Affidavit of Brian Hughes at 5, ¶ 10. At the time that Mr. Tetrick took his examination, the revised NUREG was not in effect and there was no published guidance, other than the NUREG itself, concerning the number of significant digits in an examination score or how a score should be rounded. I find, as the Staff suggests, that the Staff had an established practice — *first presented to the Presiding Officer only after issuance of the Initial Decision*. The Staff practice, which may be inconsistent with the use of a whole percentage point standard ("80%") in the NUREG,³ has required applicants to achieve a grade of 80% or greater — without rounding off — in order to pass their written examination. Staff Motion for Reconsideration at 5; Supplemental Affidavit of Brian Hughes at 8-10.

If this matter seriously affected public safety, I would consider this evidentiary point even though it is untimely. *See Midland, supra*. However, I have no reason to believe that a 0.41% difference in the score of a candidate on one portion of his examination is a valid reason for concern that his performance will be inadequate.

This decision also will have little effect on the Staff's use of a uniform passing grade. It is necessary to establish and consistently apply a passing grade for examinations, and the Staff has clarified the precise passing grade by amending the NUREG. Candidates whose scores fall even a fraction of a point below the passing grade should fail, even though they are not measurably inferior to candidates who pass by a fraction of a point. In this case, I have not decided the merits of the Staff argument about the interpretation of "80%" in a NUREG that is no longer current. My decision is based on the untimeliness of the argument and does not affect future cases. There is no reason to suspect a substantial negative effect on public safety because Mr. Tetrick had a written examination score of 79.59%, rounded off to 80% through a permissible interpretation of the language of the applicable version of NUREG-1021. I am confident that Mr. Tetrick, who has capably and respectfully conducted himself in this proceeding, will continue to improve his skills and that he will not permit his marginal score

³There is a strong presumption that the plain language of a statute or, by analogy, of regulatory guidance expresses the intent of its drafters. *Ardestani v. INS*, 112 S. Ct. 515, 116 L. Ed. 2d 496 (1991). It is appropriate to look to an extrinsic aid, such as Staff practice, only if the language of the regulatory guidance is unclear or if its apparent clarity leads to absurd results. *Blue Cross and Blue Shield of Alabama v. Weitz*, 913 F.2d 1544, 1548, *reh'g denied*, 921 F.2d 283 (1990).

on the written examination to interfere with his being an outstanding Senior Reactor Operator.

ORDER

For all the foregoing reasons and upon consideration of the entire record in this matter, it is, this 27th day of March 1997, ORDERED that:

1. The "NRC Staff's Motion for Reconsideration," March 10, 1997, is denied.

2. The "NRC Staff's Request for Issuance of an Order Staying the Effectiveness of the Presiding Officer's Initial Decision (LBP-97-2)," March 10, 1997, is denied.

3. The Staff of the Nuclear Regulatory Commission may issue to Mr. Ralph L. Tetrick a Senior Reactor Operator License for Turkey Point Nuclear Generating Plant, Units 3 and 4.

4. Because of the issuance of housekeeping stays in this case, March 27, 1997, shall be considered the date of issuance of the Initial Decision (LBP-97-2) for the purpose of calculating parties' rights and obligations concerning an appeal.

5. Pursuant to 10 C.F.R. § 2.1251, this Initial Decision constitutes the final action of the Commission thirty (30) days after March 27, 1997, unless any party petitions for Commission review in accordance with section 2.786 or the Commission takes review of the Decision sua sponte. If there is no petition for review, the date on which this decision will become final is Monday, April 28, 1997.

6. Pursuant to 10 C.F.R. § 2.786, a petition for review must be filed within fifteen (15) days after service of this Memorandum and Order, which is considered served on the date it is mailed, pursuant to 10 C.F.R. § 2.712(e). However, since service of this Decision is by mail, five days shall be added to the prescribed period of response, pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the petition for review must be served is Wednesday, April 16, 1997. Service of the petition for review must, pursuant to this Order, be made by express mail.

7. A petition for review and a response to a petition for review must meet the requirements of 10 C.F.R. § 2.786.

8. If a petition for review is filed, the answer must be filed within 10 days. Since the petition for review shall be filed by express mail, two days shall be added to the period of response pursuant to 10 C.F.R. § 2.710, which governs the computation of time. Consequently, the date the answer must be served is

Monday, April 28, 1997. Service of the answer must, pursuant to this Order, be made by express mail.

**Peter B. Bloch, Presiding Officer
ADMINISTRATIVE JUDGE**

Rockville, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of

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

WISCONSIN ELECTRIC POWER
COMPANY
(Point Beach Nuclear Plant,
Units 1 and 2)

Docket Nos. 50-266
50-301
72-5
March 4, 1997

The Director of the Office of Nuclear Reactor Regulation denies the request by Petitioner Fawn Shillinglaw, filed pursuant to 10 C.F.R. § 2.206, that the NRC take action to prohibit loading of VSC-24 casks at any nuclear site until the multiassembly sealed basket #4 at the Palisades nuclear plant has been unloaded and the experience evaluated for potential safety improvements. The Director concludes that the NRC will not permit unloading of any casks until it obtains reasonable assurance, through a variety of means, of each licensee's ability to do so safely, and therefore need not suspend any licensee's use of the general license for dry cask storage until the multiassembly sealed basket at Palisades has been unloaded.

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

I. INTRODUCTION

On November 17, 1995, Ms. Fawn Shillinglaw (Petitioner) 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 action to prohibit loading of VSC-24 casks at any nuclear site until the multiassembly sealed basket (MSB) #4 at the Palisades plant has been unloaded and the experience evaluated for potential safety improvements. In addition to Consumers Power Company, the Licensee for Palisades, other licensees that use the VSC-24 cask system are Wisconsin Electric Power Company at its Point Beach Nuclear Plant, Units 1 and 2, and Entergy Operations, Inc., at Arkansas Nuclear One, Units 1 and 2.

The petition has been referred to me pursuant to section 2.206. The NRC letter to the Petitioner dated January 18, 1996, acknowledged receipt of the petition. Notice of receipt was published in the *Federal Register* on January 25, 1996 (61 Fed. Reg. 2269).

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

II. BACKGROUND

NRC regulations contain a general license that authorizes nuclear power plants licensed by the NRC to store spent nuclear fuel at the reactor site in storage casks approved by the NRC. (*See* 10 C.F.R. Part 72, Subpart K.) In regard to dry cask storage of spent nuclear fuel at Palisades, Point Beach, and Arkansas Nuclear One, the Licensees opted to use the VSC-24 Cask Storage System designed by Sierra Nuclear Corporation. The VSC-24 Cask Storage System was added to the list of NRC-certified casks in May 1993 (58 Fed. Reg. 17,948). The associated certificate of compliance, Certificate No. 1007, specifies the conditions for use of VSC-24 casks under the general license provisions of Part 72. Section 1.1.2, "Operating Procedures," in the certificate of compliance for the VSC-24 casks requires that licensees prepare an operating procedure related to cask unloading. Specifically, the condition states:

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the SAR [safety analysis report] are considered appropriate, as discussed in Section 11.0 of the SER [safety evaluation report], and should provide the basis for the user's written operating procedures. The following additional written procedures shall also be developed as part of the user operating procedures:

1. A procedure shall be developed for cask unloading, assuming damaged fuel. If fuel needs to be removed from the multi-assembly sealed basket (MSB), either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This activity can be achieved by the use of the Swagelok valves, which permit a determination of the atmosphere within the MSB before the removal of the structural and shield lids. If the atmosphere within the MSB is helium, then operations should proceed normally, with fuel removal, either via the transfer cask or in the pool. However, if air is present within the MSB, then appropriate filters should be in place to permit the flushing of any potential airborne radioactive particulate from the MSB, via the Swagelok valves. This action will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

In July 1994, the Licensee for Palisades discovered radiographic indications of possible defects in a weld in MSB #4. MSB #4 had been loaded with spent fuel earlier that month and placed inside a ventilated concrete cask on the independent spent fuel storage installation (ISFSI) storage pad. The Licensee evaluated the flaw indications and determined that the MSB continued to meet its design basis and was capable of safely storing spent fuel for the duration of the certificate (20 years). Nevertheless, the Licensee stated that MSB #4 would be unloaded to support additional inspections and evaluations related to its future use.¹ In preparation for the unloading of MSB #4, the Licensee reviewed the unloading procedure issued in May 1993 (Revision 0) and identified several technical deficiencies. A revision of the unloading procedure (Revision 1) was subsequently developed to resolve the identified technical deficiencies. The revised unloading procedure is the subject of an ongoing NRC inspection.²

Through inspections at Palisades and other facilities, the NRC Staff identified a number of concerns regarding licensees' procedures for unloading spent fuel from dry storage casks. The NRC Staff identified examples of procedural

¹ The unloading of MSB #4 was originally planned for several months after the discovery of the radiographic indications of possible weld defects in July 1994. However, the unloading has been delayed several times and in its letter of January 17, 1997, the Licensee informed the NRC Staff that the unloading has been postponed until the fuel in MSB #4 can be reloaded into a certified storage and transportation cask. The Licensee also indicated it intends to pursue development and licensing of such a cask, has solicited and received bids from vendors, and plans to award a contract before the end of the first quarter of 1997.

² In regard to the original (Revision 0) unloading procedure at Palisades, the NRC Staff concluded that, had the Licensee attempted to unload a cask using the original unloading procedure, the Licensee would have needed to suspend activities at one or more times during the unloading process in order to implement revisions to the procedure. The NRC Staff found that this was a violation of requirements that all activities affecting quality be prescribed by procedures appropriate for the circumstances and that procedures are reviewed for adequacy. However, given the limited safety significance of the procedural deficiencies and the fact that the Licensee identified and corrected the deficiencies, the NRC dispositioned the violation as a Non-Cited Violation in accordance with the NRC Enforcement Policy. (See NRC Inspection Report 50-255/96014 and Director's Decision DD-97-1, 45 NRC 33 (1997).)

inadequacies and quality assurance shortcomings experienced during preoperational tests and actual cask loading operations at several facilities. In addition, the Staff observed that some unloading procedures implemented by licensees neglected to consider contingencies and assumptions on possible fuel degradation, gas sampling techniques, cask design issues, radiation protection requirements, and the thermal-hydraulic behavior of a cask during the process of cooling and filling it with water from the spent fuel pool. To address these concerns, the following item titled "Cask Loading and Unloading," was included in the NRC dry cask storage action plan implemented in July 1995.³

Issue: Cask Loading and Unloading

As licensees have implemented their ISFSI plans, several issues have been identified related to the loading and unloading of casks. Loading issues have centered on procedural inadequacies and quality assurance shortcomings. The unloading procedures developed by licensees tend to be simplistic. This has resulted in neglecting to consider contingencies and assumptions on failed fuel, air sampling techniques, disassembly requirements, design problems, and radiation protection requirements. The importance of these procedures should be emphasized to licensees, and technical issues related to unloading problems resolved. This issue should also be addressed for shipping casks.

The NRC action plan developed for dry cask storage was formulated to manage the resolution of a variety of technical and process issues associated with the expanding use of that technology for the storage of spent nuclear fuel. The item related to the loading and unloading of dry storage casks was added to the action plan, in part, to ensure that the importance of the unloading procedures was emphasized to licensees and technical issues related to unloading problems were resolved.

To implement the plan, the NRC Staff formed a working group to identify issues associated with loading and unloading processes for dry storage casks and to propose means of informing the industry and the NRC Staff of those issues. The working group considered industry experiences, concerns identified during reviews and inspections, and other issues related to loading and unloading procedures. The working group completed its reviews in April 1996. The concerns related to unloading procedures reviewed by the working group were found to involve either (1) isolated occurrences that had been adequately resolved by site-specific corrective actions or (2) generic issues that were addressed by incorporating remedial measures into ongoing Staff activities, such as the preparation of revised inspection procedures or other guidance documents.

³ Action plans are used by the NRC Staff to manage the resolution of significant generic issues. Such plans are prepared when the anticipated resources that will be required to resolve generic or potentially generic issues exceed certain thresholds or when the NRC Staff determines that an action plan would improve its efficiency and effectiveness.

In May 1996, an event occurred at the Point Beach plant involving the ignition of hydrogen gas during the loading of a VSC-24 cask.⁴ Completion of the NRC inspection of the revised unloading procedure for Palisades was postponed following the event at Point Beach in order to allow licensees and the NRC Staff to identify the cause of the hydrogen ignition and implement appropriate corrective actions. 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," and subsequent confirmation of corrective actions by the NRC.

On December 3, 1996, the NRC Staff informed the Licensee for Arkansas Nuclear One that it had completed its reviews and inspections associated with that facility and found that the Licensee had satisfactorily completed the commitments documented in the CAL. Shortly thereafter, the Licensee initiated cask-loading activities. The review of responses to the bulletin related to Palisades and Point Beach is ongoing and cask operations at those facilities continue to be limited by the Licensees' commitments described in CALs.

III. DISCUSSION

In support of the Petitioner's request that VSC-24 casks not be loaded until MSB #4 at Palisades has been unloaded and the unloading process has been evaluated, the Petitioner cites the action plan prepared by the NRC Staff that included the Staff's observation that some unloading procedures developed by licensees tended to be simplistic. The Petitioner asserts that because problems are discovered through experience, the proper way to unload casks will not be known until a cask is actually unloaded. The Petitioner also claims that the unloading procedures should not be left to the Licensees to develop and implement but should be the subject of detailed NRC evaluations.

The NRC Staff's concerns about the quality of Licensees' unloading procedures led it to include the issue in the dry cask storage action plan. The action plan provided a framework for the identification and resolution of various technical and administrative issues related to the use of dry storage casks. The previously mentioned actions taken by the NRC Staff and Licensees adequately

⁴ On May 28, 1996, a hydrogen gas ignition occurred during the welding of the shield lid on 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.

resolved the identified issues pertaining to cask unloading procedures. In the specific case of the unloading procedure at Palisades, the Licensee's revised procedure addressed many of the generic Staff activities on cask unloading and is currently the subject of a thorough NRC inspection that will be completed in the near future.

To fulfill some of the goals included in the action plan, the NRC Staff has emphasized the importance of unloading procedures and shared observations with licensees using or considering dry cask storage during opportunities such as the Spent Fuel Storage and Transportation Workshop held in May 1996 and meetings with individual licensees. On the basis that these discussions with the industry and other Staff actions had conveyed important operating experiences to NRC licensees, the Staff deferred issuance of an NRC information notice on the subject of loading and unloading of dry storage casks. The Staff revised inspection procedures to specifically instruct NRC inspectors to review unloading procedures developed by licensees and to identify those issues that warrant particular attention. Guidance included in NRC Inspection Procedure 60855, "Operation of an ISFSI," issued February 1, 1996, states:

For unloading activities, attention should be paid to how the licensee has prepared to deal with the potential hazards associated with that task. Some potential issues may include: the radiation exposure associated with drawing and analyzing a sample of the canister's potentially radioactive atmosphere; steam flashing and pressure control as water is added to the hot canister; and filtering or scrubbing the hot steam/gas mixture vented from the canister, as it is filled with water.

Similar guidance was included in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems, Draft Report for Comment," issued in February 1996 and will be included in the final version of the standard review plan that is currently being prepared. The revised guidance documents ensure that recent and future reviews will address the adequacy of unloading procedures developed by licensees.

The NRC Staff also reviewed the inspection history for existing ISFSIs to determine if unloading procedures were reviewed with due consideration given to the potential complications that may arise during the unloading process. The NRC Staff performed audits or inspections of those licensee programs for which the inspection record did not document whether the unloading procedures adequately addressed the major issues included in the action plan. In regard to the users of the VSC-24 cask system, inspections of unloading procedures at Arkansas Nuclear One (NRC Inspection Report 50-313/96-16, 50-368/96-16, 72-13/96-01 and Notice of Violation, dated July 31, 1996) and Point Beach (NRC Inspection Report 50-266/95011, 50-301/95011, dated November 15, 1995) considered the concerns included in the NRC action plan.

As previously mentioned, the revised unloading procedure at Palisades is the subject of an ongoing inspection, completion of which was delayed as a result of the hydrogen ignition event at Point Beach. The NRC inspection of the revised unloading procedure at Palisades is being coordinated with the Staff's review of the Licensee's response to NRC Bulletin 96-04 and is expected to be completed in the near future, notwithstanding the Licensee's decision to postpone unloading MSB #4 pending the availability of a certified storage and transportation cask.⁵ Further, the NRC has committed to state officials and members of the public that the exit meeting for the inspection of the revised unloading procedure at Palisades will be open to the public, the meeting will be noticed sufficiently in advance to allow interested parties to attend, and the NRC Staff will allocate time to discuss issues with the public following the meeting with the Licensee.

The NRC Staff agrees with the Petitioner that learning from experience is an essential part of improving the safety of nuclear power plant activities, including those associated with dry cask storage of spent nuclear fuel. This principle is reflected in the regulatory requirements pertaining to preoperational testing of dry cask storage activities, as well as various provisions of NRC-approved quality assurance programs. The issuance of Bulletin 96-04 and the CALs for licensees using VSC-24 casks is another example of the NRC Staff's efforts to ensure that applicable operating experience is incorporated into procedures at facilities licensed by the NRC. In this case, the licensees using the VSC-24 cask revised procedures to address the technical concerns identified after the event at Point Beach and agreed to defer cask operations pending the NRC's review of responses to the bulletin and confirmation of corrective actions.

As previously mentioned, the Licensee for Arkansas Nuclear One loaded VSC-24 casks following the NRC Staff's determination that the Licensee had satisfactorily completed the commitments documented in the CAL. On the basis of reviews and inspections performed to verify corrective actions associated with the bulletin, in combination with reviews performed for cask certification and previous inspections of preoperational testing and other aspects of the Licensee's dry cask storage program, the NRC Staff determined that the Licensee for Arkansas Nuclear One could perform either cask loading or unloading operations without undue risk to the health and safety of the public or its own personnel. The NRC Staff, through reviews and inspections to verify corrective actions associated with NRC Bulletin 96-04, must have confidence in the procedures implemented by the Licensee for Point Beach before the NRC permits that Licensee to resume loading or unloading of VSC-24 casks. The Staff must also obtain the necessary confidence that the Licensee for Palisades has implemented

⁵ The Licensee for Palisades responded to NRC Bulletin 96-04 by letters dated August 19 and November 12, 1996. The NRC Staff is awaiting the Licensee's response to a request for information that was issued on February 12, 1997.

the corrective actions related to NRC Bulletin 96-04 as well as the issues included in the NRC action plan before permitting the Licensee to resume loading or unloading VSC-24 casks.

Thus, only after resolution of the issues identified in NRC Bulletin 96-04 and other questions that may arise during the inspections of the Licensees' revised procedures at Point Beach and Palisades, will the NRC permit them to unload casks. As part of its review, the NRC Staff will consider matters such as the dry-run exercises licensees performed to verify key aspects of unloading procedures, as well as licensees' actual experience in the loading and unloading of transportation casks, loading of storage casks, handling of spent fuel assemblies under various conditions, and performing relevant maintenance and engineering activities associated with reactor facilities. Given that the NRC Staff will not permit unloading of any casks unless it obtains reasonable assurance of each licensee's ability to do so safely, the NRC does not have reason to require unloading of MSB #4 at Palisades before allowing resumption of normal activities under the general licenses at Arkansas Nuclear One, Point Beach, or Palisades.

The Petitioner's request is, therefore, denied.

IV. CONCLUSION

The Petitioner requested that the NRC prohibit loading of VSC-24 casks at any nuclear site until MSB #4 at the Palisades plant has been unloaded and the experience evaluated for potential safety concerns. Each of the claims by the Petitioner has been reviewed. I conclude that, for the reasons discussed above, no adequate basis exists for granting Petitioner's request for suspension of the licensees' use of the general licenses for dry cask storage of spent nuclear fuel at Palisades, Point Beach, or Arkansas Nuclear One until the MSB at Palisades has been unloaded and the experience evaluated for potential safety improvements.

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 4th day of March 1997.**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of

Docket Nos. 50-321

50-366

50-424

50-425

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

March 18, 1997

The Acting Director, Office of Nuclear Reactor Regulation, has granted in part and denied in part a petition filed by Michael D. Kohn, Esquire, on behalf of Messrs. Marvin B. Hobby and Allen L. Mosbaugh requesting action regarding the Vogtle and Hatch nuclear facilities operated by Georgia Power Company and allegedly by the Southern Nuclear Operating Company (SONOPCO or Southern Nuclear). The petition raised concerns about the management practices of GPC and Southern Nuclear with respect to operation of the facilities, treatment of employees who raise concerns, provision of information to the NRC, and alleged false testimony before the Department of Labor. Petitioners requested the NRC to take immediate steps to determine if GPC's current management has the requisite character, competence, fundamental trustworthiness, and commitment to safety to continue operating a nuclear facility.

Some concerns raised by the petition were partially substantiated. Violations of regulatory requirements occurred. The petition was granted to the extent that: the NRC issued three Notices of Violation and civil penalties to GPC for certain violations, the NRC issued letters to GPC (and GPC and SONOPCO employees) regarding the requirements of 10 C.F.R. §§ 50.7 and 50.9, the license transfer amendment proceeding evaluated many of the concerns, and the license transfer amendments issued for the facilities were conditioned to address concerns about

management. The petition was denied to the extent that the Acting Director determined that no unauthorized transfer of the Vogtle operating licenses has occurred, and concluded that none of the issues call into question the Licensee's character, competence, fundamental trustworthiness, or commitment to safety in the operation of its nuclear facilities. Therefore, further action with respect to the issues raised in the petition was denied.

ATOMIC ENERGY ACT: LICENSING STANDARDS

The general standard for integrity is whether there is reasonable assurance that the licensee has sufficient character to operate the plant in a manner consistent with public health and safety and applicable NRC requirements. The Commission may consider the acts of the licensee (and its employees) that have a rational connection to safe operation of a nuclear power plant.

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

I. INTRODUCTION

This is the final Director's Decision on the petition of Messrs. Marvin B. Hobby and Allen L. Mosbaugh (Petitioners) dated September 11, 1990, as supplemented October 1, 1990, and July 8, 1991, pursuant to 10 C.F.R. § 2.206 (petition). In CLI-93-15, 38 NRC 1 (1993), the Commission vacated and remanded a partial decision on the petition, DD-93-8, 37 NRC 314 (1993), dated April 23, 1993, and directed that the NRC Staff consider the outcome of a pending license transfer proceeding on the Vogtle facility before acting on the petition, due to the overlap in issues. After closure of the evidentiary record and before issuance of a decision, the Licensing Board terminated the Vogtle license transfer proceeding based upon a settlement agreement between Georgia Power Company (GPC or the Licensee) and the sole intervenor, Mr. Mosbaugh. Consistent with the Commission's guidance in CLI-93-15, this Director's Decision addresses the matters considered in the partial Director's Decision and the balance of the petition in light of the information disclosed in the license transfer amendments proceeding, in NRC inspections, investigations, and enforcement actions, and decisions by the Department of Labor.

Although Mr. Mosbaugh has withdrawn his interest in the section 2.206 petition,¹ Mr. Hobby's request is still pending before the NRC. Inasmuch as the petition was jointly filed by Messrs. Mosbaugh and Hobby and it is difficult to segregate their concerns, this Director's Decision addresses all matters raised in the petition, as supplemented by the hearing record.²

II. BACKGROUND

A. NRC Staff and Commission Action on the Petition

On September 11, 1990, Michael D. Kohn, Esquire, on behalf of Messrs. Hobby and Mosbaugh, filed with the U.S. Nuclear Regulatory Commission (NRC) a "Request for Proceedings and Imposition of Civil Penalties for Improperly Transferring Control of Georgia Power Company's Licenses to the SONOPCO Project and for the Unsafe and Improper Operation of Georgia Power Company Licensed Facilities" (petition). The Petitioners were formerly employed by GPC, which operates and is part owner of the Vogtle Electric Generating Plant and the Hatch Nuclear Plant. The petition was referred to the Director, Office of Nuclear Reactor Regulation (NRR), for the preparation of a Director's Decision in accordance with section 2.206. The NRC received exhibits to support the petition on September 21, 1990, and a supplement to the petition on October 1, 1990.

The Petitioners made a number of allegations concerning the management of the GPC nuclear facilities.³ Specifically, the Petitioners alleged that (1) GPC illegally transferred its operating licenses to Southern Nuclear Operating Company (SONOPCO);⁴ (2) GPC knowingly made misrepresentations in its response to concerns of a Commissioner about the chain of command for the Vogtle facility; (3) GPC made intentional false statements to the NRC about the reliability of a diesel generator (DG) whose failure had resulted in a Site Area Emergency at Vogtle; (4) a GPC executive submitted perjured testimony during a U.S. Department of Labor (DOL) proceeding under section 210 of the Energy Reorganization Act; (5) GPC repeatedly abused Technical Specification

¹ By letter dated August 2, 1996, Mr. Mosbaugh withdrew from the 2.206 petition, including "all requests for further proceedings and imposition of penalties relating to Georgia Power Company and Southern Nuclear Operating Company, as well as their directors, officers, employees, and affiliates." See Withdrawal of Allen L. Mosbaugh, dated August 2, 1996.

² Since this Director's Decision primarily addresses events that occurred prior to Mr. Mosbaugh's withdrawal, the term "Petitioners" refers to both he and Mr. Hobby. (However, the term "Intervenor" refers only to Mr. Mosbaugh).

³ Petitioners' concerns about Southern Nuclear and GPC management practices are primarily based on Vogtle-specific information. The Petitioners offered no allegations based on observations of operations at the Hatch facility.

⁴ Before its incorporation on January 1, 1991, Southern Nuclear Operating Company was known as "SONOPCO Project." Afterwards, it was commonly referred to as "Southern Nuclear."

(TS) 3.0.3 at the Vogtle facility; (6) GPC repeatedly and willfully violated TS at the Vogtle facility; (7) GPC repeatedly concealed safeguards problems from the NRC; (8) GPC operated radioactive waste systems and facilities at Vogtle in gross violation of NRC requirements; (9) GPC routinely used nonconservative and questionable management practices at its nuclear facilities; and (10) GPC retaliated against managers who made their regulatory concerns known to GPC or SONOPCO management.⁵ The Petitioners requested that the NRC institute proceedings and take swift and immediate action based on these allegations.

On October 23, 1990, Dr. Thomas E. Murley, who was then Director of NRR, acknowledged receipt of the petition and concluded that no immediate action was necessary regarding these matters. This determination was based on completed and continuing NRC inspections and investigations of the Licensee, particularly those related to the operation of the Vogtle facility.

On February 28, 1991, the NRC requested the Licensee to respond to the petition. The Licensee responded on April 1, 1991 (response).

On July 8, 1991, the Petitioners submitted to the NRC "Amendments to Petitioners Marvin Hobby's and Allen Mosbaugh's September 11, 1990 Petition; and Response to Georgia Power Company's April 1, 1991 Submission by Its Executive Vice President, Mr. R.P. McDonald" (supplement). In the supplement, the Petitioners alleged that GPC's Executive Vice President (1) made material false statements in GPC's April 1, 1991 submittal to the NRC regarding the participants in an April 19, 1990 telephone conference call, and that the submittal attempts to cover up the improper conduct by shifting blame to Petitioner Mosbaugh; and (2) made false statements to the NRC at a transcribed meeting on January 11, 1991, discussing the formation and operation of Southern Nuclear. The supplement also contained a request for a variety of relief, including that the NRC take immediate steps to determine if GPC's current management has the requisite character and competence to continue operating a nuclear facility. On August 26, 1991, Dr. Murley acknowledged receiving the supplement and informed the Petitioners that no immediate action was required and that the specific issues raised in the supplement would be addressed in his Director's Decision. On August 22, 1991, the NRC requested the Licensee to respond to the supplement. The Licensee submitted its response on October 3, 1991 (supplemental response).

On September 18, 1992, GPC filed an application to amend its licenses to transfer to Southern Nuclear its authority to operate the Vogtle units.⁶ In response to notices of the proposed issuance of amendments and opportunity to request a

⁵ Petitioner Mosbaugh had informed NRC's Office of Investigations (OI) of some of these allegations beginning in January 1990.

⁶ By separate application dated September 18, 1992, GPC also requested license amendments to transfer operating authority for the Hatch facility to Southern Nuclear.

hearing that were published in the *Federal Register* (57 Fed. Reg. 47,127, 47,135 (Oct. 14, 1992)), Messrs. Mosbaugh and Hobby filed, on October 22, 1992, a petition for hearing and leave to intervene. In a Memorandum and Order issued November 17, 1992, the Atomic Safety and Licensing Board (Board) denied Mr. Hobby intervenor status for lack of standing. On February 18, 1993 (LBP-93-5, 37 NRC 96, 111 (1993)), the Board granted the intervention petition of Mr. Mosbaugh (Intervenor) and consolidated issues raised in the petition into the following single contention:

The license to operate the Vogtle Electric Generating plant, Units 1 and 2, should not be transferred to Southern Nuclear Operating Company, Inc., because it lacks the requisite character, competence, and integrity, as well as the necessary candor, truthfulness, and willingness to abide by regulatory requirements.

The admitted bases for the character and integrity contention were Intervenor's allegations that (1) GPC knowingly misled the NRC about who controlled licensed activities at the Vogtle facility by omission or misstatements of information (thus concealing a *de facto* transfer of control of the Vogtle facility to SONOPCO Project) and (2) GPC knowingly provided inaccurate, incomplete, or misleading information regarding diesel generator (DG) starts and reliability in 1990 statements, as well as in April 1991 statements regarding the knowledge and involvement of senior GPC officials with respect to the inaccurate 1990 DG information.⁷ LBP-93-5, 37 NRC at 104-11; LBP-94-37, 40 NRC 288 (1994) (partial summary disposition of illegal transfer issue); LBP-93-21, 38 NRC 143, 148 (1993). Some of the issues raised by the petition, as supplemented, were also considered in this proceeding concerning GPC's application to transfer authority to operate the Vogtle facility to Southern Nuclear (license transfer amendments proceeding).

In a partial decision on the petition, dated April 23, 1993, DD-93-8, 37 NRC 314 (1993), *vacated and remanded*, CLI-93-15, 38 NRC 1 (1993), the Director, NRR, addressed each issue raised in the petition except for the allegations of discrimination and perjured testimony that were pending before the Department

⁷With respect to the DG reporting issue, Intervenor alluded to alleged falsehoods in GPC's April 19, 1990 Licensee Event Report 90-006 (LER) to the NRC (that reported a DG start count after the March 20, 1990 Site Area Emergency (SAE)) and a related OI investigation. See Amendments to Petition to Intervene and Request for Hearing, dated December 9, 1992 (Amended Petition) at 15-16, 18-19. Intervenor also asserted that in GPC's April 1, 1991 response to Intervenor's section 2.206 petition, Mr. R. Patrick McDonald, Executive Vice President-Nuclear Operations, knowingly submitted false information (1) concerning the participation of Mr. W. George Hairston, III, Senior Vice President-Nuclear Operations, in developing the April 19, 1990 Licensee Event Report 90-006 (LER); and (2) when GPC managers became aware of errors in the LER. Amended Petition at 16-19. In the amended petition, Intervenor noted that these and other allegations were submitted to OI beginning in June 1990 and were the subject of a section 2.206 petition filed on September 11, 1990, and supplemented September 21 and October 1, 1990, challenging the character, competence, and integrity of GPC and the proposed transferee, Southern Nuclear.

of Labor⁸ and the allegedly false GPC statements to the NRC about the DG starts. The NRC Staff determined that certain concerns raised by the Petitioners were partially substantiated, and Notices of Violation and a civil penalty were issued in response to these issues. The Director declined to take further action with respect to the matters resolved and concluded that (1) there was no unauthorized transfer of the Vogtle operating licenses, (2) GPC facilities “are now being operated in accordance with NRC regulations and do not endanger the health and safety of the public,” and (3) the information available as of that date did not “call into question the Licensee’s character, competence, fundamental trustworthiness, and commitment to safety with respect to operation of its nuclear facilities.” 37 NRC at 345.

On July 14, 1993, the Commission vacated and remanded to the NRC Staff “those portions of the section 2.206 petition decided [in DD-93-8] for the Staff’s further evaluation and final decision in conjunction with the Staff’s resolution of the other remaining matters in the petition and in light of the outcome of the transfer proceeding.” CLI-93-15, 38 NRC at 3. The Commission indicated that its decision was based on the “overlap and similarity of some issues between the section 2.206 petition and the transfer proceeding” which warranted that “the Staff’s final determination of the common issues should take into account the Licensing Board’s findings and the outcome of the transfer proceeding.” The Commission further indicated that the common concern raised by the allegations that GPC or Southern Nuclear officers (and the corporate organization responsible for operation of the Hatch and Vogtle facilities) lack integrity should not be addressed in a piecemeal fashion, but determined in an integrated manner after consideration of the remaining matters in the petition and the outcome of the transfer proceeding. The Commission, however, did not express any view on the soundness of the NRC Staff’s analysis of the issues addressed in DD-93-8 and did not bar the NRC Staff from taking prompt enforcement action at any time during the ongoing review of the matters raised in the petition. *Id.* at 3-4. Inasmuch as the hearing record supplements issues raised in the petition, and consistent with Commission guidance, these matters are addressed as part of this Director’s Decision.

B. DG Enforcement Actions

The NRC Office of Investigations (OI) documented the results of its investigation of the DG issues in a report on OI Case No. 90-020R, dated December 17, 1993 (OI Report). OI found that some GPC officials had either deliberately, or with careless disregard, submitted false or misleading information to the NRC

⁸ *Marvin B. Hobby v. Georgia Power Co.*, DOL Case No. 90 ERA-30; *Allen Mosbaugh v. Georgia Power Co.*, DOL Case Nos. 91 ERA-001 and 91 ERA-011.

during an April 9, 1990 presentation and in a related April 9, 1990 letter; in an April 19, 1990 LER; in a June 29, 1990 cover letter to the revised LER; and in an August 30, 1990 letter regarding DG start-count information.

The NRC Staff evaluated Intervenor's allegations and information in the OI Report and, on May 9, 1994, issued a Notice of Violation and Proposed Imposition of Civil Penalties (NOV) and Demands for Information (DFIs) to GPC and six GPC employees. After considering the GPC reply to the NOV, and the GPC and individual responses to the DFIs, the NRC Staff issued a Modified Notice of Violation and Proposed Imposition of Civil Penalties (Modified NOV) on February 13, 1995.⁹ In the Modified NOV, the NRC Staff concluded, among other things, that subject to commitments made by GPC and Mr. George Bockhold (Vogtle General Manager during 1990), the NRC "has no present concerns with the character and integrity of GPC or the individuals identified in Demands for Information."

C. Licensing Hearing

In January 1995, after completion of the discovery period concerning the illegal transfer issue, evidentiary sessions of the amendment proceeding on the proposed license transfer were held. Intervenor's case consisted of (1) his own prefiled testimony; (2) the testimony of Messrs. Marvin Hobby, William Shipman (who in October 1988 was the Vogtle General Manager for Support and became General Manager in January 1991), Fred D. Williams (GPC Vice President of Bulk Power Markets); (3) excerpts of prior testimony (e.g., DOL proceedings *Hobby v. GPC* and *Yunker and Fuchko v. GPC*), see Transcript (Tr.) 10,134-66, 10,170-99, 2757-58; and (4) deposition excerpts. Evidence received addressed (1) control of daily nuclear operations; (2) the development and implementation of nuclear policy decisions; (3) the employment, supervision, and dismissal of nuclear personnel; and (4) responsibility for nuclear costs. The hearing was to determine whether GPC, either through omissions or misrepresentations, misled the NRC about who was in control of the Vogtle facility. LBP-94-37, *supra*.

⁹The NOV (Staff Exh. II-46) found GPC's failure (on April 9, April 19, June 29, and August 30, 1990) to provide information to the NRC that was complete and accurate in all material respects as required by 10 C.F.R. § 50.9 constituted a Severity Level II problem and proposed a \$200,000 civil penalty. In response, GPC generally admitted each violation except the violation regarding air quality statements in the April 9 letter. See GPC Reply to NOV and DFIs, dated July 31, 1994 (Intervenor Exh. II-105). The Modified NOV (Staff Exh. II-51) withdrew the violation associated with air quality, but maintained that the remaining violations constituted a Severity Level II problem. GPC paid the civil penalty on March 1, 1995. See Letter from Mr. J. Milhoan to Mr. C.K. McCoy, dated March 13, 1995 (Intervenor Exh. II-60), at 1.

Hearings on the DG issues were held from April through September 1995, and generated a transcript record of over 12,500 pages, prefiled testimony of over 35 witnesses, and nearly 600 exhibits.¹⁰

The Board ruled that (1) the allegations in the NOV were important to the admitted contention and were within the scope of the license amendments proceeding, and (2) Intervenor could inquire as to whether GPC withheld pertinent facts from the NRC with respect to the DGs. LBP-94-15, 39 NRC 254, 255-56 (1994). The Board allowed evidence on whether GPC officials were willful or recklessly careless of the facts (as opposed to complete and accurate): (1) in the April 9 letter statement that air quality was satisfactory; (2) in the April 9 letter statement that recently obtained high dewpoint readings resulted from faulty instrumentation; and (3) in other communications with the NRC regarding high dewpoints.¹¹ See Memorandum and Order (Summary Disposition: Air Quality), dated April 27, 1995 (unpublished), at 6-9.

Some of the issues raised in the section 2.206 petition were also heard during the hearing to give Intervenor latitude in establishing that certain communications from GPC to the NRC were false and misleading and, circumstantially, to show a pattern of deception and falsehood associated with the original representations to the NRC. Memorandum and Order (Motion to Strike Mosbaugh Testimony), dated May 11, 1995, at 4-6.¹²

Intervenor's direct case included his written testimony and cross-examination of adverse witnesses (present and former employees of GPC). GPC's case included the testimony of site and corporate management regarding the Vogtle facility, including Messrs. R. Patrick McDonald (GPC Executive Vice President-Nuclear Operations), W. George Hairston, III (GPC Senior Vice President-Nuclear Operations), C. Kenneth McCoy (GPC Vice President-Vogtle Project),

¹⁰Included among these exhibits were the transcripts of audio tape recordings (and two audio tapes) secretly made by Mr. Mosbaugh in February through August 1990 at the Vogtle site. Mr. Mosbaugh gave OI 277 audio tape recordings in connection with his allegations. OI retained 76 tapes, citing conversations on 22 tapes in the OI Report. The Mosbaugh tape recordings were akin to a contemporaneous record of some events related to matters in the hearing, but some tape excerpts played in the courtroom contained numerous inaudible portions and the content, context, and tone of the remarks recorded were disputed, e.g., Tape 58, dated 4/19/90 (Board Exh. II-12). Unsuccessful or incomplete attempts to arrive at agreements on tape transcripts led to different versions of some tape transcripts being proffered by the parties.

¹¹Mr. Mosbaugh's air quality allegation asserted that Mr. George Bockhold, Vogtle General Manager, deliberately misrepresented DG air quality in the April 9 letter by withholding then recent (known) out-of-tolerance DG control air dewpoint readings, as well as erroneously asserting that high readings were due to faulty instruments and that air quality was satisfactory. OI Report (Intervenor Exh. II-39) at 95. OI substantiated this and the other allegations concerning DG information and concluded that Messrs. George Bockhold, George Hairston, Kenneth McCoy (Vice President-Vogtle Project) and William Shipman (General Manager-Plant Support) deliberately (or with careless disregard) had submitted false and incomplete information to the NRC. OI did not substantiate, however, that Mr. McDonald deliberately provided false information to the NRC in the GPC response to Intervenor's section 2.206 petition. See OI Report at 1-2.

¹²These matters included the FAVA (a radwaste microfiltration system) and "Dilution Valve" allegations provided to OI prior to the March 20, 1990 Site Area Emergency and also raised in the section 2.206 petition. The technical matters raised by the allegations were not admitted into the license transfer proceeding. May 11 Order at 7-8.

Bockhold, John G. Aufdenkampe (GPC Manager of Technical Support), Jimmy Paul Cash (a Unit Superintendent for the Vogtle facility and a degreed Senior Reactor Operator), Georgie R. Frederick (Supervisor–Safety Audit and Engineering Review),¹³ and the testimony of two former NRC employees.¹⁴ The NRC Staff witnesses were Messrs. David B. Matthews (NRR Project Director for the Hatch and Vogtle facilities from 1988 through 1995), Pierce H. Skinner (Region II Section Chief of Reactor Projects since 1991), Darl S. Hood (NRR Licensing Project Manager for the Vogtle facility from August 1990 through 1995), Edward B. Tomlinson (an NRR Senior Reactor Engineer for DGs and supporting systems since 1981), Luis A. Reyes (Region II Director of Division of Reactor Projects from 1987 to 1992, and Deputy Regional Administrator for Region II through 1997), and Roy P. Zimmerman (NRR Associate Director for Projects since June 1994).

After proposed findings of fact and conclusions of law were filed in the proceeding,¹⁵ Mr. Mosbaugh and GPC filed a joint motion requesting that the Board dismiss the proceeding and refrain from issuing an Initial Decision. On August 19, 1996, the Board issued a Memorandum and Order (LBP-96-16, 44 NRC 59) terminating the license amendments proceeding based on Mr. Mosbaugh's withdrawal as the sole intervenor pursuant to a settlement agreement with GPC.¹⁶ In LBP-96-16, the Board recognized that the Commission encourages settlements and stated:

¹³ Among the other witnesses that testified for GPC were Thomas V. Greene, Jr. (Assistant General Manager–Plant Support), Michael W. Horton (Manager–Engineering Support), Harry W. Majors (Licensing Engineer–Vogtle Project), Thomas E. Webb (Licensing Engineer–Vogtle site), Kenny C. Stokes (a Senior System Engineer in the Engineering Support Department with primary responsibility for the DGs), Lewis A. Ward (Manager of Nuclear Maintenance and Support), and W.F. “Skip” Kitchens (Assistant General Manager–Operations and Chairman of the Vogtle Plant Review Board), and Mark Briney (an acting Instrumentation and Control (I&C) superintendent in March–April 1990).

¹⁴ In 1990, Mr. Milton D. Hunt was an NRC Inspector, and Mr. Richard A. Kendall was a member of the NRC Incident Investigation Team (IIT).

¹⁵ Georgia Power Company's Proposed Findings of Fact and Conclusions of Law on Diesel Generator Reporting Issues, dated November 6, 1995; Intervenor's Final Statement of Fact and Conclusions of Law, dated November 30, 1995; NRC Staff Proposed Findings of Fact and Conclusions of Law in the Form of an Initial Decision, dated December 12, 1995; Georgia Power Company's Reply to Intervenor's and the NRC Staff's Proposed Findings of Fact and Conclusions of Law, dated December 22, 1995.

¹⁶ Although the settlement agreement was not made available to the Board or NRC, both Mr. Mosbaugh and GPC assured the Board that nothing in the settlement agreement would prohibit, restrict, or otherwise discourage Mr. Mosbaugh from raising safety concerns to the NRC in the future. Mr. Mosbaugh also stated that all of his safety or regulatory issues had been presented to the NRC. Joint Motion of Termination, dated August 2, 1996, at 10.

Mr. Mosbaugh also withdrew his complaint before DOL. On August 23, 1996, a DOL Administrative Review Board issued a “Final Order Approving Settlement and Dismissing Complaint” after reviewing the confidential settlement agreement regarding the discrimination suit of Mr. Mosbaugh (DOL Case Nos. 91-ERA-1, 91-ERA-11), finding the agreement to be “a fair, adequate and reasonable settlement of the complaints.” On August 29, 1996, the DOL Administrative Law Judge (to whom the suit had been remanded by the Secretary of Labor on November 20, 1995, for a determination regarding Mr. Mosbaugh's damages) took note of the Order by the Administrative Review Board and issued an “Order of Dismissal.”

We are satisfied, based on our analysis of the record, that the Staff has been an active guardian of the public interest at Plant Vogtle and, to the extent that they may have not already done so, that the Staff will take the record we have developed into account in exercising its continuing authority. See Notice of Violation and Proposed Imposition of Civil Penalty (NOV) and Demands for Information (DFI), May 9, 1994; Modified Notice of Violation and Proposed Imposition of Civil Penalties, February 13, 1995; Notice of Violation (Department of Labor Case Nos. 90-ERA-30, and 91-ERA-011), May 29, 1996.

44 NRC at 66.

D. Standards for Character and Integrity

In reaching this decision on the character and integrity contention, I have considered the following Commission guidance and precedent. In *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), CLI-85-9, 21 NRC 1118, 1136-37 (1985) (footnotes omitted), the Commission stated:

A generally applicable standard for integrity is whether there is reasonable assurance that the Licensee has sufficient character to operate the plant in a manner consistent with the public health and safety and applicable NRC requirements. The Commission in making this determination may consider evidence regarding licensee behavior [including the acts of licensee employees since all organizations carry out their activities through individuals] having a rational connection to the safe operation of a nuclear power plant. This does not mean, however, that every act of licensee is relevant. Actions must have some reasonable relationship to licensee's character, i.e., its candor, truthfulness, willingness to abide by regulatory requirements, and acceptance of responsibility to protect public health and safety. In addition, acts bearing on character should not be considered in isolation. The pattern of licensee's behavior, including corrective actions, should be considered.

In *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), CLI-80-32, 12 NRC 281, 291 (1980), the Commission stated that

[e]ither abdication of responsibility or abdication of knowledge, whether at the construction or operating phase, could form an independent and sufficient basis for revoking a license or denying a license application on grounds of lack of competence (i.e., technical) or character qualification on the part of the licensee or license applicant. 42 USC 2232a.

Licensee communications to the NRC, whether written or oral, must be complete and accurate as required by section 50.9. In promulgating section 50.9, the Commission emphasized that forthrightness in communications with the NRC is essential if the NRC is to fulfill its responsibilities to ensure that the use of radioactive material and operation of nuclear facilities are consistent with public health and safety. *Completeness and Accuracy of Information: Final Rule and Statement of Policy*, 52 Fed. Reg. 49,362 (Dec. 31, 1987). A determination of whether information is "complete and accurate in all material

respects” is to be judged by whether information has a natural tendency or capability to influence an agency decisionmaker and omissions are actionable to the same extent as affirmative material false statements. 52 Fed. Reg. 49,363. Thus, a statement is material if a reasonable Staff member should consider the information in question in doing his job, but the NRC need not rely on a false statement for it to be material. *See Randall C. Orem, D.O.*, CLI-93-14, 37 NRC 423, 427-28 (1993) (whether a statement induced the agency to grant an application has no bearing on materiality) and cases cited therein.

The term “material false statement” (which was often used by Intervenor in the license amendments proceeding) is limited “to situations where there is an element of intent,” i.e., egregious situations. 52 Fed. Reg. 49,365. The Commission also explained that intent is also indicated by careless disregard as:

[T]he “concept of ‘careless disregard’ goes beyond simple negligence, as the term has been applied to judicial decisions defining willful conduct as it has been applied by this agency. *See, e.g., Trans World Airlines, Inc. v. Thurston*, 83 L. Ed. 2d 523, 537 (1985); *Reich Geo-Physical, Inc.*, ALJ-85-1, 22 NRC 941, 962-63 (1985). ‘Careless disregard’ connotes reckless regard or callous indifference toward’s one’s responsibilities or the consequences of one’s actions.”

52 Fed. Reg. at 49,365.

In light of the importance of licensee communications and their role in enabling the NRC to discharge its responsibilities, this Director’s Decision examines whether GPC acted with candor and endeavored to ensure that submissions to the NRC were accurate. *See Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC 480, 486, 491 (1976) (“nothing less than simple candor is sufficient”), *aff’d sub nom. Virginia Electric and Power Co. v. NRC*, 571 F.2d 1289 (4th Cir. 1978).

III. DISCUSSION

A. Alleged Unsafe Operating Practices (Petition §§ III.5-8)

The petition included several concerns regarding unsafe operating practices at the Vogtle facility. These concerns were initially addressed in the vacated partial Director’s Decision (DD-93-8) and are presented below with supplementation based on the license amendments hearing record and minor editing.

1. Alleged Routine Entering into “Motherhood”

The Petitioners allege (*see* Petition § III.5) that GPC routinely threatens the safe operation of GPC’s nuclear facilities by allowing them to enter TS 3.0.3,

referred to in the petition as “motherhood.” Specifically, the Petitioners state that (1) GPC repeatedly allowed the Vogtle facility to enter TS 3.0.3 by rendering both trains of safety-related load sequencers for the DGs inoperable, and (2) GPC did not make the required notifications to the NRC when TS 3.0.3 was entered.

Vogtle TS 3.0.3 requires that, when a limiting condition for operation (LCO) is not met, except as provided in the associated action requirements, action shall be taken within 1 hour to place the unit in a mode in which the TS do not apply by placing it in hot standby within the next 6 hours, in hot shutdown within the following 6 hours, and at least in cold shutdown within the subsequent 24 hours.

The NRC established TS 3.0.3 to ensure that the reactor plant is shut down in a timely and orderly manner when the LCO in the TS for the specific component or system is exceeded or when a condition exists that is not addressed by TS requirements. The Licensee has satisfied the TS if it performs the final action within the time specified in the TS. If the condition requiring entry into TS 3.0.3 is corrected before commencing or completing the shutdown, the Licensee need not initiate a shutdown, or if a shutdown is already initiated, may end the shutdown and return the plant to the previous conditions.

In accordance with 10 C.F.R. § 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors, licensees are required to make immediate (i.e., within 1 or 4 hours, depending on the circumstances) reports to the NRC of any declaration of an emergency class specified in the Emergency Plan, and certain non-emergency events. Non-emergency events include such items as the initiation of any nuclear plant shutdown required by the TS, any deviation from the TS authorized by 10 C.F.R. § 50.54(x), any condition where the nuclear power plant (including its principal safety barriers) becomes seriously degraded, and any natural phenomenon or other external condition that poses an actual threat to the safety of the nuclear plant or significantly hampers site personnel in the performance of duties necessary for the safe operation of the plant. In 10 C.F.R. § 50.73, Licensee Event Report System, events are identified for which written reports will be made to the NRC within 30 days. These events include several of the events requiring immediate reports pursuant to section 50.72, plus additional events such as any event or condition that alone could have prevented the fulfillment of the safety function of certain structures or systems. The NRC’s notification and reporting regulations do not contain an explicit requirement that an entry into TS 3.0.3, in and of itself, be reported. Licensees are required by section 50.72 to notify the NRC within 1 hour of the initiation of any plant shutdown required by the plant’s TS. Thus, the NRC is promptly notified of entries into TS 3.0.3 if the plant initiates a shutdown as a result of the problem that caused entry into the TS. However, there is no requirement to notify the NRC of entries into TS 3.0.3 if a shutdown is not initiated. The NRC Staff has

no basis to conclude that the Licensee's activities constituted unsafe practices or that these activities indicated that the character of the Licensee, including those GPC individuals employed by Southern Nuclear in conjunction with the transfer of operating licenses to Southern Nuclear, is unsuitable for operating a nuclear power plant.

The NRC Staff has reviewed GPC's entry into TS 3.0.3 through various inspections conducted by region-based inspectors and through the observations of the permanently assigned resident inspection staff and concludes that GPC does not routinely enter TS 3.0.3.

In Inspection Report 50-424, 425/90-19, January 11, 1991, the NRC Staff documented that GPC management indicated that actions for an orderly shutdown would not be initiated until at least 3 hours after entry into TS 3.0.3. GPC management also indicated that it could perform an orderly, controlled shutdown within 1 hour, if necessary. GPC interpreted the action statement of TS 3.0.3 to allow 7 hours to be in hot standby, and to accomplish this, the shift crew could wait for at least 3 hours after entering the LCO before commencing a shutdown. It was also GPC's position that no notifications to the NRC were required under these circumstances. GPC's actions in this area did not differ significantly from those of other licensees, except that GPC did not immediately notify the load dispatcher¹⁷ and did not provide written guidance to the operations personnel. In Inspection Report 50-424, 425/90-19, the NRC Staff identified the lack of immediate notification as a weakness. On February 28, 1991, GPC responded to this finding by providing written guidance for the operators to use upon entering TS 3.0.3.¹⁸ The NRC Staff reviewed this guidance and, as noted in Inspection Report 50-424, 425/91-14 dated July 19, 1991, found it acceptable.

The specific example identified by the Petitioners regarding this issue concerned GPC's practice in the area of safety-related load sequencers for Vogtle's DGs. The Petitioners claim that the Licensee failed to recognize that the loss of a load sequencer resulted in the entry into TS 3.0.3 and thus required notification of the NRC.

¹⁷ The NRC confirmed that, while GPC did not follow the actions recommended in Generic Letter 87-09 (i.e., notification of the load dispatcher within the first hour and performance of a controlled shutdown throughout the next 6 hours), the NRC could find no instance of GPC ever exceeding the 7-hour time limit to be in hot standby.

¹⁸ The Licensee's written guidance for TS 3.0.3 entry was issued as TS Clarifications, which are additional pages that the Licensee maintains with the TS in the main control room. The guidance provided that upon entry in TS 3.0.3, the Unit Shift Supervisor should evaluate plant conditions and formulate a course of action, including actions to prepare for and complete a safe and controlled shutdown. In cases where a high degree of confidence exists that the technical issues can be resolved or repairs made promptly to restore component operability, an immediate power reduction is not advisable. However, actions are to be taken to ensure that an orderly shutdown will be completed within the allowable time while repairs or attempts to resolve operability are under way. Within the first hour, notifications to the load dispatcher and management should be made. If the condition still exists, power reduction should begin no later than 4 hours into the action (i.e., 3 hours of the allowable time remaining). In those cases where it is apparent that resolution of the condition will not occur within the allowable time, an orderly shutdown will begin immediately.

Each unit at Vogtle has two Engineering Safety Feature Actuation Systems (ESFASs) sequencers and both must be operable during Modes 1, 2, 3, and 4. NRC and GPC personnel determined that removing the load sequencers from service could result in entering the LCO for TS 3.0.3 or in entering TS Table 3.3-2, depending on which portion of the sequencer system was removed. Some of the circuits were included in Table 3.3-2, but the TS did not address the remainder of the system. The Operations Department had historically linked load sequencer outages to the emergency DG LCO of TS 3.8.1.1.b (78 hours to hot standby). During the NRC's special team inspection documented in Inspection Report 50-424, 425/90-19, GPC determined that TS Table 3.3-2 and TS 3.0.3 should have applied to sequencer outages. When this determination was made, GPC informed the NRC Staff that it had not reviewed past work orders for load sequencers.

At that time, the NRC Staff reviewed both the completed maintenance work orders that were performed on the sequencers on Units 1 and 2 and the related surveillance tests by the Instrumentation and Control Engineering and the Operations Departments. The NRC Staff found several instances where the work performed would have required the load sequencers to be de-energized. However, the associated unit was found not to have been in Modes 1, 2, 3, or 4 at the time this work was performed and therefore, no TS LCO applied.

Similar to the maintenance work order review, the NRC Staff reviewed related Instrumentation and Control Engineering and the Operations Departments' surveillance tests. This review did not reveal any examples of the load sequencers having been de-energized while in Modes 1 through 4 at the time the work was performed and thus no TS LCOs applied.

Accordingly, I conclude that GPC does not routinely threaten the safe operation of the Vogtle facility by allowing entry into TS 3.0.3. The Petitioners' claim that NRC notification requirements were violated upon entry into TS 3.0.3 was not substantiated.

2. Alleged Ignoring of Technical Specifications

The Petitioners claim (*see* Petition § III.6) that GPC routinely endangers the public's safety by ignoring TSs and that this is illustrated by seven cited examples.

Example (1): Opening Dilution Valves When Required to Be Locked Closed (Petition § III.6a)

The Petitioners state that the Licensee willfully and knowingly violated Vogtle Unit 1 TSs by opening dilution valves required to be locked closed by TSs. The

Petitioners claim that the valves were opened while the reactor coolant system (RCS) was at mid-loop, and that this placed the plant in an unanalyzed condition and created the risk of an uncontrolled boron dilution accident and an inadvertent reactor criticality. The Petitioners allege that the valves were opened to expedite an outage so that the plant could be placed back on line according to the outage schedule.

OI investigated this event, which occurred in October 1988 during the first refueling outage for Vogtle Unit 1. The results of that investigation are documented in OI Report 2-90-001. The OI investigators concluded that TS 3.4.1.4.2 was knowingly and intentionally violated by Vogtle Operations shift supervisors, with the express knowledge and concurrence of the Operations Manager.¹⁹ In its Report, OI also concluded that a violation of the reporting requirements of section 50.73 occurred, but that the evidence was insufficient to conclude that this was a deliberate violation of reporting requirements.

On June 3, 1991, after reviewing the OI findings, the NRC Staff issued a Notice of Enforcement Conference and Demands for Information to GPC and the Operations Manager at the time of the incident. The NRC Staff also issued Demands for Information to the Operations Superintendent and the Shift Supervisor at the time of the incident.

After reviewing the responses to the four Demands for Information (Demands), the NRC Staff held an Enforcement Conference on September 19, 1991, with GPC and the Operations Manager. Subsequently, the NRC Staff sent letters to the Operations Manager, the Operations Superintendent, and the Shift Supervisor stating that no additional actions would be taken regarding their individual NRC licenses. The NRC Staff also stated that, although the actions of these individuals did not meet NRC expectations, there was insufficient evidence to support a conclusion that their actions in 1988 constituted a deliberate attempt to disregard and intentionally circumvent the requirements of the TSs.

On December 31, 1991, after consultation with the Commission, the NRC Staff issued a Notice of Violation and Proposed Imposition of Civil Penalty of \$100,000 (Notice) to GPC. The Notice set out several violations identified during the NRC investigation conducted between February 1, 1990, and March 19, 1991, including a violation that, contrary to the requirements of TS 3.4.1.4.2, on October 12 and 13, 1988, with Unit 1 in Mode 5, loops not filled, reactor makeup water storage tank valves 1208-U4-176 and 1208-U4-177 were opened in order to add chemicals to the RCS. On January 30, 1992, the Licensee responded to the Notice, denied the violations, and protested the proposed imposition of the

¹⁹Mr. William F. "Skip" Kitchens, the Operations Manager and a PRB chairman, and Mr. Jimmy P. Cash, a Senior Reactor Operator serving as the Operations Superintendent on Shift, are also mentioned in this Director's Decision in the discussions of the DG issue. *See also* Section III.D herein regarding a January 1990 meeting between Messrs. Bockhold, Kitchens, and Mosbaugh.

civil penalty.²⁰ The NRC Staff reviewed GPC's response and, on June 12, 1992, issued an Order Imposing Civil Monetary Penalty of \$100,000 (Order). On July 9, 1992, GPC responded to the Order, submitted payment of the penalty, and noted that it did not plan to appeal this action.

The NRC Staff has also evaluated the Petitioners' concern that the plant was placed in a condition that could have resulted in an uncontrolled dilution event and inadvertent reactor criticality. The NRC Staff reviewed an analysis of this event that Westinghouse subsequently performed and GPC provided on November 21, 1989, to support proposed license amendments to change Vogtle TS 3.4.1.4.2. The change would allow the valves to be opened under administrative control to enable nonborated chemical additions to be made to the RCS during Mode 5b (cold shutdown with coolant inventory reduced to the extent that the reactor coolant loops are not filled) and Mode 6 (refueling), using a flow path via the reactor makeup water storage tank. The results of the Westinghouse analysis indicated that the minimum acceptable operator action times of 15 minutes for Mode 5b and 30 minutes for Mode 6, as specified in the NRC's Standard Review Plan (NUREG-0800), would be met. On the basis of this analysis, the NRC Staff concluded that the opening of these valves under administrative controls with the RCS in a loops-not-filled condition, including the mid-loop condition, would not result in an unsafe condition. This conclusion formed the basis for the NRC Staff's approval of License Amendment No. 28 for Vogtle Unit 1 and License Amendment No. 9 for Vogtle Unit 2, each dated February 20, 1990. The responses by GPC and specific individuals indicated that precautions were taken when the valves were opened in 1988 to ensure that the valves would remain open for no more than 5 minutes. While the NRC Staff is unable to conclude that these undocumented controls were in place, the NRC Staff does find that the actual amount of time the valves were open was of insufficient duration to create a criticality event. Therefore, the NRC Staff concludes that, although the TSs in effect at the time were violated, the actual opening of the valves in 1988 did not endanger the health and safety of the public.

Thus, to the extent that Petitioners allege that a violation associated with the operation of these dilution valves occurred, the allegation is substantiated and the NRC has taken appropriate enforcement action. However, the evidence does not substantiate that this action was willful. Rather, as indicated by the responses of the Operations Manager, the Operations Superintendent, the Shift Supervisor, and GPC to the NRC's Demands for Information and during the

²⁰ It was GPC's position that the Action Statement in the TS stating that the valve should be closed immediately if found open meant that the valve could be opened for about 5 minutes. GPC based this position upon earlier correspondence between NRC and the nuclear industry which had explored potential definitions for "immediate" actions.

Enforcement Conference, the action resulted from an incorrect interpretation of the TS requirement by the Operations Manager in 1988.

Example (2): Failure to Secure Dilution Valves as Required by TSs (Petition § III.6b)

On February 26, 1990, the NRC Staff found that the dilution valves, identified in Example 1 above, were required to be locked closed, but were not locked while at mid-loop, in violation of TSs. The Petitioners assert that this is another example of a willful violation of TSs by Vogtle senior management.

On February 26, 1990, while Unit 1 was in Mode 5 with reactor coolant loops not filled (mid-loop), the NRC Staff found that discharge valve 1-1208-U4-176 of the refueling makeup water storage tank was closed but was not secured in position as required by Action Statement (c) of TS 3.4.1.4.2. Instead of installing a mechanism to mechanically secure this valve, the Licensee placed a "hold tag"²¹ on the valve, which provided only administrative control to preclude valve operation. When the NRC Staff described this condition to the Licensee, Vogtle personnel contended that the administrative controls were acceptable to fulfill the requirements of the TS that the valve be secured in position. GPC later agreed that this method was an unacceptable interpretation of the TS and took action to install a mechanical locking device. On April 26, 1990, the NRC Staff issued Notice of Violation, 50-424, 425/90-05-01, "Failure to Mechanically Secure Valve 1-1208-U4-176 During Mode 5 as Required by TS 3.4.1.4.2.C."

During a subsequent NRC inspection (Inspection Report 50-424, 425/91-14), the NRC Staff reviewed the Licensee's associated actions in connection with this issue and closed this violation. The inspectors reviewed the locked-valve procedure, 10019-C, which had been revised to eliminate using a hold tag on valves that are required by TSs to be secured in position. To secure the valve, the Licensee routed a steel cable through drilled holes in the valve handle and mechanically secured the cable to prevent personnel from operating the valve. GPC conducted a comprehensive review of all remaining valves required by TSs to be secured to ensure that each had a locking mechanism in place. GPC committed to providing an appropriate locking mechanism for any valve secured by a hold tag and required to be secured by TSs. However, GPC found no other valves in that category.

The NRC Staff concludes that, although a violation was issued, it resulted from the Licensee's erroneous belief that use of a hold tag was an acceptable

²¹ A "hold tag" is a 3-inch by 5-inch red tag that is attached to a piece of equipment to indicate that it is not to be operated. The intent of the "hold tag" is indicated by Vogtle's Administrative Procedure 304-C, "Equipment Clearance and Tagging Procedure," which states that "A hold tag, when attached to a piece of equipment, prohibits the operation of that equipment in all circumstances."

means of satisfying the TS requirement that the valve be secured. No evidence was found of a willful violation of TSs by Vogtle senior management or other personnel. Therefore, the allegation was not substantiated.

Example (3): Miscalculation of Shutdown Margin (Petition § III.6c)

The Petitioners allege that in January 1989, two shifts of licensed operators miscalculated, because of procedural errors, the shutdown margin for Vogtle Unit 1, which was shut down at the time, and consequently that the RCS boron concentration became "dangerously low" and that the Licensee did not write a deficiency report, conduct a critique, review its actions for conformance to TSs, or submit a report to the NRC.

Vogtle TS 3.1.1.2 requires that a specified minimum shutdown margin be maintained when the reactor is in Mode 3 (Hot Standby), 4 (Hot Shutdown), or 5 (Cold Shutdown). The required minimum value is specified by graphs of shutdown margin as a function of RCS boron concentration. The minimum shutdown margin specified in TS 3.1.1.2 is sufficient to ensure, as a most restrictive condition, that if a boron dilution accident were to occur during the beginning of core life, the operator would have at least 15 minutes to take corrective action after the initiation of an alarm caused by source range high flux to avoid total loss of shutdown margin. An operator reaction time of at least 15 minutes is consistent with the associated accident analyses of the boron dilution event in the Final Safety Analysis Report (FSAR). The corresponding surveillance requirement in TS 4.1.1.2 requires that the shutdown margin be determined to be greater than or equal to the required value at least once every 24 hours by considering several factors, including RCS boron concentration, RCS average temperature, and xenon concentration.

At 5:35 p.m. on January 19, 1989, control room operators at Vogtle manually tripped the Unit 1 turbine and reactor to enter a planned outage to repair a leaking socket weld for the drain line in the loop seal downstream of the pressurizer safety relief valve. After the unit was shut down, an extra shift supervisor on shift completed Procedure 14005-1, "Shutdown Margin Calculation," which must be completed every 24 hours when the plant is in Mode 3, 4, or 5. He signed the procedure at 7:13 p.m. on January 19, 1989. However, the extra shift supervisor incorrectly completed Data Sheet 2, which applies to conditions where the average RCS temperature is equal to or greater than 557 degrees Fahrenheit (°F). This action was incorrect because he should have completed Data Sheet 4, which applies to conditions related to entering Cold Shutdown (Mode 5). The shutdown margin calculation that was completed by the shift supervisor was based upon the wrong data sheet, and resulted in a calculated

shutdown margin of 6.6% reactivity (i.e., delta k/k)²² and a required shutdown margin of 2.58% delta k/k . These results indicated to the operators that no boron addition to the RCS was required in order to enter Cold Shutdown.

On January 20, 1989, at approximately 9:00 a.m., a reactor engineer questioned the apparently low RCS boron concentration of 1333 parts per million (ppm). His concern prompted the Licensee to stop the unit cooldown until the shutdown margin calculation was verified. At 10:22 a.m., the reactor engineer completed a shutdown margin calculation that assumed an RCS temperature of 68°F and 0% reactivity for xenon worth. His calculation, which did not take into account xenon worth, showed that 1800-ppm boron concentration was necessary to obtain a shutdown margin of 4.015% delta k/k compared to a required shutdown margin of 3.47% delta k/k . This calculation failed to include credit for xenon worth, which would have added approximately 3.8% delta k/k to the shutdown margin and provided more than an adequate margin above TS requirements without further boration. Since no TS limit was exceeded, GPC was not required to submit, and did not submit, a written report to the NRC.

On January 20, 1989, at 1:38 p.m., the on-shift operations supervisor recalculated the shutdown margin that had been incorrectly calculated at 7:13 p.m. on January 19, 1989. The new calculation relied upon plant data in effect on January 19 and was based upon Data Sheet 4. The new calculation determined that the shutdown margin was 4.185% delta k/k while the required shutdown margin was 1.92% delta k/k .

The NRC Resident Inspectors reviewed Procedure 14005-1, Data Sheets 2 and 4, the calculations concerning the data sheets dated January 19 and 20, 1989, and control room logs for that period. The NRC Staff discussed the inspection findings in Inspection Report 50-424, 425/91-20, dated September 12, 1991. The NRC Staff found that the shutdown margin calculation performed at 7:13 p.m. on January 19, 1989, was incorrect in that the wrong Data Sheet of Procedure 14005-1 was used. However, the inspector found no evidence that the TS limits on shutdown margin were ever exceeded or that an inadvertent criticality could have occurred because the wrong data sheet was used. The confusing instructions on Data Sheet 2 of Procedure 14005-1 contributed to this error. On March 26, 1989, the Licensee revised this procedure to simplify, consolidate, and clarify the data sheets. The NRC Staff also confirmed that GPC failed to write a Deficiency Card for this event which would have prompted the Licensee to perform a followup review of the error. The inspectors reviewed

²² Reactivity is defined as the fractional change in neutron population from one neutron generation to the subsequent generation. Reactivity is expressed mathematically as $(K_{\text{effective}} - 1)/K_{\text{effective}}$, or as delta k/k , where $K_{\text{effective}}$ is the multiplication factor in a nuclear system expressing the change in the fission neutron population per generation.

GPC's Deficiency Card program and found it to be adequate. They could find no other instances of a failure to write a Deficiency Card.

Thus, the NRC Resident Inspectors determined that violations occurred. The extra shift supervisor failed to follow procedures in selecting the data sheet. Additionally, a shift supervisor made an error and failed to write a Deficiency Card.

Based on its review of Inspection Report 50-424, 425/91-20, the NRC Staff has determined that these violations meet the criteria contained in sections V.A and V.G.1 of the then-in-effect General Statement of Policy and Procedure for NRC Enforcement Actions (10 C.F.R. Part 2, Appendix C) for violations for which a Notice of Violation need not be issued. Section V.A allowed the NRC to exercise discretion in issuing a Notice of Violation for isolated Severity Level V violations, regardless of who identified them, provided the Licensee had initiated appropriate corrective actions before the end of the inspection. Under section V.G.1, the NRC need not issue a Notice of Violation if the violation was identified by the Licensee, is normally classified at a Severity Level IV or V, was reported if required, was or will be corrected (including measures to prevent recurrence) within a reasonable time, was not a willful violation, and was not a violation that could reasonably be expected to have been prevented by the Licensee's corrective action for a previous violation. This practice of not requiring the issuance of a Notice of Violation when a violation meets the aforementioned criteria was adopted by the NRC as a means of encouraging licensees to identify and correct violations and to avoid expenditure of limited resources for both the NRC and the licensee — resources that could be better used in improving safety.

In summary, the Licensee identified and corrected the shutdown margin calculation error, which did not result in the violation of a TS limit and did not require a written report to the NRC. Moreover, the corrected calculations of the shutdown margin do not support the allegation that the error resulted in "dangerously low" boron concentrations in the RCS or that it endangered the health and safety of the public. The NRC inspectors determined that, even though a Deficiency Card was not written, the Licensee's followup review of the error was prompt and had been completed before the end of the inspection.

Example (4): "Taking" LERs (Petition § III.6d)

The Petitioners claim that GPC employees were told, on March 22, 1990, to keep planned shutdowns on schedule by "taking" LERs.²³ The Petitioners

²³"Taking" LERs implies that personnel intentionally do not perform actions required by a TS at the specified time required by the TS action. At a later time, they subsequently acknowledge this action was not performed and

(Continued)

also contend that pressure to remain on schedule would necessarily result in an intentional violation of TS and "taking" LERs in order to remain on schedule.

This issue was reviewed as part of OI's investigation of an alleged intentional TS violation with regard to a mode change with an inoperable neutron source range monitor (*see* Example 6 hereinafter). OI's review and findings in this area are documented in OI Report 2-90-012. The OI investigation did not substantiate the alleged "taking" of LERs. The personnel interviewed stated that they had never been instructed to do "whatever it takes" to stay on schedule.

On the basis of this investigation, the NRC Staff was unable to conclude that Vogtle personnel either had a deliberate practice of, or were instructed to, "take" LERs to stay on schedule. Similarly, statements made by the Petitioners that SONOPCO's philosophy would necessarily result in managers intentionally violating TS and "taking" LERs to remain on schedule were not substantiated by the NRC Staff's review. Therefore, the allegation was not substantiated.

Example (5): Surveillance Testing of Containment Isolation Valves (Petition § III.6.e.i)

The Petitioners claim that the Licensee knowingly concealed a violation which, if uncovered, would have resulted in a safety-related shutdown of Vogtle Unit 1. The violation allegedly concerned the failure to properly test approximately thirty-nine containment isolation valves in violation of TS surveillance requirement 4.6.1.1.a.

In February 1990, after operations personnel performed a monthly TS surveillance on containment isolation valves and turned in their paperwork, the Shift Supervisor recognized an error in that only two of thirty-nine valves had been checked. The Shift Supervisor directed that all necessary surveillances be performed immediately. The Shift Supervisor then examined previous records and found that the same error had also been made the previous month, and therefore, that another violation of TS 4.6.1.1.a had occurred. The Shift Supervisor then informed the Work Planning Group of the error and this group prepared and delivered a Deficiency Card to the control room. Since the missed surveillances had already been completed by this time, no action was initiated under the TS's LCO (shutdown within 1 hour). The Petitioners state that the Deficiency Card should have been initiated earlier by the individual discovering the deficiency and that the event was mishandled to conceal the discovery time and to avoid the shutdown requirement of the LCO.

then write a report (LER) to the NRC as specified in section 50.73. Thus, this "taking" LERs would allegedly be done in order to forgo performing the activity required by a TS at a time that would cause a schedule delay.

GPC reported this issue in a timely LER 50-425/90-01, dated March 27, 1990. NRC resident inspectors reviewed the LER, as documented in Inspection Report 50-424, 425/90-10, and found that the task sheet contained in the procedure for performing this surveillance was inadequate. The format of the task sheet resulted in cognitive personnel errors because the task sheet was unclear as to the number of valves required to be tested. The NRC Staff categorized this event as a noncited violation because the criteria for exercising discretion specified in section V.G.1 of the then-in-effect Enforcement Policy (10 C.F.R. Part 2, Appendix C) were met (NCV 50-425/90-10-01).

An OI investigation did not substantiate that this violation was willful. OI concluded in OI Report 2-90-012, that the missed surveillance had been reported in an LER and resulted from an inadequate Surveillance Task Sheet that had listed equipment identification numbers of only two valves for the monthly containment integrity check. OI noted that the NRC resident inspectors had reviewed the LER and documented the event without issuing a Notice of Violation. OI also noted that the circumstances of this event were reviewed during the NRC's special team inspection at Vogtle in August 1990, which found that the Shift Supervisor did not conceal the true discovery time of the missed surveillance in order to avoid a unit shutdown and that the Shift Supervisor's actions to initiate an investigation into the adequacy of the previous month's surveillance and to concurrently perform the missed surveillances were appropriate. Since the surveillance test is of short duration, it was completed before the determination was made that the previous test had not been completed correctly. Since the surveillance test had already been repeated once the inadequacy of the previous test became known, a shutdown of the unit at that point was not required.

On the basis of the NRC Staff's inspections and the OI investigation, the Petitioners' claim that the Licensee knowingly concealed a technical violation was not substantiated.

Example (6): Changing Modes with Required Equipment Inoperable (Petition § III.6.e.ii)

The Petitioners claim that the Licensee knowingly concealed another violation on March 1, 1990, when a change from Mode 5 to Mode 6 occurred even though required equipment was not operable. Petitioners assert that the failure to comply with the TS translated into a 12-hour schedule enhancement at a critical juncture and was a willful violation.

The NRC resident inspectors, an NRC special inspection team, and OI investigators reviewed this issue. Results of these efforts are documented in NRC Inspection Report 50-424/90-10 dated June 14, 1990, and OI Report 2-90-012. GPC also documented this event in LER 424/90-004 dated May 11, 1990.

This LER described the Licensee-identified violation of TS 3.0.4 on March 1, 1990, when Unit 1 entered Mode 6 from Mode 5 with an LCO in effect for a neutron source range channel. The LER attributed the root cause to cognitive personnel error by the Shift Superintendent who failed to review the back side of the relevant LCO Status Sheet that noted the mode change was prohibited while the source range monitor was inoperable. Moreover, the Shift Superintendent had not otherwise recognized the prohibition before authorizing the mode entry.

The NRC Staff interviewed various personnel involved in the review of plant conditions and involved with documentation necessary to change modes. The interviews indicated that the Shift Superintendent and the Unit Shift Supervisor were aware of an active LCO at the time of the mode change, but neither had connected the LCO to a mode restriction. Both of these individuals indicated that there had been no unreasonable emphasis on the critical path schedule. Both denied that they had ever been given any indication or instruction to do whatever it takes to stay on schedule. They also indicated that they did not feel undue pressure to stay on schedule or any pressure to compromise plant safety even though the mode change resulted in a reduction of the critical path outage time.

The NRC Staff expressed concern that the format of the LCO status sheet contributed to the problem. Because the status sheet is a two-sided form with the remarks section on the back of the form, a cursory review of the sheet could result in any remarks entered on the back of the form being overlooked. On the basis of the NRC resident inspectors' review, the NRC determined that a violation occurred as discussed in Inspection Report 50-424/90-10. This violation was categorized as a noncited violation because the criteria for exercising discretion specified in section V.G.1 of the then-effective Enforcement Policy (10 C.F.R. Part 2, Appendix C) were met (NCV 50-424/90-10-03).

On the basis of evidence developed during the NRC inspections and OI investigation, the allegation of an intentional violation was not substantiated.

Example (7): Failure to Declare RHR Pump Inoperable and Enter LCO (Petition § III.6.e.iii)

The Petitioners allege that GPC knowingly concealed a TS violation when the "B" residual heat removal (RHR) pump was not declared inoperable after cracking of the nuclear service cooling water (NSCW) line. Specifically, the Petitioners allege that, during the second refueling outage at Unit 1 (1R2), with RHR train "A" out of service for maintenance, the RHR train "B" pump experienced excessive vibration and the NSCW motor cooler experienced a leak at its outlet. TS 3.9.8.1, "RHR and Coolant Circulation," was allegedly

violated because the Operations Department chose not to declare RHR pump "1B" inoperable in an effort to mitigate the effect on the critical work path.

The NRC Staff addressed this item in the Special Team Inspection documented in Supplement 1 to NRC Inspection Report 50-424, 425/90-19, dated November 1, 1991. In section 2.2 of the Inspection Report, the NRC Staff concluded that the Vogtle Operations Department had an adequate engineering basis for accepting operability of the RHR pump even with the pump's high vibration and the NSCW leak.

The inspection team also concluded that declaring the pump inoperable would not have affected the critical work path. The LCO actions would not have been restricted because the containment, except for ventilation, had been isolated as required by TS 3.9.4. The LCO actions would not have prevented the Licensee from continuing refueling activities in that the actions to close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere would have required only closing the containment ventilation purge valve, which has an automatic closure signal. Thus, there is no evidence that schedule considerations motivated the Licensee in this matter.

On the basis of evidence developed during NRC inspections, the allegation that GPC knowingly concealed a TS violation when the "B RHR" was not declared inoperable was not substantiated.

3. Alleged Concealment of Safeguards Problems

The Petitioners allege (*see* Petition §§ III.7a and III.7b) that GPC personnel, including a Vice President and General Manager, and a Southern Company Services Manager, knowingly and repeatedly hid safeguards problems from the NRC and willfully refused to comply with mandatory reporting requirements. The Petitioners further allege that the GPC Vice President made false statements to the NRC during an Enforcement Conference about the status of safeguards materials in Birmingham, Alabama, and that the alleged false statements probably influenced a subsequent civil penalty action taken by the NRC. The Petitioners claim that the false and misleading information presented at the Enforcement Conference and other information withheld from the NRC were highly significant. The Petitioners assert that, if the NRC had the benefit of complete, factual information, the NRC would likely have increased the Notice of Violation and Proposed Imposition of Civil Penalty in the amount of \$50,000 issued to the Licensee on June 27, 1990, into the hundreds of thousands of dollars.

The Petitioners also allege that on July 23, 1990, plant and SONOPCO senior management prevented the Site Security Manager from making a Red Phone²⁴ notification within 1 hour as required by section 73.71. The Petitioners allege that the manager was prevented from making the call in order to delay or defuse the NRC's knowledge of programmatic problems on the part of the Licensee regarding the handling of safeguards documents.

OI investigated the allegation that (1) GPC knowingly and repeatedly hid safeguards problems from the NRC and willfully refused to comply with mandatory reporting requirements, and (2) the GPC Vice President made false statements to the NRC in an Enforcement Conference concerning the status of safeguards material in Birmingham, Alabama. The results of these investigations are documented in OI Report 2-91-003. The OI investigations did not substantiate that GPC withheld pertinent information from the NRC at the time of the Enforcement Conference on May 22, 1990, or that GPC management impeded the reporting of safeguards events. On the basis of the OI investigations, the NRC Staff concluded that the Notice of Violation and Proposed Imposition of Civil Penalty of \$50,000 were appropriate.

OI also investigated the allegation that on July 23, 1990, plant and SONOPCO senior management prevented the Site Security Manager from making a Red Phone notification within 1 hour as required by section 73.71. The results of this investigation are also documented in OI Report 2-91-003. Specifically, the concern was that the Site Security Manager was allegedly prevented from making a Red Phone notification for two events. The first event was that a safeguards container had been found open and uncontrolled for half an hour in Birmingham, Alabama, in November 1989. The second event involved fourteen safeguards documents that had been found uncontrolled in the SONOPCO offices on June 15, 1990.

The first event constituted a violation of the reporting requirements of section 73.71, in 1989, when the uncontrolled container was discovered and not reported to the NRC within 1 hour. In 1990, as part of its corrective actions in response to an NRC enforcement action, GPC identified the fact that a required report for this event might not have been made in 1989.

GPC's corrective actions in response to the NRC enforcement action also identified the second event. GPC's consideration of the reporting requirements for the first event was subsequently combined with a similar consideration of the need to report the second event. The second event also was not reported within 1 hour as required by section 73.71.

After reviewing OI's investigation results, the NRC Staff concluded that the failure to make a timely report on the second event and the delay in informing

²⁴ A Red Phone refers to a Licensee's Emergency Notification System and is used for immediate telephone notifications to the NRC's Operations Center in accordance with 10 C.F.R. §§ 50.72 and 73.71.

the NRC Staff of the discovery of the failure to report the first event were due to the GPC's cumbersome system for evaluating corporate security findings through the site security organization, rather than any willful attempt to impede the reporting process.

The NRC Staff decided to take no additional enforcement action for these two issues. The decision to refrain from issuing a Notice of Violation for the delay in reporting the first event was based upon section V.G.5 of the then-in-effect Enforcement Policy (10 C.F.R. Part 2, Appendix C). This provision of the policy allowed the NRC Staff to forego a Notice of Violation when a violation is discovered as the result of corrective action for a previous enforcement action. The NRC Staff considered the violation for the delay in reporting the second event to be an additional example of a violation that the Licensee had identified previously and for which it was, at the time, taking corrective actions. Therefore, as provided by the aforementioned section V.G.5, the NRC Staff issued no Notice of Violation.

Based on the OI investigation and NRC Staff review, the allegation was not substantiated.

4. *Alleged Operation of Radioactive Waste Systems and Intimidation of Plant Review Board Members*

The Petitioners assert (*see* Petition § III.8) that GPC endangered the public's health and safety by operating radioactive waste systems and facilities known to be in gross violation of NRC requirements. The Petitioners also state that Vogtle's General Manager, Mr. George Bockhold, intimidated members of the Plant Review Board (PRB) when they attempted to consider if the use of the waste system should be resumed.

The NRC's Special Inspection Team reviewed this item and discussed its findings in Supplement 1 to Inspection Report 50-424, 425/90-19, dated November 1, 1991. The alleged improper installation and operation of the radioactive waste system is discussed in section 2.1 of the Inspection Report and the alleged intimidation of PRB members is discussed in section 2.7 of the Inspection Report.

The Petitioners allege that GPC installed and operated a radioactive waste microfiltration system without performing an adequate engineering and safety evaluation in accordance with 10 C.F.R. § 50.59.²⁵ This specific system is known as the FAVA system because it is supplied by FAVA Control Systems (FAVA).

²⁵ Title 10 of the *Code of Federal Regulations*, section 50.59, allows licensees to make changes in the facility and procedures, or conduct tests or experiments as described in the safety analysis report, without prior Commission approval, unless the proposed changes involve a change in the Technical Specifications or an unreviewed safety question.

The Petitioners further alleged that the material configuration, fabrication, and quality of the system did not meet the guidance of Regulatory Guide (RG) 1.143, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," and the requirements of the American Society of Mechanical Engineers (ASME) Code.

In late 1987, GPC temporarily installed and operated a system at Vogtle for removing niobium-95. GPC planned to replace this temporary modification with a permanent system in the future.

In February 1988, GPC experienced difficulty in removing colloidal niobium-95 with the temporary system following a reactor shutdown for maintenance work. GPC contracted FAVA to help rectify this problem. The Licensee corrected the situation by installing a 0.35-micron filter system downstream of the existing prefilters. However, a large volume of radioactive waste was generated because the 0.35-micron filters rapidly exhibited high differential pressure and had to be changed frequently. The need to change filters frequently also resulted in Radwaste Department personnel receiving additional radiation exposure.

Upon evaluating the performance of the 0.35-micron filter system, the Radwaste Department determined that the best approach to the problem was to install a backflush, precoat filter system. However, no operational data were available for a system of this type in this specific application. FAVA supplied a proprietary Ultra Filtration System (Model No. 5FD/E) for testing to evaluate whether this was a practical and effective solution to the problem. GPC installed the temporary FAVA system before the Unit 1 refueling outage and operated it under Test Procedure T-OPER-8801. The test system kept liquid effluent releases well below the TS limits. The Radwaste, Chemistry, and Engineering Departments evaluated the test results, and GPC issued a general work order to purchase a permanent system.

In the early part of 1989, the Quality Assurance (QA) Department performed an audit and identified a significant finding involving a programmatic breakdown in the procurement of the temporary FAVA system and a failure to meet commitments of the FSAR. That finding prompted the Licensee to remove the temporary FAVA system from service.

In late 1989, the Licensee sought to reinstall the FAVA system under a temporary modification because colloidal cobalt-59 and cobalt-60 had to be removed. The PRB reviewed this temporary modification and several members expressed strong objections to it based on the previous QA audit finding.

These objections prompted the Licensee to submit a Request for Engineering Assistance (REA) and perform a safety evaluation in accordance with section 50.59 in November 1989. The Licensee's engineering staff subsequently reviewed the November 1989 safety evaluation and found it to be adequate, except that it did not properly address the guidance of RG 1.143 regarding

the use of polyvinyl chloride (PVC) piping. GPC performed another safety evaluation in February 1990 to address this issue and the vulnerability of the PVC pipes to radiation degradation. In the February 1990 safety evaluation, the Licensee specifically stated that the FAVA system did not conform to the criteria of RG 1.143. However, this deviation was found to be technically acceptable for several reasons: (1) The design of the FAVA system had been previously evaluated and found to be adequate in the REA response of November 1989, except for the PVC pipes; (2) the location of the FAVA system was inside a shielded watertight vault, which provided adequate assurance that any system failures would be contained and would not create the potential for offsite releases; and (3) the presence of PVC pipe in the FAVA system, although contrary to RG 1.143, was acceptable based on subsequent design reviews because the radiation exposure of the plastic was found to be within acceptable limits.

Although the testimony of one of the PRB members indicated that the temperature effects on the use of PVC in the FAVA system were not adequately evaluated before the system was installed, the testimony of the corporate system engineer indicated that GPC had considered this before installing the system, although it was not specifically documented in the safety evaluation.

Vogtle management subsequently consulted the NRC resident inspector to seek an NRC position on placing the FAVA system back in service. The inspector was also provided additional information by other Vogtle management personnel documenting reasons why it should not be placed in service. The Licensee forwarded this package to Region II and NRR for review. In March 1990, following Region II and NRR concurrence during a telephone conference, the Licensee placed the FAVA system in service with the following NRC stipulations:

- (1) That procedures for operating the FAVA system require that an operator be present any time the system is in operation;
- (2) That all hoses to and from the FAVA system be verified to conform to RG 1.143;
- (3) That the cover over the FAVA system be securely fastened when the system is in operation to ensure that if a spraying leak developed, it would be contained in the concrete vault; and
- (4) That the design of the walls of the auxiliary radwaste building be evaluated to determine if a design change was needed to reduce the possibility of wall leakage if a hose develops a leak and sprays its contents on the walls.

The Licensee complied with these stipulations upon returning the system to operation.

The review by the NRC indicated that the FAVA system was originally installed and operated by the Licensee without an adequate safety evaluation

and did not meet the guidance in RG 1.143 in that PVC piping was used in this system. However, this deficiency was of limited duration and the Licensee, upon performing subsequent safety evaluations that were forwarded to and accepted by the NRC Staff, concluded that the system was acceptable for use. The NRC's extensive review developed no facts to support a conclusion that the Licensee willfully violated NRC requirements or willfully operated the facility in a manner to endanger the public health or safety.

The Petitioners also contend that Vogtle's General Manager intimidated and pressured PRB members during a PRB meeting. The meeting occurred in February 1990 and was for the purpose of determining the acceptability of the safety analysis for installing the FAVA microfiltration system.

As previously discussed, the Licensee performed several safety evaluations for the temporary modification to install the FAVA microfiltration system. The NRC Special Inspection Team found, through its discussions with PRB members, that, while reviewing these safety evaluations, various PRB members had expressed reservations on several occasions concerning the acceptability of the FAVA system.

Although various PRB members may have expressed reservations, the inspection team, in reviewing the PRB meeting minutes regarding this temporary modification, identified few instances of the PRB members documenting their dissenting opinions. Specifically, the minutes of PRB meeting 90-15, on February 8, 1990, documented one PRB member's negative vote and dissenting opinions regarding the acceptability of exempting the temporary modification from regulatory requirements and the adequacy of the system's safety evaluation. The only other example of a dissenting opinion was in the minutes for PRB Meeting 90-32, on March 6, 1990. This dissenting opinion related to the acceptability of voting on the FAVA system installation when the PRB member who raised the initial questions and concerns on the operation of the FAVA system was not present.

During discussions with NRC inspectors, PRB members indicated that, during the various PRB meetings concerning installing the FAVA system, they felt intimidated and pressured by the presence of the General Manager at the PRB meeting. On one occasion, an alternate voting member felt intimidated and feared retribution or retaliation because the General Manager was present at the meeting and the PRB member knew the General Manager wanted to have the temporary modification approved. However, the PRB member stated that he did not alter his vote and felt comfortable with how he had voted. This PRB member also stated that he was not aware of any occasions where he or any other PRB member succumbed to intimidation or any other occasions where he or they feared retribution.

The PRB members informed the General Manager following the meeting (PRB 90-15) that several of them viewed his presence as intimidating. On March

1, 1990, the General Manager addressed this concern by meeting with all PRB members to reiterate each member's duties and responsibilities. He specifically told the members that his presence at PRB meetings must not influence them and that alternates should be selected who would feel comfortable with this responsibility. He also addressed the difference between professional differences of opinion and safety or quality concerns, and methods for resolving each.

Thus, the NRC Staff has found that, in one case, a PRB voting member felt intimidated and feared retribution because the General Manager was present at the PRB meeting. However, this member stated that he did not change his vote in response to the General Manager's presence.²⁶ He stated that the General Manager was informed of this issue and met with the PRB to allay fears. The information obtained by the NRC Staff indicated that retribution did not occur against any PRB member for revealing a concern about intimidation. The inspection found that the instance involving a member fearing retribution was confirmed, and the absence of dissenting opinions in the PRB meeting minutes called into question the openness of discussions at PRB meetings. Further discussions with PRB members, however, indicated that the lack of dissenting opinions was due to items being discussed and reviewed until all members were comfortable with PRB decisions.

NRC resident inspectors at Vogtle frequently attend PRB meetings and have found that the subjects are candidly discussed and the issues resolved without apparent intimidation.

In summary, the allegation that GPC endangered the public health and safety by operating the FAVA system in gross violation of NRC requirements was not substantiated. The allegation that a PRB member felt intimidated by the General Manager during the meeting on the FAVA system was substantiated, but the reaction did not affect the PRB member's decision regarding safety.²⁷

²⁶ During the license amendments hearing on the DG issue, the Board heard evidence on the FAVA issue in the proceeding to determine whether or not intimidation of PRB members occurred. The PRB member who felt intimidated was not called as a witness and provided no testimony. The interview notes of Mr. Bill Lyon of the Quality Concerns Program for the Vogtle facility on February 23, 1990, confirm that at the time of the PRB's vote on FAVA, the PRB member felt undue pressure to vote early, and probably would have voted "no" had Mr. Bockhold not been present because he thought that FAVA did not meet Regulatory Guide criteria, but that, given his PRB role as a health and safety reviewer, and considering the placement of impingement barriers, there was no health and safety problem. He also stated that he would be willing to meet with the Vogtle General Manager to discuss the matter further. Intervenor Exh. II-231 at 8-9 (marked but not received in evidence).

The notes of the interview are consistent with the NRC inspection finding regarding intimidation.

²⁷ The incident, however, is another example of how the management style of the Vogtle General Manager could result in discouraging individuals from voicing concerns. *See, e.g.*, Section III.C of this Director's Decision regarding the role of the Vogtle General Manager in the inaccurate and incomplete reporting of DG information to the NRC.

B. Alleged Illegal Transfer of Licenses (Petition § III.1 with Supplement Dated October 1, 1990; July 8, 1991 Supplement § IV; License Amendment Proceeding on Illegal Transfer Issue)

The Petitioners allege that GPC improperly transferred control of its licenses to operate the Hatch and Vogtle facilities to SONOPCO. The Petitioners contend that Mr. Joseph M. Farley — who was an officer of GPC's parent company, The Southern Company, and its subsidiary, Southern Company Services — was really the Chief Executive Officer (CEO) of SONOPCO and was, in fact, responsible for operating the GPC nuclear facilities, beginning with the first of three phases in the planned transition to Southern Nuclear. Petitioners contend that Mr. McDonald, GPC Executive Vice President—Nuclear Operations, received management direction from Mr. Farley regarding Vogtle facility matters and that numerous oral and written statements regarding the organization were intentionally false to conceal Mr. Farley's role from the NRC.

The Petitioners contend that during Phase I of the transition to Southern Nuclear, GPC, in effect, transferred control of its NRC licenses to the SONOPCO Project. They base their claim, in part, on Mr. Mosbaugh having witnessed the daily operation of GPC's nuclear facilities at the site and Mr. Hobby at GPC's corporate offices. The Petitioners alleged that (Petition at 6):

The actual chain of command [was Vogtle] General Plant Manager George Bockhold to SONOPCO Vice President McCoy; McCoy to SONOPCO's Senior Vice President, George Hairston; Hairston to SONOPCO's Executive Vice President and Chief Operations Officer, R. Patrick McDonald; McDonald to SONOPCO's Chief Executive Officer, Mr. Farley.

In a supplementary filing of October 1, 1990, the Petitioners further contended that Mr. Farley, "chose the GPC Corporate Officers which would be staffing the SONOPCO Project even though he is not an officer or employee of GPC." In the July 8, 1991 Supplement (at 20), the Petitioners asserted that Mr. McDonald reported to Mr. Farley on administrative matters since the formation of the SONOPCO Project.

The focus of the license amendment proceeding on the illegal transfer issue was whether GPC, either through omissions or misrepresentations, misled the NRC about who was in control of the Vogtle facility, particularly in the context of the extensive communications with the NRC. LBP-94-37, 37 NRC at 291.

A review of the history and background of the formation of Southern Nuclear will assist in understanding this issue.

1. Background: Formation of Southern Nuclear

The Southern Company is the parent firm of five electric utilities: Alabama Power Company (APC), GPC, Gulf Power, Mississippi Power, and Savannah

Electric. Two of these utilities are associated with nuclear facilities at three different sites. GPC is the principal owner and the holder of licenses from the NRC to operate the Vogtle nuclear facility near Augusta, Georgia, and the Hatch nuclear facility near Baxley, Georgia. APC owns the Farley nuclear facility near Dothan, Alabama. The Southern Company also includes Southern Company Services, Inc., a wholly owned service organization.

In 1988, The Southern Company established the SONOPCO Project for the long-term purpose of establishing an operating company to eventually operate the nuclear power generating plants that were then operated by GPC and APC.²⁸ The establishment of a single operating company was to be accomplished in three phases. During Phase I, SONOPCO — which had not yet received the approval of the Securities and Exchange Commission (SEC) — was formed by The Southern Company as a “project” to provide support services to the operating companies (GPC and APC). In Phase II, which is now in effect for the Vogtle and Hatch facilities, SONOPCO (now called Southern Nuclear) continues to provide support services to the operating companies, but has become a legal entity, having obtained the approval of the SEC, and thereafter being incorporated by The Southern Company. Phase III begins for the Vogtle and Hatch facilities (and is currently in effect for the Farley facility) when Southern Nuclear acquires NRC licenses to operate the nuclear facilities.

Because of delays associated with reaching agreement with one of the co-owners, the transition occurred more slowly than first anticipated, and Phase I of the project lasted for approximately 2 years (1989 and 1990). During this phase, Mr. Farley was responsible for the administrative aspects of forming the new operating company. On February 24, 1989, Mr. Farley was elected Executive Vice President–Nuclear of The Southern Company and Executive Vice President of Southern Company Services, Inc. Before these elections, he had been President and Chief Executive Officer (CEO) of APC for almost 20 years.

Until Southern Nuclear acquired the NRC licenses, the GPC nuclear facilities were to remain under the direction of GPC President, Mr. A. William Dahlberg, III, with a reporting chain downward of Executive Vice President–Nuclear Operations (Mr. R.P. McDonald), Senior Vice President–Nuclear Operations

²⁸ In March 1988, GPC and APC met with NRC to discuss their plans to form a separate operating company, SONOPCO. On July 25, 1988, NRC met with GPC to discuss the corporate organization of SONOPCO and GPC, including the generic activities and initiatives involving the Vogtle and Hatch facilities. Enclosure 3 to the meeting summary prepared by NRC Region II, August 11, 1988, a Nuclear Operations–Transition Organization chart, shows the Vice President–Nuclear (Hatch), and the Vice President–Nuclear (Vogtle) reporting to Mr. W.G. Hairston, the Senior Vice President–Nuclear Operations and Mr. W.G. Hairston reporting to Mr. R.P. McDonald, the Executive Vice President–Nuclear Operations. On March 1, 1988, Mr. McDonald was elected a senior officer of GPC and named Executive Vice President–Nuclear, effective April 25, 1988. On May 4, 1988, Mr. W.G. Hairston was elected Senior Vice President–Nuclear Operations of GPC and Mr. C.K. McCoy was elected Vice President–Nuclear of GPC (GPC submittal, April 1, 1991, Attachment 1, Exh. 4).

(Mr. W.G. Hairston, III), and the vice presidents for the Vogtle and Hatch facilities (Messrs. C.K. McCoy and T.J. Beckham, respectively). The APC plants were to remain under the direction of the APC President, with a similar chain downward of Mr. McDonald, Mr. Hairston, and the vice president for the Farley facility. Mr. McDonald and Mr. Hairston were officers of both APC and GPC.

During Phase I, which began on or about November 1, 1988, technical support was provided to all three nuclear facilities by a common Technical Services Group under a Vice President of Southern Company Services, Inc., who reported to the Executive Vice President, Mr. McDonald. Administrative support to all three facilities was provided by a common Administrative Services Group under another Vice President of Southern Company Services, Inc., who also reported to Mr. McDonald. Phase I was to be effective until the SEC approved the creation of Southern Nuclear. Mr. Farley was not identified as having any responsibility for operating the GPC nuclear facilities during this phase. He was responsible for providing administrative services through Southern Company Services, Inc., and was also responsible for the formation of SONOPCO. Although not in effect during Phase I, Mr. Farley had been designated to become the President and CEO of Southern Nuclear when it was established.

Phase II began on December 14, 1990, with SEC's approval of The Southern Company's request of June 22, 1988, to form Southern Nuclear, and the election of officers on December 18, 1990; the Southern Nuclear organization was effectively implemented January 1, 1991. As part of Phase II, GPC's Executive Vice President (Mr. McDonald) and Senior Vice President—Nuclear Operations (Mr. Hairston) became officers of Southern Nuclear and reported administratively to the President and CEO of Southern Nuclear, Mr. Farley. The vice presidents of each nuclear facility also became officers of Southern Nuclear. The Vice President of Technical Services and the Vice President of Administrative Services, respectively, for Southern Company Services, Inc., became officers of Southern Nuclear, rather than officers of Southern Company Services, Inc. During this phase, GPC and APC retained their NRC licenses and the responsibility for operating their respective nuclear facilities.

In Phase III, Southern Nuclear has operating responsibility for the Hatch and Vogtle facilities in accordance with the provisions of the NRC operating licenses for those facilities.²⁹

²⁹The NRC approved license amendments on November 22, 1991, that authorized the transfer of licenses for the Farley facility from APC to Southern Nuclear.

2. *Illegal Transfer Hearing and Petition Issues*

“Intervenor’s Prehearing Statement of Issues” (Statement of Issues), dated December 12, 1994, raised twenty-eight issues to support Intervenor’s illegal transfer issue for the license amendments proceeding.³⁰ The issues were submitted in support of Intervenor’s contention that the Vogtle operating license should not be transferred to Southern Nuclear because it lacks the requisite character and integrity. The twenty-eight issues repeat and further supplement assertions in the petition regarding an illegal transfer of control of GPC nuclear facilities. These issues are summarized below based upon the more detailed discussion of each issue in the Appendix to this Director’s Decision.

The gravamen of Intervenor’s twenty-eight issues, like the related issues in the petition, as supplemented, is that the nuclear officers in SONOPCO Project reported to Mr. Farley, rather than to Mr. Dahlberg, GPC’s CEO, and that Mr. Farley controlled the Vogtle facility based upon his involvement in (1) controlling daily operations; (2) establishing and implementing nuclear policy decisions; (3) employing, supervising, and dismissing nuclear personnel; and (4) controlling costs. Intervenor also asserts that numerous documents and statements provided to the NRC regarding the organizational structure and responsibilities for managerial control of the Vogtle facility were inaccurate or incomplete because they do not show Mr. McDonald reporting to Mr. Farley or Mr. Farley functioning as the *de facto* Chief Executive Officer of the SONOPCO Project.

Issues 1, 3-5, 7, 10, 11, 15, 16, 18, and 20-24 in Intervenor’s Statement of Issues assert Mr. Farley’s role with daily operations of the Vogtle facility and allege that GPC concealed this role and a *de facto* organization by providing the NRC inaccurate and incomplete information. As discussed in the Appendix to this Director’s Decision, Intervenor’s assertion that Mr. Farley functioned as the *de facto* Chief Executive Officer of the SONOPCO Project is not supported by the hearing record. Mr. McDonald did not report to Mr. Farley regarding GPC licensed activities. The items cited do not demonstrate that Mr. Farley exercised control over licensed activities at GPC’s nuclear facilities during his involvement in the SONOPCO Project. Rather, the record shows that GPC controlled the daily operations of the Vogtle facility in accordance with a chain of command extending from the Vogtle General Manager, through the Vice President of the Vogtle facility, through the Senior Vice President–Nuclear Operations, through the Executive Vice President–Nuclear Operations, to the President and CEO of GPC. A Nuclear Operations Overview Committee of the GPC Board of Directors

³⁰ Although Intervenor identified 28 issues in his Statement of Issues, two issues were both numbered 14A and 14B, and Intervenor presented no evidence or proposed findings on Issue 25.

conducted periodic reviews of the regulatory and operational performance of GPC's nuclear plants.

Issues 1, 9, 15, 17, and 20 of Intervenor's Statement of Issues (and page 4 of the October 1, 1990 Supplement to the Petition) include allegations that Mr. Farley controlled the Vogtle facility based upon his involvement with establishing and implementing nuclear policy decisions. As discussed in the Appendix to this Director's Decision, the hearing record shows that nuclear policy decisions for the Vogtle facility were established and implemented by GPC, and there was no evidence that Mr. Farley established the outage philosophy or any other operational policies for the Vogtle facility. Mr. Farley's limited involvement in a 1989 rate case matter before the Georgia Public Service Commission (i.e., his review of draft testimony regarding alternative performance standards) did not indicate any control of GPC's nuclear operations or licensed activities. Intervenor also provided no information that The Southern Company Management Council acted as the SONOPCO Project board of directors until the Project was incorporated.

Issues 1, 6, 8, 14A, 14B, 15, 19, 21, 27, and 28 of Intervenor's Statement of Issues (and pages 1-3 of the October 1, 1990 Supplement to the Petition), include assertions that Mr. Farley exercised control over nuclear personnel matters for the Vogtle facility because he (1) selected and approved GPC's management staff; (2) reviewed nuclear personnel in 1989 as evidenced by GPC Management Council's exclusion of nuclear personnel from its 1989 companywide review of management; (3) decided that Mr. Michael Barker, a GPC employee, would not be transferred from the SONOPCO Project to the Nuclear Operations Contract Administration (NOCA) group in Atlanta; (4) prepared Mr. McDonald's annual performance appraisal; and (5) implemented changes in Vogtle personnel evaluations and pay. As discussed in the Appendix to this Director's Decision, the record does not show that Mr. Farley controlled GPC nuclear facilities by employing, supervising, and dismissing nuclear personnel, or that GPC provided inaccurate information to the NRC regarding Mr. Farley's involvement with personnel matters.

Issues 1, 6, 12, 14A, 14B, and 17 of Intervenor's Statement of Issues allege that Mr. Farley's control of GPC nuclear facilities is shown through budget and personnel pay matters in that (1) Southern Nuclear, its predecessor, and The Southern Company controlled GPC's nuclear budget since November 1988; (2) Mr. Farley implemented changes in personnel evaluations and pay for Vogtle nuclear operations personnel; and (3) the GPC Management Council did not review GPC's 1990 nuclear operating budget. Intervenor asserts that inaccurate and incomplete information was provided to the NRC regarding GPC's control of budget and personnel pay matters. As discussed in the Appendix to this Director's Decision, the hearing record does not support a conclusion that GPC misrepresented its budgets affecting the operation of GPC licensed facilities.

There is no indication in the hearing record that the particular process GPC used to develop its budget is dispositive to Intervenor's assertion that Mr. Farley, The Southern Company, or SONOPCO Project controlled the operation of the Vogtle facility. Rather, the record shows that GPC was responsible for the costs of the Vogtle facility. After review by GPC's Management Council, the operating and capital budgets were approved by GPC's President and CEO, and the capital budget was also approved by the GPC Board of Directors. The record does not support that Messrs. Farley and Edward L. Addison, the President and CEO of The Southern Company, approved GPC's nuclear budgets. As an Executive Vice President of The Southern Company, Mr. Farley was involved in reviewing the nuclear budgets as part of the normal process for preparing annual budgets in the Southern system. Given The Southern Company's holding company status, Mr. Addison's involvement in reviewing and providing guidelines and requirements for adequate earnings and reasonable capital needs was appropriate, and did not constitute control of operations at GPC facilities.

Issues 1, 2, 12, 13, 18, 19, and 26-28 in Intervenor's Statement of Issues contain assertions that GPC managers provided inaccurate or incomplete information to the NRC when describing its organization and plans to form Southern Nuclear, and when responding to the petition. The alleged misrepresentations or omissions regard statements about (1) the Vogtle chain of command, (2) Mr. Dahlberg's relationship with Vogtle site management, (3) Mr. Farley's responsibilities as Executive Vice President-Nuclear of The Southern Company, (4) the 1989 title of Mr. Dahlberg, (5) SONOPCO Project's control over the Vogtle facility since November 1988, (6) the composition of the GPC Management Council, and (7) Mr. Farley's title in 1988. As discussed in the Appendix to this Director's Decision, the record shows that GPC provided some inaccurate or incomplete information to the NRC when describing its organization and plans to form Southern Nuclear, and when responding to the petition. This information involved (1) the omission of Mr. Hairston when Mr. McDonald described the Vogtle chain of command during a March 30, 1989 meeting (which was later corrected by GPC after reviewing the meeting transcript and was already accurately depicted in the FSAR); (2) a 1989 FSAR organizational chart showing the position of Mr. Dahlberg as "Chairman and CEO" rather than "President and CEO"; and (3) GPC's April 1991 written response to the petition indicating that the GPC Management Council included all senior vice presidents (which was inaccurate because Mr. Hairston was not a member), and indicating Mr. Farley's title in 1988 to be Executive Vice President-Nuclear of The Southern Company (a position he did not assume until March 1, 1989). While the NRC expects licensees to provide complete and accurate information, the inaccurate or incomplete information at issue here was of minor safety significance in terms of the NRC Staff's understanding of the proposed license transfers, did not mislead the NRC, and was not sufficient to warrant NRC enforcement action

or conclusions that (1) GPC concealed an unauthorized role of Mr. Farley or a *de facto*, unauthorized organization for control of GPC nuclear facilities; or (2) GPC lacks the requisite character and integrity to be a licensee.

3. NRC Staff Testimony During Hearing on Illegal Transfer Issue

NRC Staff testimony (hereafter, "Staff") regarding the alleged illegal transfer of control issue was jointly presented by Messrs. Frederick R. Allenspach, an NRR technical reviewer who reviewed the Administrative Controls section of the Vogtle Technical Specifications in 1987; Darl S. Hood, the Licensing Project Manager for the Vogtle facility; and John F. Rogge, Jr., formerly the Senior Resident Inspector at the Vogtle site during the time SONOPCO and Mr. Farley are alleged to have taken operating control of the Vogtle facility. These individuals provided evidence based upon their own personal knowledge and based upon their institutional knowledge derived from their work and their relation to other members of the NRC Staff who perform activities relating to the Vogtle facility encompassing the period 1987 to 1995.

The former Senior Resident Inspector's observation that GPC officials operated the Vogtle facilities was particularly significant in that he and Mr. Allenspach participated in the December 1988 inspection of the SONOPCO Project offices, interviewed GPC management, including Messrs. McDonald, Hairston, and McCoy concerning the management chain of command through Mr. McDonald, along with the organizational structure and supporting role of the SONOPCO Project. Mr. Rogge concluded that GPC was in control of Vogtle operations and that the changes in management personnel and organization beginning in 1988 did not affect GPC's control over Vogtle. He also concluded that the control and direction of daily operations at the Vogtle facility were performed by the onsite GPC employees under the direction of Mr. McCoy. Staff, ff. Tr. 2620, at 4-6. Mr. Rogge's conclusions were based on the Vogtle FSAR statements, the Vogtle TSs, and his interviews of Licensee personnel. Tr. 2159, 2716-17 (Rogge).

While the NRC did not inspect, or require to be reported, the number of times that GPC's Executive Vice President-Nuclear communicated with the President of GPC, the NRC Staff's focus regarding the conduct of operations is where nuclear safety has its immediate and greatest impact, i.e., on the nuclear power plant itself and its immediate management. Based on frequent visits and dealings with Vogtle staff at the level of Vice President-Vogtle and the Vogtle General Manager, plant operations appeared consistent with the organization described in the FSAR. Tr. 2656-57, 2664 (Hood).

The NRC Staff witnesses' visit at the Vogtle facility and corporate offices in Birmingham, Alabama, in September 1994 confirmed the accuracy of the Updated Final Safety Analysis Report (UFSAR) descriptions and figures, and

determined that GPC controlled operation of the Vogtle facility. Their conclusions were based upon discussions with numerous managers of GPC, SNC, and Southern Company Services, regarding their organizational responsibilities and structure, including details of their respective employment and their involvement with respect to the Vogtle facility, and discussions with the NRC's Resident Inspectors stationed at the Vogtle facility regarding their observations of the day-to-day control of the facility by GPC managers and the support services of SNC and Southern Company Services employees. Staff at 9.

The NRC Staff witnesses were present throughout the hearing regarding the illegal transfer issue, heard the evidence presented by all of the witnesses, and Mr. Hood was present during most of the depositions regarding illegal transfer. In their opinion, the hearing record disclosed no evidence to indicate that the operating licenses for the Vogtle facility had been transferred by GPC to SONOPCO Project or Southern Nuclear, or to otherwise alter the conclusion in the partial Director's Decision, DD-93-8, that GPC controls operations at the Vogtle facility. Tr. 2734 (Allenspach, Hood, Rogge).

In summary, the observations and testimony of key NRC Staff personnel involved with regulatory oversight and technical review of Vogtle's conduct of operations at the time of the alleged transfer of control indicate that GPC has maintained control of Vogtle operations and licensed activities. The testimony shows that the conduct of operations and support at the Vogtle facility has proceeded, and is proceeding, consistent with the phased reorganizations that were described at the outset to the NRC whereby Southern Nuclear will eventually become the sole operator of the GPC nuclear facilities.

4. Conclusion

On the basis of the foregoing, I conclude that GPC has not transferred control of the operating license for the Vogtle facility without the prior consent of the NRC. While Intervenor identified some inaccurate or incomplete information to the NRC by GPC, this inaccurate or incomplete information was either corrected or not significant in the context of the numerous communications regarding the three-phased transfer and the NRC's focus on areas that directly impacted plant operations and licensed activities. The inaccuracies identified do not show a pattern to deceive the NRC regarding the control of the Vogtle facility. Thus, there is no basis to conclude that GPC either misled the NRC or lacks the requisite character and integrity to be a licensee.

C. Diesel Generator Reporting and Reliability Issues (Petition § III.3; License Amendments Proceeding on DG Issue)

Petitioners allege in the section 2.206 petition, and Mr. Mosbaugh contended in the license amendments proceeding, that GPC knowingly provided inaccurate, incomplete, or misleading information regarding DG testing results and reliability (including the number of starts and the moisture content (i.e., "air quality") of DG starting and control air)³¹ in 1990, as well as in April 1991 statements regarding the knowledge and involvement of senior GPC officials with respect to inaccurate 1990 DG information.³² The alleged inaccurate, incomplete, or misleading information was provided in GPC's April 9, 1990 presentation and letter to the NRC (seeking permission to restart); in the April 19, 1990 LER on the Site Area Emergency (SAE); in a June 29, 1990 cover letter forwarding the revised LER and addressing GPC's QA audit and DG recordkeeping practices; in an August 30, 1990 letter; in GPC's Petition Response of April 1, 1991, as to Mr. Hairston's involvement in developing false DG start information during the April 19, 1990 telephone call and as evidenced by the actions of GPC managers when they became aware of inaccurate start counts. Petition at 10-11; Intervenor Findings at 78-235 and 263-311.

Petitioners also claim that the inaccurate, incomplete, or misleading information was conveyed in GPC's "White Paper" response during the August 1990 special team inspection in that it (1) excluded Messrs. Hairston and McCoy from the list of participants on the April 19, 1990 telephone call; (2) stated that all revisions were reviewed by the Plant Review Board (PRB); (3) indicated that Messrs. Jimmy Paul Cash (a Unit Superintendent) and George Bockhold worked together on the DG testing slide prepared for the April 9, 1990 presentation to NRC; and (4) omitted Mr. Kenneth Burr, a Southern Nuclear corporate engineer, from the list of individuals who wrote the April 9, 1990 letter. At hearing, Intervenor Mosbaugh also cited GPC's failure to include Safety System Performance Indicator Data in GPC's April 9, 1990 letter as another attempt to mislead the NRC.

³¹The air quality issues considered during the licensing hearing concerned GPC's March-April 1990 statements to the NRC, including the NRC's Incident Investigation Team (IIT), was whether GPC officials were willful or recklessly careless of the facts (as opposed to complete and accurate): (a) in the statement in the April 9 letter that air quality was satisfactory; (b) in the statement in the April 9 letter that recently obtained high dewpoint readings resulted from faulty instrumentation; and (c) in other communications with the NRC regarding high dewpoints. Memorandum and Order (Summary Disposition: Air Quality), dated April 27, 1995 (unpublished), at 6-9. Intervenor's claim that poor air quality was the root cause of the DG failures that caused the SAE was not within the scope of the hearing contention and is not considered in this Director's Decision. *See id.* at 6.

³²The petition stated that SONOPCO provided inaccurate false information; however, only corporate managers at Mr. George Hairston's level (Senior Vice President-Nuclear Operations) and above are officers of both GPC and SONOPCO/Southern Nuclear.

OI conducted an investigation and issued a report in December 1993.³³ OI concluded that: (1) the Vogtle General Manager deliberately presented incomplete and inaccurate information to NRC in the April 9, 1990 meeting and letter with respect to DG starts and air quality measurements; (2) a group of GPC senior managers conspired to submit a false statement in the April 19, 1990 LER; (3) the GPC Senior Vice President–Nuclear Operations, with at least a minimum of careless disregard, submitted a false statement in the June 29, 1990 letter transmitting a revision to the LER; (4) the Vice President–Vogtle Project, with at least careless disregard, submitted a false and misleading statement in an August 30, 1990 letter explaining why the April 9 letter was inaccurate; and (5) the GPC Executive Vice President–Nuclear Operations deliberately provided inaccurate information in an April 1, 1991 letter discussing participants in a late afternoon conference call on April 19, 1990.

The NRC Staff evaluated the results of the OI investigation of the DG issues and concluded that, contrary to section 50.9, GPC had provided inaccurate and incomplete information to the NRC on four separate occasions as a result of an inadequate regard, individually and collectively, by a number of senior GPC officials for complete and accurate communications with the NRC. The performance failures involved in the violations constituted a Severity Level II problem as cited in the May 9, 1994 Notice of Violation and the February 13, 1995 Modified Notice of Violation (wherein the NRC imposed a \$200,000 civil penalty).³⁴

1. March 20, 1990 Site Area Emergency

On March 20, 1990, a worker accidentally backed a truck into a switchyard support column causing a loss of offsite power at Vogtle Unit 1. At that time, Unit 1 was in a refueling outage, and one of the DGs (DG-1B) had been removed from service for a maintenance overhaul. The other DG (DG-1A) was available and was called upon to start twice, but on both occasions failed to maintain running speed. On a third attempt, the diesel started, restoring power 36 minutes after the loss of offsite power. This event prompted the declaration of an SAE.

On the same day as the event, GPC conducted several troubleshooting starts on DG-1A to determine, if possible, the cause of the event. The diesel started and ran without problems each of these times. The plant staff then shifted its attention to the DG-1B in order to return it to service expeditiously. As part

³³The allegation concerning SSPI data was not submitted to OI until after the report on DG statements was published. OI did not complete activities on this issue due to the staleness of the issue and the airing of the matter at hearing before settlement was reached.

³⁴In LBP-94-15, 39 NRC at 255-56, the Board ruled that allegations in the NRC's NOV issued May 9, 1994, were important to the admitted contention and within the scope of the proceeding.

of the effort to return the DG-1B to service, GPC performed a number of post-maintenance starts and tests between March 21 and March 24. During these tests, post-maintenance difficulties were experienced, including two failures of the diesel to start on March 21 because of inadequate fuel in the fuel lines after diesel reassembly. In addition, during a run on March 22, DG-1B tripped on a high lube oil temperature signal; during a run on March 23, the diesel tripped on low jacket water pressure and low turbo lube oil pressure signals; and during a run on March 24, a high jacket water temperature alarm was received but the diesel continued to run.

Immediately after the SAE, the NRC assembled an Augmented Inspection Team (AIT), which arrived at the Vogtle facility on March 22, 1990. On March 23, 1990, the NRC issued a Confirmation of Action Letter (GPC Exh. II-4) to GPC that, among other things, confirmed that GPC had agreed not to return Unit 1 to criticality until the Regional Administrator was satisfied that appropriate corrective actions had been taken, so that the plant could safely return to power operations. The letter also indicated that equipment involved in the incident may be quarantined (minimizing personnel access to areas and equipment consistent with safety) and that GPC could take any action it deemed necessary to (1) achieve or maintain safe plant conditions, (2) prevent further equipment degradation, or (3) test or inspect as required by the plant's TSs. A quarantine order was subsequently issued by the NRC concerning DG equipment. GPC Exh. II-65.

On March 24, Mr. William Shipman (General Manager-Plant Support) and Mr. C. Kenneth McCoy (Vice President-Vogtle Project) discussed with site personnel, including Mr. Bockhold (Vogtle General Manager) and Mr. Mosbaugh (Acting Assistant General Manager-Plant Support), concerns that these test results had raised about the pneumatic controls. The site was instructed to make sure the NRC and the AIT participated in the troubleshooting activities and received any documentation, and to obtain NRC concurrence before anything was changed. Prefiled Testimony of C. Kenneth McCoy on Diesel Generator Reporting Issues, ff. Tr. 2839, "McCoy DG," at 3-4.

On March 25, 1990, the NRC upgraded the AIT to an Incident Investigation Team (IIT),³⁵ composed of NRC and industry personnel and headed by the NRC.

After recovery from the SAE, GPC assembled an Event Review Team to identify the root causes of the event and to determine appropriate corrective actions. The Event Review Team included Messrs. Jimmy Paul Cash (Unit Superintendent), Paul Kochery (Vogtle Engineering Supervisor-Operations Modifications),

³⁵The results of this investigation are documented in NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990," dated June 1990. Staff Exh. II-46, at 1; *see, e.g.*, NUREG-1410, Appendix J (GPC Exh. II-167).

Georgie R. Frederick (onsite Supervisor of the Safety Audit and Engineering Review (SAER) group), and Tom Webb (Senior Licensing Engineer).

The NRC was informed of problems that occurred during the post-maintenance testing of DG-1B as indicated by a March 24, 1990 memorandum by Mr. Kendall (an AIT and IIT member) that identified the March 23, 1990 trip (low jacket water pressure and low turbo oil pressure, also called start number 134) as being significant.³⁶ The NRC was briefed on GPC's troubleshooting plan for additional testing of DG-1A and DG-1B. Testing on DG-1B was conducted on March 27 and March 28, and included sensor calibration and replacement, testing of the pneumatic logic controls, pneumatic leak testing, an undervoltage test, and an operational surveillance. It resulted in DG-1B being declared operable on March 28. The additional testing for DG-1A, which was similar in scope, was performed between March 29 and April 1, at which time DG-1A was declared operable. Additional starts on both diesels occurred after these tests, in order to establish the reliability of the diesels.

At the NRC's request, GPC also examined whether the diesel control air system could be the cause of the March 20 DG-1A failure. GPC tested the diesel air system for moisture and conducted a review of the control air filters. High dewpoint readings were recorded on DG-1A on March 29 and additional high dewpoint measurements were recorded on or about April 5-7, 1990. GPC eventually decided that most of the high readings were inaccurate.

On April 9, 1990, GPC gave an oral presentation to the NRC in support of GPC's request to return Vogtle Unit 1 to power operations after the SAE. In response to an NRC request that GPC address DG reliability at the meeting, Mr. Bockhold, the Vogtle General Manager, presented information on DG starts since the SAE using a viewgraph slide, which listed the sequence of testing on DG-1A and DG-1B and stated that there were "18 SUCCESSFUL STARTS" for DG-1A and "19 SUCCESSFUL STARTS" for DG-1B. GPC intended to convey to the NRC in the April 9 presentation (and the NRC understood) that there were eighteen and nineteen "consecutive successful" starts without problems or failures after the March 20 SAE. A written summary of the April 9 presentation was provided to the NRC in an April 9, 1990 letter, "Vogtle Electric Generating Plant Confirmation of Action Letter," signed by Mr. Hairston and reviewed by corporate managers and Mr. Bockhold. The summary, the Licensee's troubleshooting efforts, and the NRC's inspection activities were among the bases for the NRC's decision to authorize the restart of the facility on April 12, 1990.³⁷

³⁶ An April 6, 1990 GPC list of diesel starts from March 13 through March 23, which showed the problem starts on March 22 and 23, was also provided to the IIT.

³⁷ NRC conditions regarding the quarantine of equipment involved in the SAE and other measures to facilitate the IIT's investigation of the event that were stated in the March 23, 1990 Confirmation of Action Letter remained in effect.

2. Diesel Generator Statements

a. April 9, 1990 Presentation and Letter

Intervenor alleged that GPC, by and through its officers and employees, knowingly, deliberately, and willfully submitted inaccurate information to the NRC in an April 9, 1990 oral presentation and letter regarding the number of starts of the DGs. Intervenor contended that (1) GPC submitted the numbers eighteen and nineteen successful starts with full knowledge that the numbers were incorrect, and (2) a typed "Cash List" that showed the inaccuracies was a backup slide that was circulated to corporate offices before the presentation. See Tr. 8310, 8313-15 (Mosbaugh); Prefiled Testimony of Allen L. Mosbaugh, ff. Tr. 8263, "Mosbaugh," at 43-44; Intervenor Findings 85-89.

In the Modified NOV issued February 13, 1995, the NRC Staff concluded that, contrary to section 50.9:

[I]nformation provided to the NRC Region II Office by Georgia Power Company (GPC) in an April 9, 1990 letter and in an April 9, 1990 oral presentation to the NRC was inaccurate in a material respect. Specifically, the letter states that: "Since March 20, the 1A DG has been started 18 times, and the 1B DG has been started 19 times. No failures or problems have occurred during any of these starts."

These statements are inaccurate in that they represent that 19 consecutive successful starts without problems or failures had occurred on the 1B Diesel Generator (DG) for the Vogtle facility as of April 9, 1990, when, in fact, of the 19 starts referred to in the letter associated with the 1B DG at the Vogtle facility, three of those starts had problems. Specifically, Start 132 tripped on high temperature lube oil, Start 134 tripped on low pressure jacket water and Start 136 had a high temperature jacket water trip alarm. As of April 9, 1990, the 1B DG had only 12 consecutive successful starts without problems or failures rather than the 19 represented by GPC. The same inaccuracy was presented to the NRC at its Region II Office during an oral presentation by GPC on April 9, 1990.

The inaccuracy was material. In considering a restart decision, the NRC was especially interested in the reliability of the DGs and specifically asked that GPC address the matter in its presentation on restart. The NRC relied, in part, upon this information presented by GPC on April 9, 1990 in the oral presentation and in the GPC letter in reaching the NRC decision to allow Vogtle Unit 1 to return to power operation.

GPC asserts that the April 9, 1990 presentation and letter contained incorrect DG start-count information due to poor GPC internal communications and personnel mistakes, including by Messrs. Cash and Bockhold, and it was not due to indifference as to the need for accuracy. GPC August 30, 1990 Letter (GPC Exh. II-18); GPC Response to NOV, dated August 2, 1994 (Intervenor Exh. II-105), at 2; Letter from C.K. McCoy to Mr. James Lieberman, dated February 1, 1995 (GPC Supplemental Reply to NOV).

The NRC Staff found that the count errors were caused by performance failures in collecting and reporting the data, and found no evidence that GPC

employees deliberately and knowingly submitted, or conspired to submit, incomplete or inaccurate information. See Vogtle Coordinating Group Evaluation, Conclusions, and Recommendations, dated November 4, 1994 (Staff Exh. II-50) at 1-4; Testimony of David B. Matthews, Pierce H. Skinner, and Darl S. Hood on the Diesel Generator Issue (Staff DG Panel), ff. Tr. 14,758, at 11; May 1994 NOV (Staff Exh. II-46); Modified NOV (Staff Exh. II-51). The Staff found that the errors were caused by (1) Mr. Bockhold's failure in requesting the count to instruct Mr. Cash as to his criteria for a successful start (without a problem or failure),³⁸ the point at which to begin his count, and to assess the count data provided to ensure that it was what he had requested; and (2) Mr. Cash's failure in performing and reporting his count to ensure that the data provided were what Mr. Bockhold had requested. NOV (Staff Exh. II-46) at 2-3; Staff DG Panel at 4-5, 11.

The hearing record does not support Intervenor's position that the submission of eighteen and nineteen successful DG starts reported to the NRC by GPC in the April 9 presentation, and letter of the same date, were knowingly and willfully false.³⁹ While recollections were not clear about events occurring 5 years earlier, Mr. Bockhold testified that he intended to present a number of consecutive successful starts as support for GPC's position that the DGs would perform their intended function, and instructed Mr. Cash to review the operators' logs and determine how many consecutive successful DG starts had been made with no significant problems. Prefiled Testimony of George Bockhold, Jr., on Diesel Generator Reporting Issues, ff. Tr. 3309, "Bockhold DG," at 6; Tr. 3422, 3424 (Bockhold). Mr. Cash (an experienced Unit Superintendent and member of GPC's Event Review Team for the SAE) recalled that he was to determine the number of starts after the event that were without significant problems.⁴⁰ Prefiled Testimony of Jimmy Paul Cash on DG Reporting Issues, ff. Tr. 4389, "Cash," at 2, 3.

³⁸ The term "successful start" was ambiguous in that it was subject to various interpretations and is not defined by NRC in guidance documents such as Regulatory Guide 1.108. A count of successful starts without problems or failures was dependent upon having a definition for what constituted a successful start and the point at which to begin the count. Tr. 6875-76 (Greene); Tr. 5920-22 (Horton), see Tr. 5975-99, 5962. GPC witnesses had various interpretations of (1) "successful starts," (2) what constituted a problem start, and (3) when to begin the count. Tr. 6875 (Greene); Tr. 3547 (Bockhold); Tr. 5922 (Horton).

³⁹ Intervenor asserts that (1) the failure to utilize established review and verification procedures for the April 9 letter and (2) the failure to subject the letter to PRB review is circumstantial evidence that corporate officials (who were both GPC and Southern Nuclear employees) wanted to keep the DG start information or the air quality information free of meaningful verification. Intervenor Findings 130-159. While such actions may have disclosed problems in the count data, GPC's explanation that the April 9 letter was not handled as routine correspondence in order to expedite the drafting and review process is reasonable given that the TS do not require PRB review and its desire to expedite restart. See Tr. 2958 (McCoy). The mistakes exhibited, however, are of regulatory concern as cited in the Staff's enforcement action.

⁴⁰ In his June 14, 1993 OI interview, Mr. Cash stated that he viewed a "significant problem" as something that would have prevented the DG from running in an emergency. OI Exh. 10, at 11. At hearing, Messrs. Cash and Bockhold considered a start successful without significant problems to be one where the diesel had started

(Continued)

NRC personnel at the April 9, 1990 meeting were aware of DG testing, but did not know the number of consecutive successful starts of the DGs after March 20, 1990. Tr. 14,795 (Matthews); Hunt at 3-5. *See* Tr. 4949.⁴¹

Although Mr. Bockhold (and other GPC personnel) were aware of problems on the DG-1B during overhaul, he failed to adequately specify the starting point for the count to ensure that the count did not include these problems and failed to ensure that Mr. Cash, an experienced Unit Superintendent, understood his criteria for "successful starts" without problems or failures. Mr. Bockhold did not determine the point at which Mr. Cash began his count (i.e., the specific start number, date, or time) or whether his data included any starts with problems or failures. The hearing disclosed no evidence that Mr. Bockhold or other GPC personnel had any knowledge as to the number of starts of the DGs on April 9, 1990, other than the Cash count that was among the materials assembled quickly over the weekend prior to the April 9 presentation.⁴²

There is no evidence that a "Cash List" was a backup slide for the presentation or that corporate and site personnel otherwise knew that the April 9 DG start count was wrong.⁴³ Mr. Bockhold assigned Mr. Cash to count diesel starts; Mr. Cash did count diesel starts, and the numbers eighteen and nineteen presented to the NRC on April 9 were incorrect (i.e., they should have been twenty-nine and twelve on DG-1A and DG-1B, respectively).⁴⁴ GPC has admitted that the violation occurred and Mr. Bockhold's role and responsibility in the underlying events. *See* Letter from Hairston to NRC, dated August 30, 1990 (GPC Exh.

properly and reached rated voltage and frequency. Intervenor Exh. 57 (GPC Interrogatory Response, dated Aug. 9, 1993); Tr. 3426 (Bockhold). These definitions, however, were not used in any of GPC's April-August 1990 correspondence regarding the DG start-count information.

⁴¹ For example, on Tuesday, April 10, 1990, the day after the meeting between the NRC and GPC, Mr. Rick Kendall of the NRC's IIT, informed GPC that he could not duplicate the April 9 start count and asked for the start data. GPC Exh. II-31, at 5; Prefiled Testimony of John Gilbert Aufdenkampe, Jr., on Diesel Generator Reporting Issues (Aufdenkampe), ff. Tr. 4651, at 4-5.

⁴² While it is clear that the April 9 start count was derived from Mr. Cash's efforts, there is conflicting evidence as to exactly what information Mr. Cash provided to Mr. Bockhold. On April 19, 1990, Mr. Cash told Messrs. Mosbaugh and Aufdenkampe (GPC Manager of Technical Support) that he gave Mr. Bockhold "totals" and not information on starts and stops. Tape 58 Transcript (GPC Exh. II-2) at 36. Mr. Bockhold testified during his OI interview on August 14, 1990, that Cash gave him start totals. OI Exh. 12 (Intervenor Exh. II-13) at 8. Mr. Cash stated in his August 1990 OI Interview (Intervenor Exh. 190), however, that he gave Mr. Bockhold both total start numbers and a list of starts. In his June 1993 OI interview, he said that although he could not recall specific numbers, he gave Mr. Bockhold the numbers greater than 18 and 19. OI Exh. 10, at 48-50. At the hearing he could not remember exactly what count he gave Mr. Bockhold, but believed he gave him the numbers 18 and 19 for DG-1A and DG-1B, respectively, or possibly 23 starts for DG-1B and 27 for DG-1A as was apparent from a typed listing of starts located by GPC in 1993 (Intervenor Exh. 41 and GPC 23). *See* Tr. 4547-48, 4541, 4463-64 (Cash). Even though Mr. Cash stated that GPC Exh. II-23 was a typed version of his list for April 9, he was uncertain during cross-examination and he could not recall having his handwritten list typed or including starts prior to March 20, 1990, that were recorded on the listing. In light of these statements, it is difficult to determine what information Mr. Cash provided to Mr. Bockhold.

⁴³ For example, Mr. Bockhold was not specifically told that the April 9 (and April 19) start counts were wrong until April 30 and May 2, 1990, when Mr. Mosbaugh gave him a listing of DG starts that showed the errors. *See* Bockhold at 14; Mosbaugh April 30, 1990 Memo (Intervenor Exh. II-29).

⁴⁴ *See* August 30, 1990 Letter (GPC Exh. II-18), Tables 1 and 2. The underreporting of the DG-1A start count was not relevant to the enforcement action.

II-18); Modified NOV (Staff Exh. II-51); GPC Supplemental Reply to the NOV, dated February 1, 1995.

In sum, the assertion that GPC deliberately provided false DG start information in the April 9 letter and presentation was not substantiated.

b. April 19, 1990 Licensee Event Report

Mr. Mosbaugh alleged that a disputed portion of a taped conversation from the afternoon of April 19, 1990 (Tape 58 Transcript (GPC Exh. II-2)) regarding the draft LER, is evidence that a number of GPC vice presidents and plant personnel engaged in a criminal conspiracy to intentionally submit false information to the NRC in that GPC intentionally iterated the same false April 9 count information to the NRC in LER 90-006. Tr. 8411-12, 9982 (Mosbaugh). His assertion is based on his version of the following excerpt:

Shipman: Let's see. What other questions do we got? We got the start thing straightened out.

Hairston: [Interrupting]. We got the starts — So we didn't have no, didn't have no trips?

Shipman: No, not, not

McCoy: Let me explain. I'll testify to that.

Shipman: disavow. What else do we have Jack?

GPC Exh. II-2, at 11-14.

Mr. Mosbaugh also asserts that GPC tried to exclude him from the telephone conversation taped on April 19, 1990.

In the Modified NOV issued February 13, 1995, the NRC Staff concluded that, contrary to section 50.9:

[I]nformation provided to the NRC by GPC in a Licensee Event Report (LER), dated April 19, 1990, was inaccurate in a material respect. Specifically, the LER states: "Numerous sensor calibrations (including jacket water temperatures), special pneumatic leak testing, and multiple engine starts and runs were performed under various conditions. After the 3-20-90 event, the control systems of both engines have been subjected to a comprehensive test program. Subsequent to this test program, DG1A and DG1B have been started at least 18 times each and no failures or problems have occurred during any of these starts."

These statements are inaccurate in that they represent that at least 18 consecutive successful starts without problems or failures had occurred on the DGs for Vogtle Unit 1 (1A DG and 1B DG) following the completion of the comprehensive test program of the control systems for these DGs, when, in fact, following completion of the comprehensive test program of the control systems, there were no more than 10 and 12 consecutive successful starts without problems or failures for 1A DG and 1B DG respectively.

The inaccuracy was material in that knowledge by the NRC of a lesser number of consecutive successful starts on 1A DG and 1B DG without problems or failures could have a natural tendency or capability to cause the NRC to inquire further as to the reliability of the DGs.

Staff Exh. II-51 at 1 and 20.

Under 10 C.F.R. § 50.73(a)(1), GPC was required to submit an LER, including a description of the event (10 C.F.R. § 50.73(b)(1)) and a description of corrective action taken (10 C.F.R. § 50.73(b)(3)) by April 19, 1990 (30 days after the SAE).

The evidence does not support the claim that the above words from Tape 58 demonstrate a criminal conspiracy by high officials in GPC to present false information to the NRC. Tape 58 contains multiple, disjointed, jumbled, and often inaudible conversations which do not demonstrate conspiracy to intentionally provide inaccurate information to the NRC. The NRC Staff found that the taped statements were not sufficient to establish an intention to deceive or mislead the NRC.⁴⁵ Further, there was no evidence to support Mr. Mosbaugh's claim that Mr. Mosbaugh joined the call late because GPC tried to keep him off the call with corporate managers about the accuracy of the LER. *See* Mosbaugh at 35, 48; Shipman at 5; Tr. 10,932-33, 10,976-77 (Shipman); and Tr. 4794-4801, 5428 (Aufdenkampe).

On April 10, 1990, Mr. Mosbaugh became aware of the April 9 letter and he and other site personnel (particularly Mr. Aufdenkampe) became concerned that the statement that the "starts were without problems or failures" may have been a material false statement to the NRC because of known DG failures after the SAE. Mosbaugh at 32; Tr. 4752-53 (Aufdenkampe). Mr. Richard Kendall of the IIT also asked GPC for data supporting the April 9, 1990 DG start count because he could not get the same numbers. IIT Teleconference Transcript, dated April 10 (GPC Exh. II-31).⁴⁶

Mr. Webb, an engineer in the group that reported to Mr. Aufdenkampe (who reported to Mr. Mosbaugh), used the same diesel start language for the draft LER that was in the April 9 letter. McCoy DG at 10-11; Prefiled Rebuttal Testimony of Thomas E. Webb on Diesel Generator Reporting Issues, ff. Tr. 13,096, "Webb," at 2-3; GPC Exh. II-171-B. Concerns about the accuracy of the count led the site to delete the start numbers from the draft LER and state

⁴⁵The NRC Staff version of the transcript states:

Hairston: We got the starts — so we didn't have no, didn't have no trips?

Shipman: No, not, not

McCoy: [Inaudible] three. I'll testify to that.

Shipman: [Inaudible] disavow. What else do we have Jack?

GPC also offered a transcript version of this exchange. The tape excerpt was played several times at the hearing in attempts for the Board and the reporter to discern the inaudible portions, which proved unsuccessful.

⁴⁶No listing of start counts through April 9 was ever located among the voluminous records and documents collected by the IIT.

that the diesels had been “started several times and no failures or problems have occurred during any of these starts.” Webb at 4. In response to a Plant Review Board (PRB) comment on April 18, 1990, the phrase “several starts” was replaced with “more than twenty times each” by adding April 10-18 starts in the control room logs to the numbers reported April 9. Webb at 5-7.⁴⁷ PRB Meeting Minutes 90-59 (GPC Exh. II-28) at 4; Webb at 5-6; Tr. 13,211 (Webb); Aufdenkampe at 2.

The site received notice on the morning of April 19, 1990, that Mr. Hairston wanted the phrase “greater than twenty” to be verified. Prefiled Testimony of W. George Hairston, III, on Diesel Generator Reporting Issues, ff. Tr. 3531, “Hairston DG,” at 6; GPC Exh. II-25; Stringfellow at 2; Tr. 4058 (Stringfellow); Tr. 4786-87 (Aufdenkampe); Webb at 6. The April 19 PRB, which was chaired by Mr. Kitchens, Assistant General Manager—Operations and held that afternoon, similarly advised that the phrase be verified, reworded, or deleted based on verification efforts. Tape 57 Transcript (GPC Exh. II-1) at 15-16; PRB Meeting 90-60 Minutes (GPC Exh. II-29).

After the PRB meeting, Messrs. Aufdenkampe and Mosbaugh discussed the draft LER by phone with corporate personnel and informed them that efforts to verify the count were ongoing. Mr. Mosbaugh told Mr. Shipman (General Manager—Plant Support for Vogtle Project) that there were two DG-1B trips (i.e., on March 22 at 12:43 (high lube oil temperature) and on March 23 at 17:31 (low jacket water pressure—turbine lube oil pressure)) which he believed rendered the statement inaccurate. Tape 57 Transcript (GPC Exh. II-1) at 59-60. Mr. Shipman emphasized the need to provide accurate information to the NRC, regardless of what George [Bockhold] told [Stewart] Ebnetter. *Id.* at 62.

During another phone call regarding the LER between site and corporate managers (Messrs. McCoy, Stringfellow, Bockhold, Aufdenkampe, Mosbaugh, and Bockhold),⁴⁸ Mr. McCoy also emphasized the need to be certain about the number after completion of the comprehensive control test program (hereafter “comprehensive test program” or “CTP”). Tape 58 Transcript (GPC Exh. II-2) at 8. Mr. Bockhold strongly stated that his April 9 start counts were subsequent to completion of a comprehensive test program and were “verified correct” by Mr. Cash. GPC Exh. II-2, at 8. Mr. Bockhold’s statement implied that GPC need not await the completion of site verification efforts that Mr. Aufdenkampe reported were under way to confirm the accuracy of the draft LER.⁴⁹

The term “comprehensive test program,” however, was ambiguous in that GPC had not agreed upon definition of what it meant. Neither GPC personnel

⁴⁷ Mr. Webb developed the list of starts using control room logs knowing that an up-to-date start log with numbered starts was not available. Webb at 6-7.

⁴⁸ This call is often referred to as “Call A” on the April 19 LER.

⁴⁹ Mr. Webb’s effort to verify the count was accomplished from noon to around 4 p.m. on April 19 and was in progress during the call.

at the site on April 19, 1990, nor the NRC inspection staff present during troubleshooting, knew the parameters of the comprehensive test program (i.e., when it began or ended).⁵⁰ The change of the start-count wording from “since March 20” to “subsequent to this test program” [the CTP] defined a different starting point for counting diesel starts and created ambiguity in the LER. The LER words were changed without completely verifying the facts, or defining the time period involved as Mr. Webb (the individual who performed the count for the LER) was never instructed to collect consecutive successful starts without problems or failures after the comprehensive test program.⁵¹ GPC’s reliance on verbal assurances and inadequate verifications is a second instance cited in the violation of inadequate verification of information to be provided to the NRC.⁵² While it is unclear whether GPC site personnel realized that the list compiled on April 19, 1990, showed that the April 9 start count of eighteen consecutive starts on DG-1B was inaccurate, it is clear that the list neither confirmed nor disputed the accuracy of the April 19 LER in that Mr. Webb was not told to get consecutive successful starts or starts after completion of the CTP. *See* Webb List (GPC Exh. II-71); Webb at 6-8.

Even though Mr. Mosbaugh questioned the accuracy of the count after the CTP, and suggested that it might not end until the undervoltage (UV) test just before the DGs were declared operable, site and corporate personnel (Messrs. Mosbaugh, Shipman, and Aufdenkampe), approved the LER with the “comprehensive test program” language included. Tape 58 Transcript (GPC Exh. II-2) at 8, 22-23.⁵³ The record shows that GPC’s (including Mr. Mosbaugh’s) incomplete efforts to verify the LER start count caused erroneous DG start information to be submitted in the April 19 LER. GPC inserted the words “comprehensive test program” with the intent to exclude the problem starts identified and relied on incorrect, verbal assurances that the count statement “at least eighteen times each” was correct. *Id.* at 8-34. Although they acknowledged during discussions of the draft LER that they did not know the

⁵⁰ Among those who did not know what the CTP was, what its parameters were, or when it started or stopped were Messrs. Cash (Tr. 4471), McCoy (Tr. 6995), Webb (Tr. 5696-97, 13,128), and Stringfellow (Tr. 4069-74) of GPC; and Messrs. Hunt (Tr. 4993), and Kendall (Tr. 5036), the NRC employees who monitored diesel testing and other activities in 1990.

⁵¹ A copy of Mr. Webb’s list (with notations in black and red ink that were written by Mr. Mosbaugh at a later date) was admitted as GPC Exh. II-71. *See* Webb at 6-7. The list contained some information on stops and starts, and noted that the total starts identified through March 20–April 18 were 32 for DG-1A and 27 for DG-1B. The totals shown were not an accurate count of *consecutive successful starts without problems or failures*, but merely totaled all starts identified after March 20. For example, the list did not identify the problem on start 136 or two starts on the morning of April 19. *See* GPC Exh. II-71 (“Webb List”): August 30, 1990 Letter (GPC Exh. II-18), Attachment B; June 29, 1990 QA Audit Report (GPC Exh. II-15).

⁵² The audio tape recording of conversations on that date shows that Mr. Mosbaugh and Mr. Aufdenkampe did not examine Mr. Webb’s list until after the site had approved the revised language in the LER. *See* Tape 58 Transcript at 8-34. The list did not contain a notation as to when a UV test was run on either diesel.

⁵³ This conversation (i.e., when the site approved the last revision of the LER, is often referred to as “Call B” regarding the LER.

starting point for the count (i.e., the first start following completion of the CTP), Messrs. Mosbaugh, Aufdenkampe, and Shipman failed to clarify and verify the starting point for the count of successful consecutive DG starts reported in the LER. There is no evidence, however, that any GPC or SONOPCO employee involved knew the exact number of starts following the CTP on April 19 or had a listing of starts (whether prepared by Mr. Cash or Mr. Webb) before the LER was approved.⁵⁴ The inadequate verification efforts were geared toward defending information already provided to the NRC by changing the description of the period for the count (the CTP actually identified a subset of the consecutive successful starts without problems or failures after the SAE). GPC's lax verification efforts were caused in part by unjustified assurances by Mr. Bockhold that information (which was assembled quickly using ambiguous definitions) had been verified before being presented. As a result, GPC did not identify inaccuracies in the April 9 and April 19 start counts and the mistakes of Messrs. Bockhold and Cash in collecting and reporting the initial count. This failure was among those cited as a basis for the Severity Level II violation against GPC.

Therefore, the allegation that GPC employees, either individually or collectively conspired deliberately to provide inaccurate information was not substantiated.

c. June 29, 1990 Cover Letter and Revised LER

The Petitioners allege, as supplemented by Intervenor in the licensing hearing, that GPC deliberately submitted false information to the NRC in a June 29, 1990 cover letter to a revised LER, concerning the reasons for the error in the LER in that (1) Messrs. Hairston and McCoy knew that the information was false, (2) neither Mr. Bockhold nor Mr. Cash informed Mr. Mosbaugh that there was a listing of the April 9 start data when Mr. Mosbaugh questioned the count, (3) there were different reasons for the error stated in the various drafts of the cover letter, (4) the Quality Assurance (QA) audit (which was the basis for some of the statements in the cover letter to the LER Revision) was narrow in scope and did not review all pertinent information, and (5) GPC was on notice that the reason stated in the letter was false. Intervenor Findings 350-351; *see* Petition at 10-11.

In the Modified NOV, the NRC found that, contrary to the requirements of section 50.9, the LER cover letter, dated June 29, 1990, was inaccurate and incomplete in material respects as evidenced by the following examples:

⁵⁴ Accurate information was available in the Unit 1 Control Log which recorded the time and date of DG starts and stops, and noted alarms and other pertinent information. Mr. Cash had used this log and the Shift's Supervisor's Log for the April 9 counts.

The letter states that: "In accordance with 10 C.F.R. 50.73, Georgia Power Company (GPC) hereby submits the enclosed revised report related to an event which occurred on March 20, 1990. This revision is necessary to clarify the information related to the number of successful diesel generator starts as discussed in the GPC letter dated April 9, 1990. . . ."

1. The LER cover letter is incomplete because the submittal did not provide information regarding clarification of the April 9, 1990 letter.

The incompleteness was material in that the NRC subsequently requested GPC to make a submittal clarifying the April 9, 1990 letter.

The letter states that: "If the criteria for the completion of the test program is understood to be the first successful test in accordance with Vogtle Electric Generating Plant (VEGP) procedure 14980-1 "Diesel Generator Operability Test," then there were 10 successful starts of Diesel Generator 1A and 12 successful starts of Diesel Generator 1B between the completion of the test program and the end of April 19, 1990, the date the LER-424/1990-06 was submitted to the NRC. The number of successful starts included in the original LER (at least 18) included some of the starts that were part of the test program. The difference is attributed to diesel start record keeping practices and the definition of the end of the test program."

2. The last sentence in the above paragraph is inaccurate because diesel record keeping practices were not a cause of the difference in number of diesel starts reported in the April 19, 1990 LER and the June 29, 1990 letter. The difference was caused by personnel errors unrelated to any problems with the diesel generator record keeping practices.

The inaccuracy was material in that it could have led the NRC to erroneously conclude that the correct root causes for the difference in the number of diesel starts reported in the April 19, 1990 LER and the June 29, 1990 letter had been identified by GPC.

3. The last sentence in the above paragraph is also incomplete because it failed to include the fact that the root causes for the difference in the number of diesel starts reported in the April 19, 1990 LER and the June 29, 1990 letter were personnel errors. First, the Vogtle Plant General Manager who directed the Unit Superintendent to perform the start count (which formed the basis for the April 19, 1990 LER) failed to issue adequate instructions as to how to perform the count and did not adequately assess the data developed by the Unit Superintendent. In addition, the Unit Superintendent made an error in reporting his count. Second, the [Acting Assistant General Manager-Plant Support⁵⁵], the General Manager for Plant Support and the Technical Support Manager failed to clarify and verify the starting point for the count of successful consecutive DG starts reported in the April 19, 1990 LER.

The incompleteness was material in that, had correct root causes for the difference in the number of diesel starts reported in the April 19, 1990 LER and the

⁵⁵The NRC corrected Mr. Mosbaugh's position designation in a letter from Mr. J.L. Milhoan, NRC, to Mr. C.K. McCoy, GPC, dated March 13, 1995.

June 29, 1990 letter been presented, this information could have led the NRC to seek further information.

Staff Exh. II-51, NOV at 2-3.

GPC asserts that the incomplete and inaccurate statements regarding the reasons for the errors in the LER (and April 9 letter) were based on reasonable attempts to provide an explanation based on the results of the QA audit report (GPC Findings at 140-63) and admits and accepts responsibility for the incompleteness of the letter (GPC Findings 285, 347). GPC maintains that DG record-keeping practices contributed to the reporting of erroneous counts (noting that the NRC Staff acknowledged that those practices may have contributed to violations as events unfolded). GPC Findings 286-291.

The NRC Staff viewed the performance failures of GPC site and corporate personnel, particularly by those who were on notice of Mr. Mosbaugh's concerns that the cover letter to the LER Revision was inaccurate and incomplete (i.e., Thomas Greene, the Vogtle Assistant General Manager—Plant Support; Michael Horton, the Vogtle Manager—Engineering Support; Mr. Frederick, the Supervisor—SAER; and Harry Majors, a Licensing Engineer for the Vogtle Project) as serious, but found that there was insufficient evidence to conclude that GPC intentionally provided inaccurate or misleading information. *See* Staff DG Panel at 6-11; NOV (Staff Exh. II-46); and Modified NOV (Staff Exh. II-51).

(1) "PRIOR" KNOWLEDGE OF MESSRS. HAIRSTON AND McCOY AND NARROW-SCOPE AUDIT

Petitioners are correct that the QA audit was narrow in scope. There is no evidence, however, that either Mr. Hairston or Mr. McCoy knew that incomplete and inaccurate reasons were stated in the June 29, 1990 LER Revision cover letter as to why the LER contained erroneous start-count information⁵⁶ or that they intended to deceive the NRC. On the contrary, as described below, the events leading to the development of the letter show that these GPC officials and other GPC employees, endeavored, albeit unsuccessfully, to provide correct information.

On April 20, 1990, Mr. Webb was surprised by the LER phrase "subsequent to the test program" and thought the LER could be inaccurate because, on April 19, he had identified only about ten or eleven starts after operability of

⁵⁶For example, both Mr. Hairston and Mr. McCoy acknowledged during the hearing — as GPC conceded in its response to the enforcement action — that errors in the April 9 letter and presentation and the April 19 LER were also due to inadequate performance by GPC personnel, including Messrs. Cash and Bockhold. *See* McCoy DG at 21; Tr. 11,557-59 (Hairston); GPC Supplemental NOV Reply at 2-3.

the DGs. Webb at 8-9. Mr. Mosbaugh later generated his own list of DG-1B starts using the Unit 1 Control Log, the Shift Supervisor's Log, and the Diesel Start Completion Sheets, and, on April 30 and May 2, 1990, he informed Mr. Bockhold and Mr. Aufdenkampe that the April 9 and April 19 counts were wrong and for different reasons.⁵⁷ Tr. 5211-12 (Mosbaugh); Tape 75 Transcript (GPC Exh. II-34 and Staff Exh. II-38) at 31. Mr. Bockhold instructed Mr. Mosbaugh to see that the LER was revised and indicated he might correct the April 9 start count in a planned mid-May 1990 submittal on DG component testing. Mosbaugh at 37; Tape 90 Transcript (Staff Exh. II-14) at 1-2; Bockhold at 15.

By May 8, 1990, when Mr. Mosbaugh chaired the PRB in his capacity as Acting Assistant General Manager-Plant Support,⁵⁸ the PRB approved a draft revised LER which stated that:

After the 3-20-90 event, the control systems of both engines were subjected to a comprehensive test program which culminated in control logic tests on 3-30 for DG1A and 3-27-90 for DG1B. Subsequent to this test program, DG1A and DG1B had been started 11 times each (through 4-19-90) and no failures or problems have occurred during any of these starts.

PRB Meeting 90-66, GPC Exh. II-37. Other revisions followed that updated the consecutive successful starts through May 14, 1990,⁵⁹ and were transmitted to the corporate office licensing engineer who was responsible for drafting the revised LER. Webb at 9-10; Tr. 4047-50 (Stringfellow).

The site's inability to come up with a firm count number frustrated Mr. Hairston, however, in that he had to report to the NRC Regional Administrator, Mr. Stewart Ebnetter, on May 14, 1990, and on June 14, 1990, that the start-

⁵⁷ Mr. Mosbaugh gave Mr. Bockhold a handwritten list of DG-1B starts that confirmed that there were only 11 DG-1B starts after the "UV Test" (the end of the CTP in his opinion). Mosbaugh at 36; Intervenor Exh. II-29; Tape 90 Transcript (Staff Exh. II-14) at 8. He also told Mr. Bockhold that the April 9 and April 19 counts were wrong for different reasons.

⁵⁸ During a May 10, 1990 PRB meeting (PRB Meeting Minutes 90-67 (GPC Exh. II-39)), Mr. Mosbaugh (acting as Chairman of the PRB) assigned Mr. Bockhold the action of determining how the April 9 letter would be corrected, but on May 24, 1990, Mr. Bockhold closed the action item without correcting the April 9 letter. Aufdenkampe at 17; Mosbaugh at 38; Intervenor Exh. II-33. Mr. Mosbaugh believes he was removed from the PRB due to his concerns about false statements to the NRC. Mosbaugh at 37-38; The hearing record revealed only that, on May 10, 1990, Mr. Mosbaugh was removed from the PRB and became a Technical Assistant to Mr. Bockhold because Mr. Greene resumed his positions as Assistant General Manager-Plant Support after attending Senior Reactor Operator training. Prefiled Testimony of Thomas V. Greene, Jr., on Diesel Generator Reporting Issues, ff. Tr. 6716 (Greene), at 1.

⁵⁹ It was standard practice for an LER to update information previously provided to the NRC. Tr. 13,137 (Webb). GPC ultimately decided to forego the term successful start and report valid tests and failures, as defined in RG 1.108, extending through June 7, 1990. Revised Prefiled Rebuttal Testimony of Thomas E. Webb on Diesel Generator Reporting Issues, "Webb Revised," ff. Tr. 13,168 (Webb), at 9-13. See LER Revision and Cover Letter (GPC Exh. II-16); GPC Exhs. II-171L through 171T. This approach, while providing unambiguous information regarding DG starts, did little to correct the statement of consecutive starts without problems or failures through April 9 or 19 in that it reported starts using a different criterion and over a different period than stated in the prior documents. The cover letter only corrected the April 19 start count (10 and 12 for DG-1A and DG-1B, respectively) based on the narrow-scope audit.

count numbers were revised. Mr. Hairston directed Mr. McCoy to keep the NRC informed of efforts to correct the count.⁶⁰ Hairston DG at 9-13; Tr. 3214 (McCoy).⁶¹ When he saw that the draft LER revision and cover letter contained no explanation as to why the start data were different, Mr. Hairston directed that a QA audit be conducted to determine (1) the correct start count and (2) the reason GPC could not get the number straight. Hairston DG at 11-12; Tr. 3631 (Hairston). He also informed Mr. Ebnetter that he would submit a revised LER after completion of the QA audit. Hairston DG at 12-13.

There is no basis to conclude that either Mr. Hairston or Mr. McCoy knew that the information provided in the June 29 cover letter was false. Mr. Hairston's actions demonstrated a concern for accuracy and an attempt to discern why erroneous information was given to the NRC. He and Mr. McCoy read the audit report and the table of starts appended to it to ensure that the count information was correct. Hairston DG at 14. Mr. Hairston also instructed that the QA audit results be provided to the Resident Inspector at the Vogtle site and that an explanation of the differences in the count numbers between the LER and the revised LER be explained in the transmittal letter to the revised LER. Hairston DG at 14-15. Mr. Hairston and Mr. McCoy adopted the implied finding in the audit report that DG record-keeping practices were the source of the erroneous information provided on April 19. Hairston DG at 16-17; McCoy DG at 19-21.

Unfortunately, (1) the narrow scope of the QA audit resulted in GPC selecting an incorrect or incomplete reason for the LER error; and (2) neither Mr. Hairston, Mr. McCoy, nor the other GPC employees involved noticed that the QA audit showed that the April 9 start count was wrong.

The audit's failure to examine the performance of site personnel in collecting and reporting the initial counts rendered GPC unprepared to reach a complete assessment of the causes of the April 9 start-count errors. There was no evidence that the narrow scope of the audit was part of an effort to deceive the NRC.

The QA audit report specifically stated that the audit was narrow in scope and did not identify a specific cause for the LER count errors, but implied they were caused by the failure to specify a starting point for the count and

⁶⁰ On June 15, 1990, Messrs. Aufdenkampe and Mosbaugh told the NRC resident inspectors about the errors and that the correct numbers depended on when you start counting. Aufdenkampe at 18. After Messrs. Brockman and Ebnetter received calls that the DG start information was incorrect, the NRC met to discuss whether the erroneous count was cause to reconsider the April 12 restart decision. Tr. 15,319-20, 15,330-31, 15,332 (Reyes). Mr. Reyes, the Deputy Regional Administrator for Region II, recalled that eight starts would have been sufficient in his opinion. Tr. 15,336-37 (Reyes). Mr. Reyes believed that GPC's testing, corrective actions and confirmatory testing after the event provided assurance that problems with the DGs during the SAE had been resolved. Tr. 15,322-23 (Reyes).

⁶¹ Intervenor asserted that the phone call was too short to convey DG information and, instead was about an event at Hatch occurring on that date. Intervenor Findings 339-346. Such speculation is not sufficient to rebut GPC's testimony regarding these calls.

the lack of up-to-date DG record-keeping practices.⁶² The QA audit report, however, alluded to this faulty conclusion without confirming that accurate start data were not otherwise available in April 1990 (i.e., from the Unit 1 Control Log that Mr. Cash had also used, which, unlike the Shift Supervisor's Log, contained sufficient information to derive accurate count data).⁶³ The audit was also inadequate in scope because it did not examine the performance of Mr. Bockhold and Mr. Cash in collecting and reporting the initial April 9 data (the failure to define the criteria for "successful start" and the period for the count), the assurances of Mr. Bockhold that deterred site verification efforts, or the failure of site and corporate personnel to define the CTP.⁶⁴ Thus, the audit failed to identify their inadequate performance as causes for the erroneous information reported on April 9 and in the April 19 LER.

While better DG record-keeping practices (i.e., no delays in routing or completing start completion sheets, and an up-to-date DG Start Log with starts numbered) would have made count information easier to retrieve, it is clear that previous erroneous start counts were caused by (1) the performance failures of Messrs. Bockhold and Cash in initially collecting and reporting the data (particularly with respect to the ambiguous term "successful start" and the undefined period for the count)⁶⁵ and (2) GPC's decision to reiterate the count (as modified by the term CTP) without completing adequate verification efforts. There is no evidence that Messrs. Hairston and McCoy were specifically aware of this cause of the errors, as there was no evidence that Mr. Mosbaugh's reasons for believing the letter was inaccurate were ever communicated to them. Thus, there is no basis to support Mr. Mosbaugh's assertion that GPC intended to mislead the NRC.⁶⁶

⁶² The DG Start Log, compiled from completion sheets filled out by operations personnel and reviewed by the DG Engineer, Mr. Stokes, was not up to date on April 19 as there were delays in the routing of the Completion Sheets from the operators to the Engineering Support Department (headed by Mr. Michael W. Horton) and operators had not filled out a sheet every time the DG was started. Prefiled Testimony of Georgie R. Frederick on Diesel Generator Reporting Issues, ff. Tr. 4125, "Frederick," at 7.

⁶³ Pursuant to GPC procedures, the Unit 1 Control Log was to contain the start time, stop time, and any significant status changes for each DG start. Procedure 10001-C, Logkeeping (Staff Exh. II-31) at 2; Tr. 4232 (Frederick). The starts with problems and/or failures (Starts 132, 134, and 136) were all recorded in the Unit 1 Control Log (Staff Exhs. II-23, II-24); Tr. 4232 (Frederick). The counts reported in the April 19 LER (and the April 9 letter start count) included starts before the operability test was conducted.

⁶⁴ Mr. Frederick was aware that Mr. Cash had prepared the information for Mr. Bockhold's presentation, and had assumed that a separate count had not been made for the LER. Mr. Frederick had not contacted Mr. Cash during the audit to avoid biasing the results of the audit. Frederick at 9-12. This approach, while reasonable from an auditor's perspective, was not prudent given the performance problems associated with collecting the DG start counts and Mr. Mosbaugh's statements to Mr. Frederick early in the audit period that he should examine the role of personnel errors in the erroneous counts. Tape 160 Transcript (Staff Exh. II-16) at 24.

⁶⁵ GPC did not define what constituted a "successful start without problems or failures" in the audit report and did not agree on a definition until the August 30, 1990 letter that submitted accurate DG-1A and DG-1B start counts for April 9.

⁶⁶ Messrs. Bockhold, McCoy, and Hairston also failed in their review of the document to ensure that information provided to the NRC in the June 29 cover letter was complete due to their failure to "clarify" the April 9 letter and to provide a relevant discussion of the start count.

Even though senior managers may not be intimately familiar with site activities, the NRC expects that they will ensure that adequate care and attention are given to written and verbal communications with NRC. When they do become personally involved and have information made available to them (i.e., the audit table showing the April 9 start count was wrong), they should take necessary steps to ensure that inaccurate information is promptly corrected.

(2) GPC NOTIFIED THAT THE LER COVER LETTER WAS FALSE AND INCOMPLETE

Petitioners are correct that GPC was on notice that the draft LER was inaccurate and incomplete. Statements made by Mr. Mosbaugh in conversations with a number of GPC employees substantiate this claim. This fact, however, does not dictate a conclusion that GPC intended to submit false information to the NRC. Rather, it is another example of inadequate performance whereby GPC failed to correct erroneous and incomplete information.

On June 29, 1990, during a phone call with corporate personnel, and polling PRB members, on the LER Revision and transmittal letter, Messrs. Frederick, Greene, Horton, and Harry W. Majors (a corporate licensing engineer for the Vogtle Project)⁶⁷ did not fully consider and resolve the concerns raised by Mr. Mosbaugh during the polling of the PRB members that (1) the letter failed to clarify the DG starts reported on April 9, (2) DG record-keeping practices were not a cause of the difference in the DG starts reported in the April 19 LER because adequate information was available when the counting errors were made, and (3) the erroneous counts resulted from personnel errors in developing the count. Tape 187 Transcript (Staff Exh. II-18) at 2-28. Their actions played a part in GPC submitting incomplete and inaccurate information in the revised LER.

Site personnel were aware, as of June 15, 1990, that (1) Mr. Hairston was concerned about the erroneous start counts because he had attested to the information later found to be inaccurate, (2) site verification efforts had been inadequate and relied primarily on hearsay, and (3) Mr. Hairston planned to explain in the cover letter to the revised LER or elsewhere why the LER was wrong⁶⁸ and what corrective action was taken to prevent recurrence in the future. Tape 157 Transcript (Staff Exhs. II-35, II-35A) at 10-13.

⁶⁷ Mr. Majors was to complete the LER revision package and ensure that the DG start counts were consistent with the QA audit results. Filed Testimony of Harry W. Majors on Diesel Generator Reporting Issues, ff. Tr. 6212, "Majors," at 1.

⁶⁸ One of the last drafts of the cover letter to the revised LER stated that the revised LER was being submitted "to correct information related to the number of successful Diesel Generator starts subsequent to the comprehensive test program as discussed in the LER and the April 9 letter." GPC Exh. II-171T. The statement was not in the final cover letter.

Mr. Frederick, the onsite Supervisor of the SAER group, who reported to a corporate manager in Birmingham, supervised the audit conducted June 11-29, 1990, which he understood was to determine accurate numbers for the LER start counts.⁶⁹ His staff reviewed DG test data sheets generated during troubleshooting, maintenance, and surveillance testing, as well as the Unit 1 Shift Supervisor's Log kept in the control room and the Diesel Start Log (with numbered starts) maintained by the DG system engineer. Frederick at 4-5; QA Audit Report, dated June 29, 1990 (GPC Exh. II-15).⁷⁰ Unable to identify a GPC definition of "CTP," the report concluded that the CTP ended upon completion of the operability run pursuant to Vogtle surveillance procedure No. 14980. In reaching this definition, Mr. Frederick reasoned that the test program ended once the machine was declared operable. Thus, the report concluded that there were ten and twelve consecutive successful starts on DG-1A and DG-1B, respectively, as of April 19. Frederick at 6-7; GPC Exh. II-15.

Messrs. Horton, Frederick, Greene, and Majors were specifically notified about Mr. Mosbaugh's concerns regarding the accuracy and completeness of the letter, but failed to resolve them. Mr. Frederick knew the audit was narrow in scope, that the audit had not identified the specific cause of the error in the LER, and had been notified that he should examine the personal errors of Messrs. Cash and Bockhold, but unreasonably relied on his narrow-scope audit and dismissed the concerns raised by Mr. Mosbaugh.⁷¹ Mr. Horton, a voting PRB member, thought the June 29 cover letter statement about DG record-keeping practices was inaccurate because the DG Start Log was not used, but abandoned this argument when informed that Mr. Hairston drafted the language.⁷² Messrs. Majors and Greene too quickly dismissed the concern that the letter was incomplete in that it did not "clarify" the April 9 count. Further, Mr. Greene, faced with a unit down, adopted the corporate view rather than resolving the concerns of an individual who had been personally involved in the development of the LER. *See* Tape 187 Transcript (Staff Exh. II-18) at 1-28.

⁶⁹This was the stated purpose of the audit and did not implement Mr. Hairston's instruction that the reasons for the error also be determined.

⁷⁰The Petitioners assert that delays in completing the revised LER are evidence that GPC tried to mislead the NRC. There was no record evidence to support this proposition. Rather, the record revealed inept and protracted GPC efforts to arrive at updated counts and Mr. Hairston's decision to have the revision await the results of the QA audit. Completion of the audit was delayed due to difficulty in locating the pertinent records (the set in the vault was not complete and up to date) and some records (e.g., the DG Completion Sheets, which are routed through the plant mail system) were not all located until the end of the audit. Frederick at 5-6; QA Audit Report, GPC Exh. II-15 (McCoy M). Both documents were issued on June 29, 1990.

⁷¹Mr. Frederick later stated that (1) record keeping and the personal errors of Mr. Cash in making his count and Mr. Bockhold in instructing him also contributed to the error and (2) as he was unaware of Mr. Hairston's instruction for the audit to determine why mistakes were made, he had limited the root-cause determinations (e.g., inadequate training, inadequate procedures). Tr. 4270-71, 4274 (Frederick).

⁷²In his DFI response and during the hearing, Mr. Horton accepted responsibility as a PRB member for the inaccuracy in the June 29 cover letter (e.g., Tr. 5897) and admitted that he had not adequately addressed Mr. Mosbaugh's concerns (Tr. 5942).

The hearing record and DFI responses indicate that Messrs. Horton, Frederick, Greene, and Majors failed to resolve the concerns of accuracy and completeness that were raised by Mr. Mosbaugh due to a combination of factors, including the fact that (1) Mr. Mosbaugh challenged language that was personally drafted by Messrs. Hairston and McCoy, (2) Mr. Frederick held strongly to his belief based upon a narrow-scope audit that DG record-keeping caused the errors, (3) the DG record-keeping practices explanation appeared reasonable,⁷³ and (4) they believed Mr. Mosbaugh's opinions were entitled to little weight. *See* Staff Exh. II-18; Frederick at 11-12; Horton at 5-6; Majors at 4-8; Greene at 4-8; Tr. 6913 (Greene); DFI Responses: Frederick at 8-10, Horton at 2-5, Majors at 4-11, Greene at 5-13.

The actions of the individuals involved did not meet NRC expectations for ensuring that information communicated to the NRC is complete and accurate in all material respects. Their actions show a reluctance to question information developed at the corporate office (unless they had direct information to the contrary). They do not show, however, a concerted effort to mislead the NRC.⁷⁴

(3) MULTIPLE EXPLANATIONS FOR DG START-COUNT ERRORS

Petitioners claim that the various explanations regarding the DG start-count information that appeared in drafts of the cover letter to LER revision indicate that GPC endeavored to mislead the NRC. Petition at 11-12. The record shows that the drafts were part of GPC attempts to defend or explain previous DG start-count information without fully understanding what caused the errors. The allegation of intentional deception was not substantiated.

GPC's failure to resolve concerns raised about the accuracy of DG start-count information both prior to and on June 29, 1990, resulted in site and corporate personnel believing that the April 19 LER was sufficient to clarify the April 9 count as they did not realize that the numbers for, and interval of, the counts were different. GPC had not yet defined what constituted "a successful start without a problem or failure" and did not recognize that the LER Revision

⁷³ A single source document like a DG Start Log with completion sheets and numbered starts would have made the task of assembling and examining the start data easier. Aufdenkampe at 19-20; McCoy at 19-21. The hearing revealed that the updated DG Start Log (through May 2, 1990) (Staff Exh. II-22) did not record the problem during DG-1B start 136 and recorded it and starts 132 and 134 as successful starts. Tr. 4230 (Frederick) and Tr. 6879-80 (Greene).

⁷⁴ In hearing testimony and in DFI responses, GPC employees often asserted that GPC failed to meet its obligations under section 50.9 due to Mr. Mosbaugh's actions. *E.g.*, DFI Responses (Frederick at 9-10; Greene at 8, 10-13; Majors at 7-10; Bockhold at 8-9; GPC at 4-6, 12). While the enforcement action identified Mr. Mosbaugh as being among the employees who contributed to the Severity Level II problem, blaming Mr. Mosbaugh detracts from meaningful examinations of the source of GPC's errors and discourages accountability and responsibility. Also, if GPC had adequately resolved Mr. Mosbaugh's claims, in June 1990, that an examination of actions by GPC personnel was necessary to understand and correct errors, it might not have taken until August 30, 1990, to get an accurate count for April 9.

count of valid starts through June 7 did not clarify the start data presented in the April 9 letter and April 19 LER. The reliance on different types of starts for a different interval and the various explanations set forth in the drafts epitomize GPC's failure to adequately investigate the basis for the information originally conveyed on April 9 and to determine why errors were made. The use of the term "clarify" in the cover letter to the LER revision and ignorance regarding the cause of misinformation made it difficult for various GPC managers and their subordinates to provide a consistent explanation for the mistakes. The DG record-keeping explanation adopted was based on the QA audit that was not adequate to explain the causes of the count errors. The record contains no evidence of intentional efforts to deceive the NRC, but ample evidence of evolving explanations showing GPC's reluctance to admit its mistake, promptly correct the misinformation, and identify the multiple performance problems of senior GPC personnel before April 9 and April 19.

(4) SUMMARY

The record shows that (1) GPC was clearly aware, as early as May 2, that the April 9 letter and April 19 LER were incorrect and (2) GPC failed to take sufficient actions to correct the April 9 letter and determine the reasons for the errors in the two submittals. While GPC undertook efforts to correct the LER, it narrowly focused on that submittal and did not examine the actions of the individuals involved or determine whether accurate information was available from plant records.

The failure of GPC to correct the DG start counts in the April 9 letter and to provide complete reasons for the inaccurate DG start counts in the LER, was in part due to the erroneous belief that the two submittals addressed the same count information given that the April 19 start count was derived from the April 9 presentation. There is no evidence that any GPC employee knew the record-keeping statement was false or incomplete and no evidence of any deliberate efforts to conceal information from the NRC.

d. August 30, 1990 Letter

Intervenor contends that GPC deliberately (or with careless disregard) provided inaccurate or incomplete information in an August 30, 1990 letter to the NRC in an effort to "cover up" problems in developing the April 9 letter, in particular the (1) "top-down" drafting of the letter, (2) contradictory public statements by Mr. McCoy, and (3) the steering of the August 30, 1990 PRB meeting that approved the letter. Mosbaugh at 59-60, Tr. 10,394-95 (Mosbaugh); Intervenor Findings at 213-20.

In the Modified NOV, the NRC cited GPC for two instances in which inaccurate and incomplete information was provided in the August 30, 1990 letter:

The letter states that: "The confusion in the April 9th letter and the original LER appear to be the result of two factors. First, there was confusion in the distinction between a successful start and a valid test. . . . Second, an error was made by the individual who performed the count of DG starts for the NRC April 9th letter."

1. These statements are inaccurate in that confusion between a successful start and a valid test was not a cause of the error regarding DG start counts which GPC made in its April 9, 1990 letter to the NRC.

The inaccuracy was material in that it could have led the NRC to erroneously conclude that the correct root causes for the error in the April 9, 1990 letter had been identified by GPC.

2. The statements are also incomplete. While an error was made by the Unit Superintendent who performed the count of diesel starts for the April 9, 1990 letter, the root causes of the error in that letter were not completely identified by GPC. Specifically, the Vogtle Plant General Manager who directed the Unit Superintendent to perform the start count failed to issue adequate instructions as to how to perform the count and did not adequately assess the data developed by the Unit Superintendent. In addition, the Unit Superintendent did not adequately report his count to the Vogtle Plant General Manager.

The incompleteness was material in that, had the correct root causes for the error in the April 9, 1990 letter regarding DG start counts been reported, this information could have led the NRC to seek further information.

GPC contends that the inaccuracies in the letter did not result from wrongdoing on the part of any GPC employee, but acknowledges that Mr. Bockhold should have taken greater care with respect to the letter and allowed greater involvement by his staff. GPC contends that any misstatements or omissions were unintentional. *See* GPC Findings 398-400.

The NRC Staff found no evidence that showed GPC deliberately provided inaccurate and incomplete information in the letter, but found that Mr. Bockhold's actions and inactions as a senior manager contributed to the perpetuation and escalation of errors and omissions, and that Mr. Bockhold's management style rendered the performance of others ineffective. *See* Staff Exh. II-51 (cover letter) at 2-3; Staff Exh. II-49 (DFI regarding Bockhold) at 9-10.

(1) "TOP-DOWN" DRAFTING OF AUGUST 30 LETTER

During an Operational Safety Team Inspection conducted from August 6 to 17, 1990, to examine the technical validity and safety significance of the allegations submitted to the NRC, *see* Intervenor Exh. II-83, the NRC informed GPC that the June 29, 1990 submittal failed to address the April 9, 1990 data and requested that GPC clarify DG starts reported on April 9, 1990.

Mr. McCoy, aware of NRC concerns that erroneous start-count information was intentionally provided in the April 9 letter, committed, during an August 17 meeting with the NRC special inspection team, to correct the DG start data and explain the errors in the April 9 letter. Tape 258 Transcript (Staff Exh. II-19) at 1. Despite this knowledge, no root-cause evaluation or other investigation of the DG start-count errors was initiated. Instead, GPC's August 30 letter (which was drafted at corporate headquarters under the direction of Mr. McCoy and provided correct data for April 9) was dispatched without an assessment of the actions of Mr. Bockhold and Mr. Cash who developed the erroneous information contained in the April 9 letter. As a result, Mr. McCoy failed to exercise sufficient oversight and GPC again failed to identify its mistakes and take steps to ensure that the deficient conduct was not repeated.

There is no evidence to substantiate the claim that the initiation of a draft at the corporate offices was an effort to conceal information from the NRC. Site approval was sought as evidenced by Intervenor's tapes. *See, e.g.*, Tape 258 Transcript (Staff Exh. II-19). Those who were most knowledgeable (albeit somewhat uninformed) about DG start data and the causes of the error were involved in reviewing and approving the correspondence.

(2) STEERING OF PRB MEETING

The August 30 letter was the first time that GPC defined the term "successful start" and attempted to explain why the April 9 start counts were erroneous. The actions of Mr. Bockhold, the Vogtle General Manager, significantly hampered efforts to provide accurate information about why errors were made.

The PRB functions as an advisory group to the General Manager. During the August 30, 1990 PRB meeting that was reviewing a draft of the August 30 letter to the NRC, Mr. Bockhold changed the word "error" to "confusion" in the phrase explaining the reason for errors in the April 9 letter and the April 19 LER. As revised, the erroneous information was due to "the *confusion* between the distinction between a successful start and a valid test." Tape 184 Transcript (Staff Exh. II-19) at 1-3 (emphasis added). When questioned whether Mr. Cash (who had collected the April 9 DG start data) was confused about the distinction between a successful start and a valid test, Mr. Bockhold admitted that Mr. Cash was not confused when he collected the data, but claimed that the sentence

was accurate because other people were confused afterward. *Id.* at 6-8.⁷⁵ Mr. Bockhold also made several comments indicating that he wanted unanimous approval and discouraged some PRB members from suggested revised wording for the letter. Staff Exh. II-19 at 3, 9-11. His forceful, overbearing, and, at times, precipitous demeanor, (*see* Tr. 5769-76 (Aufdenkampe)) and failure to examine his own role and responsibility, contributed significantly to misinformation being provided to the NRC throughout April-August 1990.⁷⁶

Confusion after April 9 (whether by GPC or NRC personnel) could not have caused the erroneous count information provided on April 9. This example of Mr. Bockhold's forceful management style shows an environment where the PRB reviewing the draft letter could not adequately resolve a concern about the accuracy of the "confusion" statement or inquiry as to the role played by a superior in the development and reporting of misinformation on April 9. Mr. Bockhold's failure to encourage his staff to have a questioning attitude thwarted efforts to ensure the accuracy and completeness of communications with NRC. There is insufficient evidence to conclude that this defensive posture was part of efforts by Mr. Bockhold to deceive the NRC.

(3) INACCURATE PUBLIC STATEMENTS BY MR. McCoy

Intervenor asserted that because the reasons for LER errors stated in a 1990 press release by Mr. McCoy (Intervenor Exh. II-67A) (i.e., employees did not use all of the available data and used operator logs only) were different than those stated in the August 30 letter (which stated that "confusion" between a successful start and a valid test and a personnel error by the individual who performed the count caused the error) shows that GPC lacks the willingness to seek the truth. Mosbaugh at 60; Intervenor Findings at 399-400.

The mere fact that a GPC officer stated more than one reason why GPC had submitted erroneous information is not a basis for concluding that GPC was unwilling to seek the truth given what the record shows about GPC's inadequate attempts to determine why erroneous information was submitted. Inasmuch as the press release contains scattered quotes from Mr. McCoy, it is difficult

⁷⁵ Given that the QA audit report showed that there were only two valid tests (as defined by RG 1.108) on the diesel during this period (GPC Exh. II-15, Attachment B; Tr. 3279-80 (McCoy)), this was not the likely source of count errors.

⁷⁶ This incident and the PRB meeting on the FAVA system, *see* Section III.A.4, *supra*, are both examples of Mr. Bockhold's forceful management style. On April 30, 1990, senior officials of the NRC met with Messrs. McDonald, Hairston, McCoy, and others to express NRC concerns about the "cowboy" or "cavalier" attitude that Mr. Bockhold (and GPC) exhibited in dealings with the NRC. Tr. 14,850-65; Tr. 14,955-56 (Matthews). GPC and Mr. Bockhold have since acknowledged the role Mr. Bockhold's management style played in GPC communicating inaccurate and incomplete information and Mr. Bockhold has accepted responsibility for his mistakes. Letter from G. Bockhold to J. Lieberman, NRC, dated February 1, 1995. The NRC Staff also noted that GPC communications substantially improved after Mr. Shipman assumed Mr. Bockhold's position in the Fall of 1990. Tr. 15,194 (Matthews).

to determine whether any statements are quoted in context. Consequently, it is difficult to draw negative conclusions about GPC's character based on the statements.

e. OSI White Papers, Response to Section 2.206 Petition, and SSPI Data

(1) WHITE PAPERS TO NRC INSPECTION TEAM

Intervenor asserted that, during the NRC's special team inspection on operating practices and allegations (the "OSI" Inspection) conducted at the Vogtle facility in August 1990 (*see* Intervenor Exh. II-83), GPC intentionally provided false information (1) by indicating that Messrs. Cash and Bockhold sat together in Mr. Bockhold's office to work on the DG testing slide, (2) by omitting Mr. Burr from the list of individuals who wrote the April 9 letter, (3) by excluding Messrs. Hairston and McCoy from the listed participants in the April 19 phone call that added the words "subsequent to the test program," and (4) by stating that all revisions of the LER were reviewed by the PRB. Intervenor Findings at 357-376.

GPC contends that no negative inference should be drawn from any inaccuracies in the White Papers as they resulted from honest attempts to respond to questions posed by the NRC. GPC Findings 403-415.

During the August 1990 special team inspection addressing NRC concerns about GPC's operating philosophy and allegations about inaccurate information being supplied to the NRC, GPC responded to questions posed by the NRC in various "White Papers." McCoy DG at 22-23; *see* GPC Exh. II-126; Intervenor Exhs. II-131, II-95.

There is no evidence to support the claim that the inaccuracies in the documents resulted from deliberate efforts to mislead the NRC and conceal the participation of senior GPC officials. As is evident from the discussion on the Tape 253 Transcript (GPC Exh. II-122; Intervenor Exh. II-148), the recollections of various GPC employees were cloudy as to who participated in decision-making and who prepared documents. GPC employees freely stated their opinions as to who participated in various decisions and there was nothing to put GPC on notice that the information to be submitted was inaccurate. In addition, the White Paper expressly conveyed "GPC's belief" at the time when (based upon information developed during the licensing hearing and enforcement proceeding) GPC's investigation of issues was incomplete. Thus, there is no indication that the mistakes were intentional.

(2) STATEMENTS IN RESPONSE TO SECTION 2.206 PETITION

Intervenor also contends that GPC intentionally tried to conceal Mr. Hairston's participation in the April 19 call regarding the LER when Mr. McDonald signed GPC's response to the section 2.206 petition and later clarifications.

There is insufficient evidence to show that GPC intentionally provided inaccurate information. There is no evidence that Mr. McDonald was specifically aware of Mr. Hairston's participation on the April 19 call and Tape 58 (GPC Exh. II-2) shows that Mr. Hairston joined the call after the wording regarding the Comprehensive Test Program was added and did not participate in "Call B" when Messrs. Shipman, Aufdenkampe, and Mosbaugh finalized the LER language. *See* Tape 58 Transcript (GPC Exh. II-2; Staff Exh. II-45 (Vogle Coordinating Group Report)). The failure to identify various participants on the calls indicates faulty recollection of GPC employees (shown to be inaccurate by the Intervenor's recordings) and is among the numerous mistakes GPC made in providing information on the DG issue. Performance failures, not deception, appear to be the likely cause.

(3) SSPI DATA

Intervenor asserts that GPC's failure to include "bad" 1990 Safety System Performance Indicator (SSPI) data in the April 9, 1990 letter to the NRC and to give such data to the IIT is evidence of a pattern of willfulness by GPC and argues that the data should have been included in the April 9, 1990 letter. Intervenor Findings 44-73; Mosbaugh at 99-104; Tr. 10,369 (Mosbaugh). GPC contends that exclusion of the 1990 data, which was based upon only a few months rather than a full year, did not represent a relevant and material omission concerning the Vogtle DGs. GPC Findings at 191-98.

The fact that the data were not included in the final version of the April 9 letter is not significant. The record shows that the NRC asked GPC to address the reliability of the DGs as part of the April 9 presentation. The SSPI data given to the IIT addressed the years 1987, 1988, and 1989 and was incomplete for 1990. Intervenor Exhs. II-89, II-91.

In a conversation taped by Mr. Mosbaugh on or about April 2, 1990, Mr. Bockhold discussed with Mr. Mosbaugh a document containing SSPI data for Vogtle DGs and indicated the data were to be given to the IIT and Mr. Brockman of the NRC. Mosbaugh at 101; Intervenor Exh. II-89. Contrary to Intervenor's assertion that it was hidden from the IIT, a document containing the SSPI data was among the documents collected by the IIT after the SAE. *See* IIT Document No. 143 (Intervenor Exh. II-89).

Intervenor's allegation that a draft of GPC's April 9, 1990 letter that contained the SSPI data was telecopied to the GPC corporate office and the NRC was not

proven. NRC Staff records show that draft information transmitted to Messrs. Brockman (Region II) and Matthews (NRC Headquarters) on April 5 and 6, 1990, did not contain the data. *See* Intervenor Exhs. II-65, II-65A; *see* Tr. 3287-90.

The NRC's interest relative to restart was to understand the basis for GPC's position that the DGs were operable and that GPC's corrective actions had been effective. The NRC was not seeking a numerical value like SSPI (which represents the time that a given unit, on average, annually is unavailable), either historically or currently, as part of its restart decision and does not normally rely on such data.⁷⁷ *See* NRC Staff's Reply to Intervenor's First Set of Interrogatories, dated September 15, 1993, at Interrogatory 11.

There is no basis to conclude that the data should have been included in the April 9 letter in order to address the NRC's inquiry about DG reliability and operability. Mr. Bockhold's decision not to include the data for the first few months of 1990 was not unreasonable. Intervenor has not shown that the information was necessary for a decision on whether the short-term corrective actions were sufficient to provide reasonable assurance to permit restart, and it is clear that the information was made available to the NRC.

(4) CONCLUSIONS REGARDING WHITE PAPERS, SECTION 2.206 RESPONSE AND SSPI DATA

There is no evidence to support Intervenor's assertion that GPC knowingly submitted false information regarding Mr. Hairston's participation on the April 19 call about the LER. The misstatements are readily explained by faulty recollection, and do not indicate that GPC intentionally misrepresented Mr. Hairston's participation. The audio recording made on that date shows that he was not a significant participant in discussions about the accuracy of the LER.

Similarly, there is no basis to conclude that Mr. Bockhold was deceitful in failing to include Safety System Performance Indicator Data in the April 9 letter in that the information, although incomplete, was provided to the IIT. There is no evidence that the information omitted was requested by the NRC or reasonably should have been included in the letter.

⁷⁷The Vogtle TSs address DG reliability by requiring increased frequency of DG testing if a specified number of failures occurred during the last 20 or 100 valid tests. The TSs also require special reporting of DG test results. These requirements of the TSs are totally unrelated to SSPI data. SSPI data for individual DGs are calculated by dividing the unavailable hours (planned, unplanned, and estimated) by the total number of hours the DG is required to be operational during the SSPI assessment period. GPC Exh. II-140. Such data have little or no value with respect to DG operability and the effectiveness of corrective actions to allow restart.

f. Statements Concerning Air Quality in the April 9 Letter and to the IIT

(1) INTRODUCTION

A sufficient air supply is needed both to start the diesel engine and to operate the engine controls. This air is supplied to each diesel engine by an independent, redundant starting air system that includes an air compressor, an after-cooler, a refrigerant air dryer,⁷⁸ an air receiver, intake air filters, starting valves, air distributors, instrumentation, controls, alarms, and the associated piping to connect the equipment. Alarms annunciate on the local control panel in the diesel building and in the Unit's main control room to enable operators to monitor the DG starting air system. Vogtle SER § 9.5.6 (Board Exh. II-4) at 9-68.

The control air is supplied by the starting air system from a point downstream from the air receivers. Control air is used by the pneumatic logic components and sensors to control and protect the diesel engine. The control air passes through a 5-micron filter and then through a pressure regulator that maintains control air pressure at 60 psig. *See* NUREG-1410, at 3-47 (Intervenor Exh. II-10).

One of the ways of monitoring the quality of DG starting air is through dewpoint measurements taken by attaching the dewpoint testing equipment at a pressure gauge fitting on the air receiver. The temperature range of acceptable dewpoints at the Vogtle facility is 32-50°F. Dewpoint measurements obtained at the Vogtle facility on the DG air system are documented in Maintenance Work Orders (MWOs), which are used to perform the Preventive Maintenance (PM) checks of the DG air dewpoints. *See* Intervenor Exh. II-78, at 5-10; *see* Mosbaugh at 69-70; Intervenor Exh. II-169.

The April 9, 1990 letter submitted to the NRC to support GPC's request for restart stated the following with respect to air quality:

GPC has reviewed air quality of the D/G air system including dew point control and has concluded that air quality is satisfactory. Initial reports of higher than expected dew points were later attributed to faulty instrumentation. This was confirmed by internal inspection of one air receiver on April 6, 1990, the periodic replacement of the control air filters last done in March 1990 which showed no indication of corrosion and daily air receiver blowdowns with no significant water discharge.

⁷⁸The air dryer at Vogtle is located upstream of the air receiver; the dryer removes water vapor from the compressed air before the air reaches the receiver and is designed to run continuously. FSAR § 9.5.6 at 9.5.6-4 (Board Exh. II-3); Board Exh. II-4 at 9-68. Compressed ambient air, saturated with water vapor, enters the dryer and is precooled by the outgoing refrigerated air by an air-to-air heat exchanger. The precooled air then enters the air-to-refrigerant heat exchanger (i.e., the refrigeration evaporator) where it is cooled by the dryer's refrigeration system. As the air cools, water vapor condenses into liquid droplets which are separated out of the air stream by a moisture separator, and automatically discharged by a draintrap. Board Exh. II-3 at 9.5.6-4.

GPC Exh. II-13, at 3. On May 9, 1994, the NRC issued the NOV to GPC, which included a Violation B on air quality based on (1) GPC's failure to provide complete information regarding control of DG air quality (i.e., dewpoints) in the April 9, 1990 letter by only stating that initial reports of high dewpoints were attributed to faulty instrumentation and (2) GPC's failure to state that high dewpoints for Vogtle Unit 1 were also attributable to system air dryers occasionally being out of service for extended periods and to system repressurization following maintenance. Staff DG Panel at 7; Staff Exh. II-46, at 3-4.

After reviewing GPC's response to the NOV, the NRC Staff concluded that as of April 9, 1990, GPC had an adequate technical basis to support a finding that air quality was acceptable, and that dewpoint information of a historical nature, i.e., from before the SAE, was not necessary for the April 12, 1990 restart decision. Staff Exh. II-50, at 5-6; *see also* Staff DG Panel at 9. In the Modified NOV, dated February 13, 1995, the NRC withdrew Violation B. Staff Exh. II-51, Appendix at 2-3.

Intervenor asserted that the air quality statement in the April 9 letter is materially false and deliberately misleading in that (1) high dewpoints were not due to "faulty instrumentation" (Intervenor Findings at 285) and (2) the results of the April 6, 1990 inspection of the air receiver, the inspection of air filters, and the daily air receiver blowdowns did not support a conclusion that air quality was satisfactory (Intervenor Findings at 306-09).⁷⁹ *See also* Petition at 9. Intervenor also alleged that GPC was recklessly careless in communications regarding high dewpoints and concealed high dewpoint readings from the IIT. *See* Mosbaugh at 66-92.

GPC maintains that the letter conveyed its judgment that, as of April 9, 1990, the diesel control air quality relative to moisture or humidity was satisfactory based upon the April 6 air receiver inspection and the daily air

⁷⁹ Intervenor also alleged that water was collected from the diesel air system prior to April 9, 1990, in that (1) he saw a jar of 8 ounces of yellowish fluid in Mr. Kochery's office on March 30, 1990; and (2) a taped (and partially inaudible) conversation indicates that the water came from diesel pneumatic tubing (air system "trip lines") that were disassembled on March 29. Mosbaugh at 93-94.

DG vendor representatives who were present during the March-April 1990 disassembly of most of the diesel sensing lines and performed the diesel logic functional testing, including the disconnection of all protective trip lines within the engine control panel, did not recall observing or hearing about any water or moisture problems in the diesel starting or control air in March-April 1990. Rebuttal Testimony of Sheldon OwYoung and Robert Johnston on Air Quality Statements, ff. Tr. 12,428, "OwYoung-Johnston," at 4-5; Tr. 12,741, 12,758-59 (OwYoung, Johnston). Others present in Mr. Kochery's office had no recollection of the incident and even disputed Mr. Mosbaugh's transcribed version of the March 30, 1990 tape segment. Tr. 7552-53, 7568-70 (Stokes); Chenault Rebuttal at 3-4 (ff. Tr. 14,020); *see also* Tr. 14,071-73, 14,076 (Chenault).

In addition, in May 1994, the NRC Staff inspectors examined whether water had been in the diesel control air system in 1990. The Staff identified numerous examples of out-of-specification dewpoints, but found no evidence of actual water formation in the diesel control air system lines or corrosion. Staff Exh. II-5, at 1, 6-8; *see also* Testimony of Edward B. Tomlinson and Pierce H. Skinner on Air Quality, ff. Tr. 14,497, "Staff AQ," at 10-11. Thus, there is no evidence to substantiate the claim that water was in the trip line.

receiver blowdowns, which did not indicate a high-humidity environment in the starting air system. *See* GPC Exh. II-55A (Tape 41 Transcript), at 2; Supplemental Testimony of George Bockhold on Air Quality Statements, ff. Tr. 6397, "Bockhold AQ," at 5-6. The statement that "initial reports of higher than expected dewpoints" was not intended to describe all past maintenance issues or to refer to any dewpoint readings taken after March 29, 1990. *Id.*; Tr. 6582 (Bockhold).

The NRC Staff concluded, based on the hearing record, that (1) the air quality portion of GPC's April 9 letter was incomplete in that it did not reference the fact that the Instrumentation and Control (I&C) technicians were unfamiliar with the use of the VP-1114 instrument, and initially misused it, in taking dewpoint measurements in early April 1990; and (2) the reference in the April 9 letter to "initial reports" should reasonably include high dewpoint measurements taken prior to April 9. Tr. 14,756-57, 15,111 (Matthews). The NRC Staff found that out-of-specification dewpoint readings identified by Intervenor during the hearing (Intervenor Exh. II-169) did not show that air quality was unsatisfactory since inspection of the receivers and controls showed no evidence of corrosion or a long-term water problem. Tomlinson and Skinner at 12-13.

(2) ACCURACY OF STATEMENT THAT AIR QUALITY WAS SATISFACTORY

Mr. Mosbaugh alleges that corrosion seen during the April 1990 inspection of an air receiver is evidence that air quality was not satisfactory, as stated in the April 9 letter. *See* Mosbaugh at 82-83.

One DG-1A air receiver tank (K02) was inspected by GPC and NRC Staff representatives on April 6, 1990. *See* Affidavit of Milton D. Hunt, dated March 1, 1995, ff. Tr. 4882, "Hunt Affidavit," at 5; Prefiled Testimony of Kenneth Stokes on Diesel Generator Air Quality Statements, ff. Tr. 6962, "Stokes," at 2-3; Rebuttal Testimony of Harvey Handfinger, ff. Tr. 11,346, "Handfinger," at 2; Tr. 11,450-56 (Handfinger).⁸⁰ The metal was clearly visible inside the receiver and there were no loose rust particles in the tank,⁸¹ water droplets on the tank walls, or other signs of moisture during the inspection. Tr. 11,374, 11,450-56, 11,483. The fact that there were normal rust spots on the welds inside the tank and that the control system air filters appeared "new" also indicated that air quality was not a problem. Hunt Affidavit at 5-6; Tr. 4930 (Hunt).

⁸⁰ Mr. Harvey Handfinger was GPC's Manager of Maintenance, reporting to Mr. Kitchens (Assistant General Manager-Operations). Mr. Mark Briney, as the acting Instrumentation and Control Superintendent, reported to Mr. Handfinger.

⁸¹ Mr. Shipman's April 11, 1990 notes (GPC Exh. II-147) showed that there was minor "flash" corrosion or rust observed on the weld seams of the air receiver tank, as expected, given that welded joints on the carbon steel tank form a thin "rust" or corrosion film immediately after welding. Rebuttal Testimony of William B. Shipman (ff. Tr. 10,890), "Shipman Rebuttal," at 14.

Mr. Mosbaugh did not dispute the statement from the April 9 letter that air quality was satisfactory when that statement was read to the IIT on April 11, 1990; and Messrs. Kochery and Stokes indicated that the statement was correct, even though the 50-degree dewpoint requirement had not always been met. Tape 41 Transcript (Staff Exh. II-15) at 1-2, 5-7.

Based on the evidence set forth above, particularly the absence of significant rust, corrosion, or moisture,⁸² the statement in the April 9, 1990 letter that air quality was satisfactory was not inaccurate.

(3) INCOMPLETE REASONS FOR HIGH DEWPOINT READINGS

The record shows that GPC's statement that "initial reports of higher than expected dewpoints were later attributed to faulty instrumentation" is incomplete in that it failed to indicate that high readings were also obtained after the SAE due to technicians being unfamiliar with backup equipment, but there is no basis to conclude that GPC intended to mislead the NRC.

On March 9, 1990, there were out-of-specification dewpoint readings of 61°F and 66°F taken on DG-1A air receivers K01 and K02, respectively. GPC believed the high readings were valid since humidity would have risen while DG-1A was out of service and disassembled from March 1 to March 13, 1990, for overhaul maintenance and testing in that the receivers had been depressurized and opened to the room atmosphere. Prefiled Testimony of Lewis A. Ward on Air Quality Statements, ff. Tr. 7740, "Ward AQ," at 3-4; Tr. 7878-80 (Ward).⁸³ After overhaul maintenance, air receivers are recharged using multiple "bleed-and-feed" cycles, as necessary, until the dewpoint is within the acceptable range. The dewpoint readings were within specification on March 12, 1990, and the DG was declared operable on March 13, 1990. Ward AQ at 4; GPC Exh. II-62.

On March 28, 1990, air quality, including the possibility of small debris or moisture in the diesel air system, was discussed at a meeting with the IIT where GPC stated it would determine the last recorded dewpoints for DG-1A and take another dewpoint reading in an effort to identify the cause of the March 20, 1990

⁸² If any water had ever formed in the pneumatic control air system, water would likely accumulate in the bowl of the control air filter in the diesel engine control panel, but there was no evidence of water in that filter before or during the March 1990 outage. OwYoung-Johnston at 5-6; *see also* Tr. 12,495-502 (OwYoung, Johnston). Moisture corrosion problems in the diesel air start system in 1990 would have also caused degradation due to corrosion or corrosion products which would have been obvious during the inspection and testing of the diesels, but there was no evidence of corrosion during the inspection and testing of the diesels following the SAE. Testimony of Kenneth Stokes on Air Quality Statements, ff. Tr. 6962, "Stokes," at 4. For example, the logic board, which was removed and replaced subsequent to the DG-1B start on March 24, 1990 (Start 136), showed no signs of a water or moisture problem and inspections during each 18-month replacement of the control air filters revealed no moisture problem. Tr. 7704, 7685-86 (Stokes).

⁸³ Mr. Ward attributed the high readings to an actual high-humidity condition as a result of DG-1A, including its air start system, being out of service and disassembled from March 1 to March 13, 1990, for overhaul maintenance and testing. Ward AQ at 4.

spurious trips on DG-1A. *See* GPC Exh. II-49 (IIT Transcript), at 95-96; *see also* Bockhold AQ at 1. GPC Instrumentation and Control (I&C) technicians performed the monthly preventative maintenance dewpoint check on DG-1A on March 29, 1990, recorded out-of-specification high readings of 80°F and 60°F, and documented them on an MWO for evaluation and trending purposes.⁸⁴ *See* GPC Exh. II-155, at 1; MWO 1-90-01513 (GPC Exh. II-155); Rebuttal Testimony of Mark Briney on Diesel Generator Reporting Statements, ff. Tr. 12,075, "Briney," at 5; MWO 1-90-01651, dated March 30, 1990 (Intervenor Exh. II-143).

During an April 3, 1990 telephone conference with IIT and Region II personnel, GPC (Mr. Bockhold) stated that the air quality was satisfactory, but did not mention dewpoint readings. *See* GPC Exh. II-50 (IIT Transcript) at 59-60; *see also* Bockhold AQ at 2. Mr. Bockhold testified that he was not aware of the March 29 high reading on that date and probably focused on the clean condition of the air filters.⁸⁵

On April 5, 1990, GPC initiated a blowdown on the DG-1A air receivers to check for the presence of moisture, a feed-and-bleed of the DG-1A air receivers to lower the dewpoint, and a check of all the diesel control system air filters for the presence of moisture. *See* Briney at 5-6; GPC Exh. II-156. Dewpoint readings of 84°F and 82°F were obtained on DG-1B. *See* GPC Exh. II-156, at 1; Intervenor Exh. II-169, at 3.

On April 5 through April 6, a series of high dewpoint readings on DG-1A was obtained using the Alnor VP-2466 dewpoint instrument. *See* Intervenor Exh. II-143 at continuation sheets 1-3; Intervenor Exh. II-169, at 2. On April 6, Mr. Bockhold informed the IIT that he was aware (on April 5) that there were high dewpoint readings for the DG-1A on March 29,⁸⁶ and that GPC

⁸⁴ Intervenor's allegation that the March 29 rejection of a Deficiency Card shows that GPC intended to conceal the high dewpoints readings from NRC (Intervenor Findings 605-606), was not substantiated. The problem was adequately documented by means of a Maintenance Work Order, an act inconsistent with an intent to keep information from the NRC.

⁸⁵ Mr. Bockhold admitted that some of his responses to the IIT that day may, in retrospect, have been misleading. Tr. 6460-63, 6507-08 (Bockhold).

Intervenor's allegation that Mr. Bockhold was made aware of the March 29, 1990 high readings on or about March 29, and that he deliberately withheld this information from the IIT during an April 3, 1990 teleconference (*see* Intervenor Findings 533, 536), however, was not supported by the evidence. Mr. Bockhold could not recall being aware of the high readings prior to April 5. *See* IIT Transcript (GPC Exh. II-51) at 1, 4-5; Tr. 6566 (Bockhold). Messrs. Hunt and Bockhold understood that dewpoints above 32-50°F were not of immediate concern for operability of the diesels but could cause parts in the diesel air system to corrode if they occurred over the long term. Tr. 4898-99 (Hunt); GPC Exh. II-51, at 6-8; Tr. 6466-67, 6558-59, 6608-09 (Bockhold). There is no evidence that any GPC employee, including Mr. Mosbaugh, believed the diesels were inoperable due to poor air quality or shared such a view with Mr. Bockhold. *See* Tr. 6697 (Bockhold). Therefore, there is insufficient evidence to conclude that, by April 3, 1990, Mr. Bockhold knew about the March 29 dewpoint readings or withheld that information from the NRC's IIT.

⁸⁶ An NRC Region II inspector, Milton Hunt, reviewed MWOs on the diesels and discovered the March 29, 1990 high dewpoint readings on DG-1A air receivers in early April 1990 and informed GPC. *See* Hunt Affidavit at 5; Tr. 6566 (Bockhold).

(Continued)

thought the dewpoint sensor instrument was bad and was trying to obtain a backup instrument.⁸⁷ See GPC Exh. II-51 (IIT Transcript) at 1, 4-5. On the afternoon of April 6, following the series of high readings on DG-1A, GPC tried to determine whether there was an actual high dewpoint condition or faulty instrumentation and used a backup EG&G dewpoint instrument (VP-1114) to verify the accuracy of the Alnor VP-2466 readings on DG-1A. Tr. 12,081-82 (Briney). See Intervenor Exh. II-143 at continuation sheet 3; Intervenor Exh. II-169, at 2. The vendor's instruction manual for the VP-1114, however, could not be located and the I&C technicians taking the measurements lacked training on the VP-1114.⁸⁸ *Id.* at 12,082-83; see also Tr. 12,784 (Hammond).

On April 7, 1990, an I&C technician took dewpoint measurements on the Unit 1 and Unit 2 air receivers using three different instruments — the Alnor VP-2466, the EG&G VP-1114, and the recently acquired General Electric (G.E.) rental Alnor Model 7000. The VP-2466 and VP-1114 readings were out-of-specification high while the G.E. rental instrument readings were out-of-specification low. See Intervenor Exh. II-217, at 3; see also Intervenor Exh. II-169.

GPC's acting I&C Superintendent could not draw any definitive conclusions from the out-of-specification dewpoint results obtained on April 6-7, but was convinced that eight independent air systems would not simultaneously fail to provide satisfactory air to the receivers. Briney at 7-8; see also Tr. 6554-55 (Bockhold).⁸⁹

The then NRC inspector believed that he saw a listing of dewpoint readings taken April 6-7 before he left the site on April 7, 1990,⁹⁰ and was aware

Intervenor alleged during the hearing that GPC, in its 1994 NOV response and in the 1995 prefiled air quality testimony of Mr. Bockhold, intentionally falsely asserted that GPC self-reported the March 29 high dewpoint readings to the NRC (Intervenor Findings 537-538, 540-541). While Mr. Hunt's subsequent testimony shows GPC's statement to be in error since Mr. Hunt testified he discovered the March 29 high dewpoint readings, no evidence was presented to substantiate the claim that the error was intentional.

⁸⁷ Although Intervenor is correct that Mr. Bockhold's April 6, 1990 statements to the IIT that there was not a backup dewpoint analyzer at the plant was inaccurate (Intervenor Findings 543-544), there is no basis to conclude that the statement was intentionally false, particularly since the backup instrument (VP-1114) was used subsequent to the telephone conference with the IIT.

⁸⁸ Intervenor established at the hearing that the EG&G Model 911 instrument had been used by I&C technicians on one occasion in March 1989. See Tr. 12,216-17 (MWO 18900822 reflects dewpoint readings taken by I&C technician using an EG&G instrument).

⁸⁹ Intervenor's allegation that GPC engaged in intentional willful conduct in claiming that the VP-2466 dewpoint instrument was defective (Intervenor Findings 547-550, 578-579, 583, and 604) was not substantiated. By April 6, 1990, GPC had a reasonable basis to suspect the Alnor VP-2466 instrument was faulty in that (1) the extended calibration due date for the instrument was about to expire; (2) the last in-specification reading was on March 29, 1990, for DG-1B; and (3) all of the April 5 dewpoint readings on the DG-1A and DG-1B using the VP-2466 instrument were out-of-specification high. See GPC Exh. II-159; see also Briney at 13; see also Exh. II-169, at 2-3.

⁹⁰ Mr. Bockhold had provided a list of high dewpoint measurements to Mr. Hunt, and Mr. Hunt suggested that GPC borrow dewpoint test equipment from the V.C. Summer Nuclear Plant in order to accurately measure dewpoint readings and verify the condition of the air. Tr. 6537, 6563 (Bockhold); Hunt Affidavit at 5; Tr. 4924-25, 4935 (Hunt).

of GPC's opinion that the high readings were due either to faulty dewpoint equipment or operator error. Affidavit of Milton D. Hunt, ff. Tr. 4882, "Hunt Affidavit," at 5; Tr. 4924-25, 4930-31, 4933-36 (Hunt). See GPC Exh. II-52.

GPC later determined (based on an EG&G instrument borrowed from V.C. Summer around April 8 with its instruction manual) that the initial readings taken with the VP-1114 instrument had been used improperly (without the required flow meter) on April 6-7, 1990.⁹¹ Briney at 8-9; Tr. 12,088, 12,340 (Briney); Tr. 6513 (Bockhold); Intervenor Exh. II-169. By April 8, 1990, readings on both units that were taken using the flow meter (VP-1114 and FS-3529) were in specification (and in close agreement) except readings on the DG-2A K02 receiver (where the dryer was found to be turned off).⁹² GPC concluded that the prior Alnor readings from the VP-2466 instrument were not valid. Tr. 12,166 (Briney), 12,857-59 (Hammond). See Briney at 9; Tr. 12,203, 12,206 (Briney); Intervenor Exh. II-169.

During a morning conference call on April 9, 1990, Mr. Lewis A. Ward, Manager of Nuclear Maintenance and Support located in the corporate office, told the IIT that with the borrowed instrument, all of the April 8 dewpoint readings were within specification. See GPC Exh. II-61, at 4. Mr. Skip Kitchens, Assistant Plant General Manager—Operations, then stated that a high DG-2A dewpoint reading believed to be caused by an air dryer being inadvertently turned off (probably on April 6) was being addressed by blowing down the air receiver. There was no mention of I&C technician errors.⁹³ See IIT Transcript (GPC Exh. II-61) at 4-8. In response to an IIT request, GPC committed to provide a history of dewpoint data for the past year. *Id.* at 7-9.

During the April 9 meeting with the NRC in Atlanta, the NRC was told that air quality was good, that high readings were attributed to a faulty dewpoint instrument, and that an April 6 inspection of an air receiver, as well as inspections of the control air filters and daily air receiver blowdowns, confirmed that air quality was acceptable. Intervenor Exh. II-71, Project No. 006214.

During the April 11 teleconference with the IIT, Mr. Bockhold (referencing the table of dewpoint measurements dating back to March 1989 that had been prepared to address the NRC's request for data) stated that air quality had been and remained satisfactory for a number of reasons, including the April 6 air receiver inspection, which showed only light corrosion around the welds and

⁹¹ During the hearing, an NRC Staff witness, Mr. Pierce Skinner, contacted an EG&G representative who told him that it would have been extremely difficult for an I&C technician to throttle flow to the correct level without a flow meter. Tr. 14,644-45 (Skinner). Incorrect flow causes errors in dewpoint readings. *Id.*

⁹² Intervenor's claim that all eight air receivers had experienced high, out-of-specification dewpoints due to personnel inadvertently or intentionally turning off the air dryers (Intervenor Finding 581) was not substantiated. Intervenor provided no evidence to support his claim and Mr. Hunt recalled that the dryers were out of service or off only a few times. Tr. 5008-10 (Hunt).

⁹³ The notes of Mr. Bailey, taken during GPC's April 9, 1990 meeting with NRC in Atlanta, also reflect that this high dewpoint reading was reported to the NRC. See Intervenor Exh. II-70, at 5.

a minor amount of oil on the bottom. *See* GPC Exh. II-56, at 6-7; Rebuttal Testimony of W.F. Kitchens, ff. Tr. 13,590, "Kitchens," at 9; *see also* GPC Exh. II-56, at 2. The data provided to the IIT (GPC Exh. II-57) did not include the high dewpoint readings from April 5-7, 1990, because GPC did not believe the readings were accurate or reliable. Kitchens at 9.⁹⁴

(4) CONCLUSIONS

The April 9 letter was incomplete, as it did not indicate that high readings were also caused by technicians being unfamiliar with a dewpoint instrument. By April 9, 1990, senior GPC management at the Vogtle facility (Messrs. Bockhold and Kitchens) and in Birmingham, Alabama (Mr. Ward), knew about the problems the I&C technicians had in using the VP-1114 instrument correctly. While the letter's reference to "initial reports" is ambiguous, all high dewpoint measurements taken near the time of the SAE and prior to April 9 could have influenced an NRC decision on restart.

The evidence does not establish that GPC acted with reckless disregard for the truth, intentionally misrepresented information, or conspired to mislead the NRC in communications regarding DG air quality. GPC took reasonable steps to determine air quality (including the receiver inspection), performed blowdowns on the air receivers to remove any moisture that could affect DG performance, and generally kept the NRC informed about their activities. While GPC provided incomplete information about the causes of high dewpoint readings based on the belief that recent out-of-specification readings were not valid, and there may have been some delays in sharing information about dewpoints with the NRC, the evidence considered as a whole falls short of demonstrating that GPC engaged in making willful or recklessly careless misrepresentations, and does not otherwise show that GPC lacks the requisite character and integrity to operate a nuclear plant.

g. Conclusions Regarding Diesel Generator Statements

Petitioners allege that GPC, deliberately or with careless disregard, submitted false and misleading information regarding DG starts (1) in an April 9, 1990 presentation and letter to the NRC (seeking permission to restart after the SAE); (2) in an April 19, 1990 Licensee Event Report (LER) 90-006 on the SAE by

⁹⁴Intervenor's allegation that GPC intentionally concealed the VP-1114 "confirmatory readings" (Intervenor Findings 555-565, 575, 590-596) was not substantiated. VP-1114 readings were among those given to the NRC, but questions about the accuracy of those readings were resolved by FS-3529 readings taken on April 8. Given that the NRC was interested in dewpoint readings (and not necessarily the particular equipment used to obtain them) and that VP-1114 readings were included on the listing provided to the IIT, there is insufficient evidence to support Intervenor's claim. *See* GPC Exh. II-51, at 7-8; GPC Exh. II-57; Intervenor Exh. II-169, at 2.

means of a conspiracy among GPC managers; (3) in a June 29, 1990 cover letter forwarding the revised LER; and (4) in an August 30, 1990 letter. Petitioner also alleges that GPC knowingly submitted false or misleading statements (1) concerning DG air quality in the April 9, 1990 letter (and in contemporaneous discussions with the NRC's IIT); and (2) in GPC's April 1, 1991 response to Intervenor's section 2.206 petition with respect to Mr. Hairston's involvement in developing the false start information (i.e., during an April 19 call) and when GPC managers became aware of inaccurate start counts. These claims were not proven.

Although Petitioners are correct that misinformation was provided to the NRC in various communications related to DGs, the weight of evidence fails to show that GPC knew the information was false or incomplete. The repeated failure of GPC to provide accurate and complete information relating to the count of DG starts in April 1990 stemmed from GPC performance failures that do not amount to deliberate efforts to deceive or mislead the NRC or to avoid regulatory requirements. The erroneous counts of eighteen and nineteen consecutive successful starts without problems or failures for DG-1A and DG-1B, respectively, as of April 9 (instead of twenty-nine and twelve) were caused by GPC's use of ambiguous terminology to show diesel reliability during a poorly defined period. When questions arose about the accuracy of the data, GPC managers relied primarily on verbal assurances that defended the information and revised the count description without (1) examining the causes of the initial misstatements, (2) determining accountability, and (3) promptly correcting erroneous information that was presented to the NRC. The reliance on verbal assurances and incomplete site verification efforts on April 19 did little to address or identify mistakes by the General Manager in requesting and presenting the start count, and the Unit Superintendent in reporting the start data he collected. Consequently, the count reported included problems or failures and was not a count after the CTP (which GPC later determined commenced with the surveillance test where a DG is declared operable).

There was no evidence that any of the current GPC or Southern Nuclear personnel who were involved (Messrs. Bockhold, Cash, Shipman, Aufdenkampe, McCoy, Hairston, Frederick, Greene, Horton, Majors, Kitchens, and Ward) conspired, or acted individually, to submit information they knew to be false from March 20 through August 30, 1990, regarding DG testing or air quality. Clearly, these statements reflect only a portion of the many exchanges between the NRC and GPC concerning efforts to determine the causes of the SAE. The failure of GPC personnel, individually and collectively, to take steps to ensure that the NRC was provided with complete and accurate information during this period nonetheless is a very significant regulatory concern that constituted a Severity Level II problem at the facility — conduct far below NRC expectations.

Based on observations of NRC Headquarters and Regional inspection staff throughout April through August 1990, GPC took sufficient actions to ensure that the DGs were reliable and operable. GPC's performance fell short, however, with respect to the level of importance and diligence afforded some communications to the NRC *and* prompt resolution of concerns about the accuracy and completeness of information provided to the NRC. This sometimes "cavalier" GPC attitude led GPC to fix the words (rather than to verify and reverify facts) in communications to the NRC. Mr. Bockhold's management style contributed to an atmosphere whereby site employees were reluctant to question the accuracy or completeness of communications to the NRC, unless they specifically knew that the information was wrong.

It is unreasonable that it would take over 4 months (until August 1990) to get an accurate start count for April 9 and take 4 years (until GPC's 1994 NOV Response) to understand why errors were made. Nevertheless, GPC now recognizes its role in providing incomplete and inaccurate information to the NRC and its failure to take steps to ensure communications that satisfy the requirements of section 50.9.⁹⁵ GPC site and corporate managers and GPC employees (including members of the PRB) have accepted responsibility for the mistakes made in 1990 as indicated in responses to the NOV and Demands for Information, and in testimony during the hearing. GPC no longer asserts that Mr. Mosbaugh and Mr. Cash, alone, are responsible for incorrect DG start counts.

In the end, whether the start counts were twenty-nine and twelve (instead of the eighteen and nineteen reported on April 9), or whether all causes of high dewpoint readings were reported, did not affect the soundness of the decision that the DGs were ready to perform their function. The incomplete and inaccurate information was material in that it had the ability to influence the NRC in its dealings with GPC. Correct and complete information may have led the NRC to inquire further before authorizing restart in April 1990

⁹⁵ Corrective action taken by GPC management in response to the NOV included: (1) making the NOV available to all employees and committing to post an NRC Order if one were to be issued; (2) emphasizing the importance of thorough record keeping during off-normal hours in a letter from GPC's Senior Vice President-Nuclear Operations to the Vice Presidents for the Hatch and Vogtle facilities; (3) stressing the importance of effective communications and the effective resolution of concerns in letters from the Executive Vice President-Nuclear Operations to nuclear operations employees; (4) posting copies of section 50.9 for all employees to read; (5) discussing GPC's policy of open, complete, and accurate communications with the NRC in meetings between the Senior Vice President-Nuclear Operations and employees at the Hatch and Vogtle sites, and distributing letters to all employees on the same subject; (6) observing communications with the NRC to ensure that the enforcement action does not adversely affect the completeness of statements; (7) making GPC's reply to the NOV available for all GPC employees to read; (8) counseling the Unit Superintendent and Vogtle General Manager by GPC's Senior Vice President-Nuclear Operations; (9) issuing an "Oral Reminder" to the Unit Superintendent pursuant to the Positive Discipline System; and (10) prohibiting the 1990 Vogtle General Manager from resuming a line management position with GPC or Southern Nuclear nuclear facilities through February 1, 1998, pending completion of personal training and 60 days prior notice to the NRC. See GPC Reply to NOV and DFIs, dated July 31, 1994, as supplemented February 1, 1995.

and about GPC operations, in general, at Vogtle. These events illustrate the need for improvement in communications, both within GPC and with the NRC, and the need for Licensee personnel to maintain a questioning attitude about explanations and data provided to the NRC.

The repeated involvement of Mr. Bockhold in GPC's submission of incomplete and inaccurate information to the NRC is significant. Mr. Bockhold ably handled technical issues, but his sometimes overbearing and forceful management style, his reliance on rewrites rather than reverifications, and his failure to examine his own inadequate performance contributed in no small measure to the Severity Level II problem. GPC, Southern Nuclear, and Mr. Bockhold himself acknowledged his deficient conduct and, by letters dated August 5, 1994, as supplemented February 1, 1995, made commitments that he would not resume line management responsibilities at GPC or Southern Nuclear plants unless he had satisfactorily completed training in management communications and responsibilities, and the NRC received 60 days prior notice of the assignment.⁹⁶ This commitment was reiterated in correspondence regarding the applications to transfer the authority to operate the Vogtle and Hatch facilities to Southern Nuclear and was included in the orders authorizing those transfers.

D. Management Attitudes and GPC Credibility (Intervenor's Proposed Findings for Hearing on DG Issue at 68-78, 225-60)

Intervenor argues (Intervenor's Proposed Findings at 68-78, 225-60) that evidence of the bad character of the proposed transferee, includes: (1) GPC's operating philosophy of power generation above safety,⁹⁷ (2) intimidation of Mr. Mosbaugh in the January 1990 meeting where Mr. Bockhold had written the word "backstabbing" on the board after Mr. Mosbaugh's allegations that Mr. Kitchens had violated TS requirements by opening dilution valves,⁹⁸ (3)

⁹⁶ Mr. Bockhold further committed that he would not assume a line management position at any nuclear power plant prior to February 1988 without satisfying the conditions stated.

⁹⁷ Intervenor asserts that Mr. Hairston's statement that he has two goals in operating a nuclear plant, i.e., "staying on the line and short refueling outages," Tr. 9387-88 (Hairston), indicates that Mr. Hairston places continued operation and short outages over safety. Intervenor Proposed Findings at 69. Mr. Hairston testified that safety is not a goal, but a foundation for generating power. These opinions are not evidence of a poor attitude toward safety.

⁹⁸ While Mr. Mosbaugh's perception of the "backstabbing" incident may have led him to believe that GPC suspected him of prompting inquiries by the NRC, Messrs. Bockhold and Kitchens both testified that they were not aware at that time that Mr. Mosbaugh had given any allegation to the NRC, and Mr. Bockhold believed that the word referred to an undesirable working relationship between Mr. Mosbaugh's organization and Mr. Kitchen's organization that needed to be resolved. Bockhold Rebuttal at 2-4; Kitchens Rebuttal at 2-4; Tr. 13,597-601 (Kitchens); Tr. 13,347-48. Thus, it appears that the incident is an example of Mr. Bockhold's forceful and sometimes overbearing management style.

Mr. Bockhold's emphasis on a "yes sir" attitude,⁹⁹ (4) the GPC employee survey results,¹⁰⁰ (5) Mr. Bockhold's apparent disdain for regulatory involvement and attitude about conveying information to the NRC,¹⁰¹ (6) Mr. Bockhold's handling of the FAVA microfiltration system concern, and (7) the selective memory and opinions of Mr. Hairston.¹⁰² Intervenor's Findings at 69-78, 225-60.

I am not persuaded that any of these events are evidence of a lack of character. The intensity with which Mr. Mosbaugh pursued his concerns for over 5 years is an indication of the isolation he felt in an organization that did not adequately resolve his concerns. Mr. Mosbaugh's deeply held belief that GPC suspected either he or his department was relaying concerns to the NRC led him to tape surreptitiously conversations at the Vogtle facility.

The NRC Staff also held serious concerns about corporate and site management which, in addition to the allegations received by that time, led the NRC to convene a meeting with senior GPC officials on April 30, 1990, to candidly discuss these concerns, particularly with respect to the performance and attitude of

⁹⁹ Mr. Mosbaugh asserts that, during a February 7, 1990 meeting on reorganization and personnel downsizing, Mr. Bockhold mentioned his training in saying "yes sir" and told Mr. Mosbaugh if you "can't conform and accept, then you need to get out." Mr. Mosbaugh interpreted the remarks to mean that he should conform to management's view of the "dilution valve" matter during his upcoming OI interview. Intervenor's Proposed Findings at 39-43. Mr. Bockhold could not recall the remarks, but believed the meeting was about accepting upper management's directions regarding reorganization philosophy or the elimination of particular positions in the organization. Bockhold Rebuttal at 5-6. Whether or not Mr. Mosbaugh is correct about the reason for the statement, the statement, if made, would exemplify Mr. Bockhold's overbearing management style.

¹⁰⁰ In his proposed findings (at 235), Intervenor states that the results of a survey of nuclear personnel taken in the spring of 1990 showed that 73% of Vogtle employees agreed with the statement "Employees are afraid to voice an opinion that management does not want to hear" and 52% of Vogtle employees agreed with the statement "I am afraid to voice an opinion that my management does not want to hear." GPC's response to these survey results and the problems revealed by the issues in DG information disclosed by the NOV was to remind employees that conditions adverse to nuclear safety should be brought to management's attention and are to be addressed and resolved. Hairston Rebuttal (ff. Tr. 13,439) at 2-6; GPC NOV Reply at 6.

¹⁰¹ Intervenor asserts that GPC's (1) untimely recognition of the NRC's 1990 onsite problems with Mr. Bockhold's attitude and communications, (2) failure to acknowledge personnel errors as a root cause in the NOV Response, and (3) Mr. Hairston's testimony regarding Mr. Bockhold's performance is evidence that GPC management still shows a lack of concern for completeness and accuracy of information submitted to the NRC. Intervenor's Proposed Findings at 244-47. Mr. Hairston's 1990 actions (including telephone calls to Mr. Ebnetter) and his testimony that Mr. Bockhold's management style sometimes caused him (Mr. Bockhold) to miss opportunities, does not indicate a lack of concern for accurate and complete communications with NRC. NRC Staff has observed improved communications and performance once Mr. Bockhold was no longer in a Vogtle line management position. Tr. 15,194 (Mathews). These improvements, the corrective actions taken, and GPC's Response to the Modified NOV (including the commitments regarding Mr. Bockhold), provide reasonable assurance that the problems of the past have been addressed.

¹⁰² Intervenor asserts that it is incredible that Mr. Hairston (1) did not recall the discussion about DG starts during the April 19, 1990 telephone call ("Call A") between GPC corporate and site personnel, but did remember his prior call that same day with reactor operators, and (2) had a limited understanding of dewpoints. Intervenor Proposed Findings at 253-54. Tape 58 shows that Mr. Hairston had limited involvement in "Call A" and merely asked if the absence of trips in the count had been verified. GPC Exh. II-2, at 11-14. By contrast, he spoke at length with an operator about whether he had correctly described his observations and actions in the DG room during the SAE. Thus, it is not unreasonable that Mr. Hairston might have a more vivid recollection of one incident occurring on that date.

the Vogtle General Manager, George Bockhold. During the succeeding months, Mr. Bockhold played a major role in the failure of GPC to submit complete and accurate information to the NRC. GPC's communication record improved once Mr. Shipman replaced Mr. Bockhold in October 1990.¹⁰³

The NRC Staff concluded that problems experienced by GPC have been addressed and that GPC has accepted responsibility for its performance failures in its response to the NOV and in testimony during the license amendments proceeding. Corrective actions have included corporate statements to employees emphasizing the need for open and frank communications at the facility, and the Southern Nuclear and GPC commitments with respect to management training for Mr. Bockhold. These corrective actions and improvements in performance indicate that GPC or Southern Nuclear do not lack the requisite character and attitude to be an NRC licensee. Consequently, I do not conclude that these events are evidence of bad character.

E. Discriminating Against Employees for Engaging in Protected Activities (Petition §§ II.a, III.4; July 8, 1991 Supplement § II)

Petitioners assert that Mr. Hobby, who was GPC's General Manager of Nuclear Operations Contract Administration (NOCA) from December 1988 to April 1990,¹⁰⁴ was discharged from GPC after attempting to bring to GPC management's attention his concern that it had improperly transferred control of its nuclear licenses to SONOPCO. Petitioners state that Mr. Hobby had earlier been instructed by GPC Vice President of Bulk Power, Fred R. Williams, to destroy all copies of the confidential memorandum dated April 27, 1989, that had been written by Mr. Hobby and co-signed by GPC Senior Vice President—Fossil and Hydropower, George F. Head, expressing concern for the perception that GPC may have improperly transferred control of its nuclear facilities. Petitioners also assert (Petition § III.9.d) that GPC and SONOPCO management retaliated against managers who make their regulatory concerns known to them.¹⁰⁵

On February 6 and 28, 1990, Mr. Hobby filed complaints with the Department of Labor (DOL) contending that he had been discharged for engaging in protected activity in violation of section 210 (now 211) of the Energy Reorganization Act (42 U.S.C. § 5851) of 1974, and the regulations promulgated by DOL at 29 C.F.R. Part 24. Each of the above issues that Mr. Hobby identified

¹⁰³ Mr. Hairston testified that Mr. Bockhold's management style sometimes led Mr. Bockhold to miss opportunities and that, although qualified, it was unlikely that Mr. Bockhold would return to line management at a nuclear power facility. Tr. 11,551-54 (Hairston).

¹⁰⁴ Mr. Hobby was also Assistant to GPC Senior Vice President, Mr. George Head, until Mr. Head retired in May 1989. Mr. Head's position was then filled by Mr. Kerry Adams.

¹⁰⁵ Although not expressly stated in the petition, the complaints of both Messrs. Hobby and Mosbaugh in their respective DOL discrimination suits are pertinent to this concern.

in the petition to the NRC with respect to his discharge was included in the complaints. *See* DOL Case 90-ERA-30.

On August 4, 1995, the Secretary of Labor (Secretary) issued a Decision and Remand Order, finding that in 1990, senior managers of GPC discriminated against Mr. Hobby when his position was eliminated and he was forced to resign from GPC.¹⁰⁶ The Secretary determined that GPC terminated Mr. Hobby for engaging in protected activities, which included raising safety concerns related to the operation of the Vogtle facility in the April 27, 1989 memorandum. This Decision and Remand Order rejected the DOL Administrative Law Judge's Recommended Decision and Order that had been issued on November 8, 1991, which found that actions taken against Mr. Hobby were not motivated by his engaging in protected activities. The Secretary remanded the complaint to the Administrative Law Judge to determine a complete remedy.¹⁰⁷

On October 4, 1995, the NRC conducted a predecisional enforcement conference regarding the Secretary's Decision and Remand Order to discuss the apparent violation, the root cause, and GPC's corrective actions to preclude recurrence (*see* Conference Summary dated October 11, 1995). The Conference was open to the public in accordance with section V of the NRC Enforcement Policy, NUREG-1600, and written comments were subsequently submitted by Mr. Hobby for NRC consideration in reaching its enforcement decision.

The Commission's regulations in section 50.7, "Employee Protection," prohibit discrimination by a Commission licensee against an employee for engaging in protected activities. On May 29, 1996, the NRC issued a Notice of Violation to GPC for two separate violations of section 50.7 — one in accordance with the Secretary's finding regarding Mr. Hobby, and the other in accordance with the Secretary's finding that Mr. Mosbaugh had been discriminated against by being discharged for making audio tape recordings that constituted evidence gathered in support of a nuclear complaint, and for engaging in other protected activities. The Notice of Violation regarding Mr. Mosbaugh was in accordance with the Secretary of Labor's Decision and Remand Order in DOL cases 91-ERA-001 and 91-ERA-011 on November 20, 1995, finding that Mr. Mosbaugh's sus-

¹⁰⁶ The Secretary also found that other acts of discrimination occurred such as relocation of Mr. Hobby's office, restrictions of his access to the building, and revocation of his executive parking privileges.

¹⁰⁷ This DOL case (90-ERA-30) also considered Petitioners' assertion (*see* Section 2.206 Petition § III.4; July 8, 1991 Supplement § II) that Mr. McDonald knowingly submitted false testimony in another DOL proceeding ("*Yunker/Fuchko*") in an attempt to demonstrate that Messrs. Gary Yunker and John Fuchko were not improperly kept out of a GPC position that would participate in the SONOPCO Project. Petitioners claim that Mr. Hobby advised GPC's counsel before the DOL hearing that Mr. McDonald's proposed testimony was false and that GPC's counsel responded by advising Mr. Hobby that his testimony would have to be changed. In his Decision and Remand Order of August 4, 1995, the Secretary stated, in relevant part: "Because I found other evidence sufficient to establish that Complainant [Mr. Hobby] engaged in protected activity on January 2, [1989 (the prehearing meeting),] it was unnecessary to consider at that juncture whether counsel attempted to suborn Complainant to perjury. Even if counsel did, that evidence would not alter this decision." Decision and Remand Order at 13. *See also id.* at 5, 9-13.

pension and discharge were acts of retaliation for engaging in protected activity. The NRC stated that these violations were of very significant regulatory concern because they involved acts of discrimination by senior corporate management, and the NRC categorized each of the two violations as Severity Level I. Because the 5-year period provided in the Statute of Limitations for imposing a civil penalty had expired, no civil penalty was proposed for the violations. The NRC took this enforcement action to emphasize the importance of ensuring that employees who raise real or perceived safety concerns shall not be subject to discrimination for raising those concerns and that every effort will be made to provide an environment in which all employees may freely identify safety issues without fear of retaliation, harassment, intimidation, or discrimination.

The NRC also issued separate letters to each of the senior corporate managers the Secretary identified to be involved with the discriminatory actions.¹⁰⁸ In these letters, the NRC recognized that the discrimination found by the Secretary occurred over 5 years ago, prior to implementation of 10 C.F.R. § 50.5, "Deliberate Misconduct," and that the NRC, therefore, was taking no enforcement action against these senior managers. The NRC expressed concern that the discriminatory actions found by the Secretary could have had a chilling effect on other GPC employees; emphasized that harassment, intimidation, and discrimination against a licensee's employees for their engaging in protected activities is unacceptable; and provided official notice as to the enforcement actions against individuals that the NRC is authorized to take under section 50.5.

During the enforcement conference and in a written reply dated June 27, 1996, GPC denied the violations, objected to the NRC's reliance on the Secretary's decisions that were not yet final agency action, and acknowledged its right to appeal the Secretary's decisions once they become final.

Mr. Hobby's allegation that he was unlawfully dismissed because of a concern about the improper transfer of control of licensed activities is substantiated by the Secretary's decision of August 4, 1995. Mr. Hobby's regulatory concern regarding transfer of control constituted a protected activity.¹⁰⁹ Therefore, Mr. Hobby's dismissal because he expressed this regulatory concern is a violation of section 50.7. I am satisfied that the NRC has taken appropriate enforcement action to prevent the recurrence of violations of section 50.7 in the future, and to ensure a proper environment in which employees can express regulatory concerns without fear of retaliation, harassment, intimidation, or discrimination. To the extent that Petitioners' request for NRC involvement relates to matters

¹⁰⁸ In the *Hobby* case, the Secretary identified Messrs. Fred Williams, Dwight Evans, H.G. (Grady) Baker, Jr., and Thomas Boren. In the *Mosbaugh* case, the Secretary identified Messrs. A.W. Dahlberg and Ken McCoy.

¹⁰⁹ As I discuss in Section III.B of this Director's Decision, I am satisfied that the alleged transfer of control of licensed activities for GPC nuclear facilities did not, in fact, occur. This fact does not, however, alter the finding that Mr. Hobby engaged in a protected activity.

properly within the jurisdiction of the NRC, the request has been granted by means of these enforcement actions.

I find no reason to withhold my Decision on this 2.206 petition because of GPC's right to appeal the Secretary's decision when it becomes final. Further NRC action in the event of a successful appeal is not precluded by my Decision at this time.

F. Conclusions Regarding GPC's Character

The NRC reviews, inspections, and investigations related to the issues in the petition, as supplemented by the license transfer amendment proceeding, revealed a number of instances where the NRC was given incomplete and inaccurate information associated with the proposed license transfer to Southern Nuclear and DG reporting. The allegations that there was an illegal transfer of authority to control operations at the Vogtle and Hatch facilities and that GPC and Southern Nuclear otherwise lacked the character and competence to operate a nuclear power plant were not substantiated.

With respect to Petitioners' claim that GPC and Southern Nuclear routinely engaged in unsafe operating practices, the NRC found instances where GPC had violated NRC requirements, but the matters identified do not support Petitioners' allegation that GPC or Southern Nuclear (1) praised managers for taking risks, (2) did not take any adverse action against managers or employees who engage in nonconservative and questionable compliance practices, and (3) refused to critically investigate events or practices resulting in LERs.

With respect to GPC communications related to the proposed license transfer to Southern Nuclear, the NRC Staff found that there were instances where the NRC was provided inaccurate or incomplete information about the existing and proposed organizational structure in the formation of Southern Nuclear during an oral presentation to the Commission in March 1989 while discussing the chain of command for the Vogtle facility, in GPC's written response to the petition, and in licensing correspondence supporting the applications for transfer. These inaccuracies, when considered in the context of the extensive interactions between GPC and the NRC, were not significant and are not evidence of an intent to misrepresent or deceive the NRC. Thus, the misstatements do not warrant NRC enforcement action.

The NRC Staff did confirm that significant violations of Commission regulations have occurred at the Vogtle facility since 1987 and these violations have resulted in escalated enforcement actions by the NRC. The violations involved (1) opening "dilution valves" required to be locked closed; (2) providing inaccurate or incomplete information to the NRC regarding DG testing after the March 20, 1990 SAE; and (3) discriminating against Messrs. Mosbaugh and Hobby for engaging in protected activities.

The Staff's review of the boron dilution violation revealed that the GPC employee did not meet TS requirements or NRC expectations, but there was not a sufficient basis to conclude that the individual had intentionally violated a TS requirement. GPC and the individual admitted the mistaken TS interpretation.

Based on the findings of the DOL, the Staff concluded that GPC had discriminated against the Petitioners because they engaged in protected activities, which was a Severity Level I problem. This NRC enforcement action was taken to emphasize GPC's obligation to ensure that employees who raise real or perceived safety concerns are not subjected to discrimination and that assiduous efforts are required in order for employees to have an environment where they may freely identify safety issues without fear of retaliation, harassment, intimidation, or discrimination. GPC has taken corrective action consistent with these goals.

The failure of GPC to provide the NRC with complete and accurate information relative to DGs throughout 1990 that were cited in the Modified NOV were serious. The significance of the performance failures of GPC stems not from the effect such inaccuracies had on the safety of plant operation, but because the circumstances surrounding the communications demonstrate an inadequate regard by a number of senior Licensee officials, and by GPC management as a whole, for providing complete and accurate information to the NRC. Information about the DGs and GPC's determinations about the causes of errors were important for the NRC to determine whether GPC was fulfilling its responsibilities as a licensee.

GPC was clearly aware of the NRC's interest in the DGs because the NRC specifically asked GPC to address DG reliability as part of its restart presentation of April 9, 1990. GPC should have taken steps to ensure the completeness and accuracy of its submittals, but instead, at times, engaged in poorly defined efforts to obtain information to satisfy the NRC on an issue having a direct bearing on the NRC's decision to allow restart. This performance is not acceptable.

It is also significant that GPC missed repeated opportunities to ensure completeness and accuracy of information and to promptly correct information when its own staff questioned the accuracy of the April 9 information and subsequent explanations about inaccurate information. Even though senior GPC management became involved, GPC did not recognize the need to correct the April 9 start data until the NRC's request during the August 1990 inspection. Further, GPC continued to submit information that was inaccurate and incomplete and did not recognize the implications of its performance failures until they were identified by the NRC in the enforcement action almost 4 years later.

The NRC Staff has concluded, however, that the performance problems exhibited throughout these events are not sufficient to establish that Southern Nuclear, and the GPC employees who would work for that company as a result of a transfer of the Hatch and Vogtle operating licenses to Southern Nuclear,

lack the requisite character to be a licensee. GPC's overall performance in keeping the NRC informed of post-repair and troubleshooting activities, GPC's technical competence in addressing those matters, Mr. Hairston's efforts to keep the NRC informed about errors identified as GPC became aware of them, and the corrective actions taken by GPC management in response to the NOV (which include measures to ensure effective communications and resolution of employee concerns, and measures emphasizing open, complete, and accurate communications with the NRC), are among the indications of GPC's diligence, competence, and character. Testimony of Messrs. Roy P. Zimmerman and Luis A. Reyes on the Character and Integrity Contention, ff. Tr. 15,256, "Zimmerman-Reyes," at 5-7. The NRC Staff's evaluation of GPC's response to the May 9, 1994 NOV on the DG issue and GPC and individual responses to the DFIs issued to Messrs. Bockhold, McCoy, Greene, Horton, Frederick, and Majors revealed that GPC officials have accepted responsibility for, and regret, their part in GPC's deficient performance. The NRC Staff remained concerned, however, about whether GPC, Southern Nuclear, and Mr. Bockhold fully understood the ramifications of the DG enforcement action and the future performance of Mr. Bockhold in line management positions at nuclear power facilities. Staff Exh. II-51 (cover letter).

I find that GPC's tendency to defend information provided during the restart presentation, rather than to verify the accuracy of the data, was inconsistent with the simple candor upon which the NRC relies to discharge its responsibility for ensuring public health and safety. *See North Anna*, CLI-76-22, 4 NRC at 491. There is not a sufficient basis, however, to conclude that GPC endeavored to intentionally mislead the NRC or otherwise engaged in a pattern of deception and falsehood in its licensing communications. The failures can be traced to (1) the collective performance of senior GPC managers, including the management style of the General Manager who repeatedly failed to ensure that complete and accurate information was provided to the NRC; (2) the reluctance of site and corporate personnel to question the views of superiors; and (3) the inadequate efforts to verify information submitted to the NRC.

Based on a review of the facts set forth above, including the evidentiary record of the adjudicatory proceeding, the enforcement actions taken against GPC (i.e., regarding opening "Dilution Valves," DG reporting, and section 50.7 violations), and the favorable performance of GPC (and corrective action taken) since 1990, there is no basis to conclude that Southern Nuclear lacks the requisite character, integrity, and competence necessary to operate the Vogtle and Hatch facilities in accordance with the Commission's rules and regulations. The individuals employed by GPC and Southern Nuclear have not been shown to have intentionally submitted to the NRC information that was inaccurate, incomplete, or misleading in a material respect. Rather, the performance problems exhibited in GPC communications to the NRC were due

to the failures of certain individuals to take steps necessary to ensure the accuracy and completeness of information and to promptly correct such misinformation. In recognition of the role, management style, and repeated performance failures of the former General Manager, the license transfers for the Vogtle and Hatch facilities have been conditioned to limit his involvement in line management activities consistent with commitments of GPC and Southern Nuclear.

IV. CONCLUSION

As discussed above, NRC has conducted several inspections, investigations, and technical reviews regarding the concerns in the petition, and proceedings before NRC and DOL have been conducted regarding most of the concerns. Some of the concerns raised by the Petitioners were substantiated. Violations of regulatory requirements have occurred in the operations of the Vogtle facility. Notices of Violation and civil penalties have been issued to the Licensee, letters have been issued to several individuals, and certain conditions regarding one individual are being imposed by the NRC in conjunction with the license transfers. To this extent, the Petitioners' request for action pursuant to section 2.206 had been granted.

On the basis of the NRC Staff's review and the license amendments hearing record, I conclude that no unauthorized transfer of the Vogtle operating licenses occurred, and that the GPC nuclear facilities are being operated in accordance with NRC regulations and do not endanger the health and safety of the public. On balance, the evidence does not support the conclusion that GPC, the SONOPCO Project, or Southern Nuclear deliberately provided false or misleading information to the NRC or that Southern Nuclear or GPC (including the GPC employees that would be employed by Southern Nuclear as a result of the license transfer) lack the requisite character and integrity to be an NRC Licensee as required by section 182 of the Atomic Energy Act, 42 U.S.C. § 2232, and 10 C.F.R. § 50.80. Thus, there is no basis upon which to grant Petitioners' request that the operation of the facility be suspended.

With respect to Petitioners' request that the NRC institute proceedings and impose civil penalties based on the matters addressed in the petition, the issues in the petition that give rise to substantial health and safety issues have, in fact, been the subject of a lengthy proceeding and escalated enforcement actions by the NRC. Also, based upon the findings of the DOL, the NRC has addressed both Petitioners' specific concerns that they were discriminated against for engaging in protected activities (and the associated issue that GPC retaliates against managers who make their regulatory concerns known) by taking escalated enforcement actions against GPC. Based on actions already taken by the NRC Staff, there is reasonable assurance that the GPC facilities operate with

adequate protection of the public health and safety. Therefore, I decline to take any further action with respect to matters raised in the petition. To this extent, the Petitioners' request for action pursuant to section 2.206 is denied.

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

FOR THE NUCLEAR
REGULATORY COMMISSION

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

Dated at Rockville, Maryland,
this 18th day of March 1997.

APPENDIX

ALLEGED ILLEGAL LICENSE TRANSFER ISSUES

"Intervenor's Prehearing Statement of Issues" (Statement of Issues), dated December 12, 1994, raised twenty-eight issues to support Intervenor's illegal transfer issue for the license amendments proceeding.¹¹⁰ The issues were submitted in support of Intervenor's contention that the Vogtle operating license should not be transferred to Southern Nuclear because it lacks the requisite character and integrity. The twenty-eight issues repeat and further supplement assertions in the petition regarding an illegal transfer of control of GPC nuclear facilities.

¹¹⁰ Although Intervenor identified 28 issues in his Statement of Issues, two issues were both numbered 14A and 14B, and Intervenor presented no evidence or proposed findings on Issue 25.

I. ALLEGED INACCURACIES ABOUT MR. FARLEY'S ROLE IN THE CONTROL OF THE VOGTLE FACILITY

The gravamen of Intervenor's twenty-eight issues and related issues in the petition, as supplemented, is that the nuclear officers in SONOPCO Project reported to Mr. Farley, rather than to Mr. Dahlberg, GPC's CEO, and to demonstrate that Mr. Farley controlled the Vogtle facility based upon his alleged involvement in (1) controlling daily operations; (2) establishing and implementing nuclear policy decisions; (3) employing, supervising, and dismissing nuclear personnel; and (4) controlling costs. Intervenor also asserts that numerous documents and statements provided to the NRC regarding the organizational structure and responsibilities for managerial control of the Vogtle facility were inaccurate or incomplete because they do not show Mr. McDonald reporting to Mr. Farley or Mr. Farley functioning as the *de facto* Chief Executive Officer of the SONOPCO Project.

A. Controlling Daily Operations

Intervenor asserts in Issue 1 that GPC misled the NRC about the corporate management structure over the Vogtle facility during a March 30, 1989 meeting in that Mr. McDonald's description of the chain of command ignored Mr. Farley's role as the chief executive over the Southern Company's nuclear division, which commenced exercising operating responsibility over GPC's nuclear plants in November of 1988. Intervenor asserts that Mr. McDonald inaccurately stated that he solely reported to GPC's CEO, Mr. Dahlberg. Intervenor claims that Mr. McDonald reported to Mr. Farley who reported to Mr. Edward L. Addison, the President and CEO of The Southern Company. Similarly, in Issue 10, Intervenor alleges that GPC's April 1, 1991 Petition response falsely stated that certain organizational charts filed with the SEC and included with a May 15, 1989 memorandum from Mr. Fred Williams to Mr. Hobby, accurately depicted GPC's organizational structure before the incorporation of Southern Nuclear in that they do not show that Mr. McDonald reported to Mr. Farley or that Mr. Farley functioned as the *de facto* Chief Executive Officer of the SONOPCO Project.

The hearing record does not support Intervenor's claim that Mr. Farley exercised control over GPC's nuclear facilities beginning in November 1988. Mr. Farley testified that he had neither the authority, nor attempted to control management decisions about licensed activities or personnel matters concerning the Vogtle facility. Prefiled Testimony of Joseph M. Farley, ff. Tr. 1749, "Farley," at 17-18, 22; Tr. 1801-02 (Farley). Mr. Shipman (who in October 1988 was the Vogtle General Manager for Support and in January 1991 became the Vogtle General Manager), Mr. McCoy (GPC Vice President-Plant Vogtle), and Mr.

Hairston testified that Mr. Farley did not issue orders or instructions regarding the operation of the Vogtle facility or any aspects of the facility or otherwise become involved in the management of personnel or activities at the Vogtle facility. Tr. 1976 (Shipman); Prefiled Testimony of C. Kenneth McCoy, ff. Tr. 1560, "McCoy," at 19; Prefiled Testimony of W. George Hairston, III, ff. Tr. 1688, "Hairston," at 47-48; Tr. 1726-28, 1740 (Hairston). In addition, Mr. McDonald testified that Mr. Farley never influenced him regarding operation of the Vogtle facility. Prefiled Testimony of R. Patrick McDonald, ff. Tr. 1249, "McDonald," at 25; Tr. 1550-51 (McDonald).

The record of the *Hobby* DOL proceeding indicates that GPC President, Mr. Dahlberg, testified that the operation of GPC's nuclear facilities is his direct responsibility; that Mr. McDonald takes his management direction from Mr. Dahlberg regarding the operation of GPC's nuclear plants; and that Mr. McDonald reports to Mr. Dahlberg for management operations dealing with GPC plants (*Hobby* DOL Transcript at 305, 307, 309). Mr. Farley stated that he did not have any responsibility for operating GPC's nuclear facilities and that Mr. McDonald did not report to him with respect to the operation of Hatch and Vogtle (*id.* at 567, 568). Mr. McDonald stated that he reported to Mr. Dahlberg regarding the operation of GPC's nuclear facilities (*id.* at 613, 614).

In a deposition of May 5, 1990, taken in the same *Hobby* DOL proceeding, at 13 and 14, Mr. McDonald stated that he had no reporting responsibilities to Mr. Farley. A May 15, 1989 memorandum from Mr. Fred D. Williams, the GPC Vice President for Bulk Power Markets, to Mr. Hobby, forwarded a copy of the most recent published organization chart which showed that Mr. McDonald reported to Mr. Dahlberg for operation and support activities of the Vogtle and Hatch facilities, and that Mr. Hairston reported to Mr. McDonald.

While the record shows that Mr. Farley received verbal reports from Messrs. McDonald, Hairston, McCoy, Louis B. Long (SCS Vice President-Technical Services), and Charles McCrary (SCS Vice President-Administrative Services) concerning the performance of GPC's nuclear units, and attended staff meetings (Issue 15), this does not support a determination that Mr. Farley was part of the management structure over the Vogtle facility. As the future CEO of Southern Nuclear and as manager over certain support services provided to the Vogtle facility, Mr. Farley periodically briefed The Southern Company Board of Directors, received information, and attended meetings. Such activities do not amount to control of operations or other licensed activities at the Vogtle facility.

Intervenor asserts that, during a deposition, Mr. Shipman stated that Mr. McDonald and Mr. Hairston reported to Mr. Farley. Mr. Shipman testified during the license amendments hearing that he understood Mr. McDonald reported to Mr. Farley for certain things and there were certain things that Mr. McDonald did not report to Mr. Farley on. Tr. 1966 (Shipman). This is consistent with Mr. Farley's testimony that Mr. McDonald would informally report to him

with regard to governmental affairs, such as congressional proceedings, and administrative matters unrelated to the operation of the plants. Such activities do not indicate that Mr. Farley had line management responsibilities or that Mr. McDonald reported to Mr. Farley with respect to any licensed activities involving the Vogtle facility.

The Petitioners claim that control of operating the nuclear facilities is based upon Mr. Hobby having witnessed the day-to-day operation at GPC's corporate offices (Petition at 5-6). During the hearing, however, no direct evidence was offered to support the claim that Mr. McDonald reported to Mr. Farley regarding the operation of the Hatch or Vogtle facilities. Messrs. Hobby and Mosbaugh both acknowledged that they had no personal knowledge that Mr. McDonald received direction from Mr. Farley regarding the operation of the Vogtle or Hatch facilities. Tr. 2157-58 (Mosbaugh) and Tr. 2377 (Hobby); *Hobby* DOL Transcript at 239). Mr. Mosbaugh admitted that he had no first-hand knowledge of the day-to-day interaction among Messrs. McCoy, Hairston, McDonald, and GPC officers, and had never been in the Birmingham, Alabama offices of SONOPCO. Tr. 2128 (Mosbaugh).

Intervenor also asserts (Issue 1) that Mr. Dan Howard Smith, a Department Manager with Oglethorpe Power Corporation (a co-owner of the Vogtle facility), had observed that Mr. Farley was the chief executive of the SONOPCO Project, that Mr. McDonald reported to Mr. Farley who reported to Mr. Addison (the President and CEO of The Southern Company), and that Mr. Farley's control over nuclear operations might violate the terms of the operating licenses for GPC's nuclear facilities.¹¹¹ However, Mr. Smith testified at his deposition that after reading the transcript of the March 30, 1989 meeting on the Vogtle Unit 2 full-power license, during an April 1989 co-owner's committee meeting, GPC provided a chart, at his request, that clarified the reporting chain. Smith Deposition at 22-23, 36-37.

Intervenor's reference to Mr. Hobby's memorandum of April 27, 1989, which alluded to concerns about Mr. McDonald's reporting relationship (Issue 1), does not establish that there was an improper exercise of control by Mr. Farley and The Southern Company. Mr. Rogge, the NRC Senior Resident Inspector, testified that "No one to my knowledge ever expressed a concern that GPC was not in control of operations at Vogtle." Testimony of Frederick R. Allenspach, Darl S. Hood, and John F. Rogge on the "Illegal Transfer" Issue, ff. Tr. 2620, "Allenspach, Hood, and Rogge," at 6.

In Issue 3, Intervenor asserts that 1988 amendments to FSAR Chapter 1 inaccurately depicted the corporate organization for the operation of the Vogtle facility because FSAR § 1.4.1.2, "Description of Corporate Organization," did

¹¹¹ Mr. Hobby's Memorandum of April 27, 1989 (Exhibit A of the September 21, 1990 Supplement to the Petition) refers to Mr. Smith's concern about control of GPC facilities.

not state that "The Southern Company had newly established a nuclear division with responsibility for operating GPC's nuclear plants."

The NRC was given timely notification of the plans to form a separate operating company by virtue of the meetings held on February 16 and May 3, 1988, with the Commissioners and others to brief the NRC about The Southern Company's tentative plans to form a separate nuclear operating company and to review the several phases that would have to be involved, pending SEC approval, and ultimate license amendments, as well as by meetings held March 2 and 18, 1988, and July 25, 1988, with NRC personnel. Farley at 11-12. Therefore, its omission from FSAR § 1.4.1.2 by the 1988 amendments was not significant in terms of NRC awareness.

In Issue 4, Intervenor claims that the 1988 amendment to FSAR Chapter 13 (i.e., Vogtle FSAR Amendment 39, dated November 23, 1988) was inaccurate because it described the Executive Vice President-Nuclear Operations (Mr. McDonald) as an officer of both GPC and APC who is "responsible to the chairman and CEO of each company for all aspects of operation of the nuclear generating plants in the Georgia Power and Alabama Power systems, as well as technical and administrative support activities provided by SCS," but did not indicate that Mr. Farley was the functioning chief executive of SONOPCO Project. Intervenor claims that the amendment was also misleading because technical and administrative services reported to an executive officer of the SONOPCO Project, with Mr. Farley serving as chief executive officer.

As President and CEO of APC in November 1988, Mr. Farley was not part of Vogtle line management, and he exercised no line management responsibility over licensed activities at the Vogtle facility. A September 21, 1988 memorandum by Mr. Addison noted that Mr. Addison had asked Mr. Farley to guide the formation of the new company (Southern Nuclear) and that Mr. McDonald was serving as Executive Vice President of GPC and APC and was responsible for the operation of the Hatch, Vogtle, and Farley nuclear facilities. Thus, the absence of Mr. Farley from the Chapter 13 organizational charts and descriptions submitted by Vogtle FSAR Amendment 39 is not an inaccuracy.

Services by SCS to GPC were provided in accordance with a January 1, 1984 services agreement between them. Messrs. Louis Long, SCS Vice President of Technical Services, and Charles McCrary, SCS Vice President of Administrative Services, reported to Mr. McDonald with respect to the Vogtle facility, not to Mr. Farley. On April 24, 1989, the arrangement was made formal by a letter of agreement between Messrs. McDonald and H. Allen Franklin, the President of SCS at the time. McCoy at 8; Hairston at 21 and Tr. 1712; Deposition of Meier at 40-41. Therefore, Intervenor's claim of inaccuracy is not supported by the record.

In Issue 5, Intervenor states that the organizational chart, Figure 13.1.1-1, was inaccurate in the Vogtle FSAR amendment, dated March 28, 1990, because

it failed to depict Mr. McDonald's reporting relationship to Mr. Farley and it showed the Administrative and Technical Services Vice Presidents reporting to Mr. McDonald and then to Mr. Dahlberg. The hearing record does not support Intervenor's assertions.

Figure 13.1.1-1, as revised March 28, 1990, accurately shows that the Executive Vice President—Nuclear Operations, an officer of both APC and GPC, reported to the President and CEO of GPC on Vogtle matters since Mr. Farley was not involved in the operation of the Vogtle facility or activities authorized by the Vogtle licenses. Figure 13.1.1-1 also accurately depicted the Vice President for Administrative Services and the Vice President for Technical Services reporting to Mr. McDonald and then to Mr. Dahlberg. Under a services agreement between SCS and GPC, Mr. Dahlberg had the authority to direct activities of these SCS officers for the functions they were performing in support of plant operation (Hairston at 35). The fact that Mr. McCrary reports to Mr. Farley concerning certain administrative matters unrelated to plant operations, including the formation of Southern Nuclear and general industry activities (*see* Farley, ff. Tr. 1749, at 16; Hairston at 33; McCoy at 11), is not relevant to Vogtle licensed activities and does not indicate that Mr. Farley controlled operations at the Vogtle facility.

In Issue 18, Intervenor alleges that, during a January 11, 1991 meeting with the NRC, Mr. McDonald falsely stated that Mr. Farley had no responsibilities for administrative matters related to the SONOPCO Project. *See also* July 8, 1991 Supplement to Section 2.206 Petition, § IV. Based on the meeting transcript and Mr. McDonald's testimony, the January 11, 1991 statement was not inaccurate.

Mr. McDonald testified during the hearing that his statement on page 42 of the meeting transcript¹¹² was that prior to Phase II (the incorporation of Southern Nuclear), Mr. Farley had been performing a job as a Vice President of The Southern Company, had been providing certain services to Mr. McDonald under a contract with SCS, and had no responsibility for certain administrative support that was depicted on organization charts discussed during the meeting. Administrative support was being performed by Mr. McCrary for Mr. McDonald pursuant to an April 24, 1989 agreement. While Mr. McCrary provided administrative services to support Mr. Farley's responsibility to guide the formation of Southern Nuclear and Mr. Farley's general industry activities, Mr. McCrary did not report to Mr. Farley with respect to the administrative support function for the Vogtle facility. McDonald at 9.

¹¹²The meeting transcript, at page 42, shows that Mr. McDonald (referring to an organizational chart) states:

Yes. A month ago there was no line here. Mr. Farley was performing his job as a Vice President of the Southern Company. He had no responsibilities for this administrative support. That administrative support that we had basically was being done, and he was a part of a contract — it was a contract to me from Southern Services for providing essentially much the same support we have here now.

In Issue 7, Intervenor states that the March 1991 FSAR amendment revising Figure 13.1.1-1 is false because it shows the Executive Vice President–Nuclear Operations, Mr. McDonald, reported to the President and CEO of Southern Nuclear, Mr. Farley, for Southern Nuclear matters only, and because it shows that Mr. McDonald reported to the President and CEO of GPC for GPC matters. Intervenor claims that (1) Mr. McDonald reported to Mr. Farley on matters pertaining to Vogtle, (2) both Messrs. McDonald and Farley reported to Southern Nuclear Board of Directors on matters pertaining to GPC's nuclear operations, and (3) Mr. Farley reported to The Southern Company CEO, Mr. Addison, and to The Southern Company Management Council. Intervenor similarly alleges in Issue 22 that GPC's April 1, 1991 response to the petition falsely asserts that during Phase II (after incorporation of Southern Nuclear), all Southern Nuclear management in the reporting chain above the Vogtle Plant General Manager were officers of GPC because Mr. Farley stated during his deposition that he was never an officer of GPC.

Once Southern Nuclear was incorporated, Mr. Farley became its President and CEO and Mr. McDonald, who retained his positions as Executive Vice President of GPC and APC, became the Southern Nuclear Executive Vice President. Hairston at 37-38. Thus, Mr. McDonald reported to Mr. Farley, and they both reported to the Southern Nuclear Board of Directors, regarding Southern Nuclear matters. However, for licensed activities at the Vogtle facility, Mr. McDonald continued to report directly to GPC President and CEO, Mr. Dahlberg. Farley at 17-19; McDonald at 4; McCoy at 13. Since Mr. Farley was CEO of Southern Nuclear during Phase II, and was not part of the management chain for the Vogtle facility,¹¹³ Intervenor's assertions that Figure 13.1.1-1 and GPC's petition response were inaccurate were not substantiated.

In Issue 11, Intervenor alleges that in the April 1, 1991 response to the petition, GPC falsely represents that Mr. Farley did not have management control over GPC licensed activities or GPC personnel matters.

The record shows that Mr. Farley did not have control over GPC's licensed activities. Mr. McDonald, who signed the April 1, 1991 response, testified that Mr. Farley did not exercise any management control over GPC's licensed activities, and that he (McDonald) was not aware of a single instance where Mr. Farley controlled, or made, a GPC staffing or operating decision. McDonald at 10. Neither the hearing record nor results of NRC's regulatory oversight

¹¹³ The agreement executed by GPC and Southern Nuclear (GPC Hearing Exhibits 20 and 21) expressly stated that Southern Nuclear would not perform any activities in connection with the nuclear plants that were required by the operating licenses to be performed by the Licensee, GPC. Hairston at 36-38. As part of his responsibilities as Executive Vice President–Nuclear of The Southern Company, Mr. Farley briefed the Southern Company Board and Mr. Addison on nuclear developments. Farley at 21. This responsibility to provide information does not constitute control of licensed activities at the Vogtle facility.

support Intervenor's assertion that Mr. Farley had management control over GPC licensed activities or GPC personnel matters.

In Issue 20, Intervenor claims that statements by Mr. Stephen H. Chesnut (a GPC manager-in-training in August 1990), recorded on Mr. Mosbaugh's Tape No. 260, and statements during Mr. Shipman's August 1994 deposition, show that SONOPCO Project managers observed that Mr. Farley, rather than Mr. Dahlberg, controlled GPC's nuclear operations.¹¹⁴ *See also* October 1, 1990 Supplement to Petition at 4-5.

Given that (1) Intervenor's testimony concerning Mr. Chesnut's statements on Tape No. 260 was stricken from the record, (2) Intervenor subsequently withdrew the tape transcript, (3) Intervenor did not call Mr. Chesnut as a witness (*see* Tr. 1909-11, 2047), and (4) Mr. Shipman, a SONOPCO Project manager, testified that he never had any doubt that the responsibility for the licensing and operation of the GPC nuclear facilities rested with Mr. Dahlberg (Tr. 1982-83),¹¹⁵ there is no basis to conclude that these SONOPCO Project managers believed that Mr. Farley controlled GPC's nuclear operations or other licensed activities.

In Issue 23, Intervenor alleges that GPC's April 1, 1991 response to the petition falsely asserts that Mr. Dahlberg is contacted on a daily basis by GPC nuclear operating officers concerning the status of GPC nuclear plants in that "phone records" showed differently. Intervenor did not submit any "phone records" or other evidence to support his assertion.

The testimony of Mr. Dahlberg and Mr. McDonald established that Mr. Dahlberg or his staff received daily reports from a GPC nuclear officer concerning the status of GPC's nuclear plants and was contacted if some unusual or unexpected operational event occurred. Dahlberg at 16-17. McDonald at 3, 22. *See also* Tr. 1135, 1154 (Dahlberg).

Accordingly, the hearing record does not support Intervenor's allegation in Issue 23 that GPC's April 1, 1991 statement is inaccurate.

¹¹⁴ Similarly, in Issue 21, Intervenor alleges that in its April 1, 1991 response to the petition, GPC falsely asserts that (1) Vogtle project management does not assume that Mr. Farley, rather than Mr. McDonald, controls Vogtle's operations; and (2) Mr. McDonald reports to Dahlberg on all matters concerning the operation of GPC's nuclear facilities.

Mr. McDonald testified that he was confident that Vogtle managers understood that he, and all other GPC officers, managers, and employees, reported to Mr. Dahlberg on all matters pertaining to the operation of GPC's nuclear facilities as specified in the FSAR, and Intervenor's assumption that Mr. Farley was in control was based on statements by Mr. McCoy that had been taken out of context. McDonald at 17, 20-21.

¹¹⁵ Mr. Shipman said he had corrected his deposition statement (Intervenor Exh. 10) that, in April 1990, Mr. Hairston reported to Mr. Farley through Mr. McDonald to correctly indicate that Mr. Hairston reported to Mr. Dahlberg through Mr. McDonald. Tr. 1992-95; Licensee Exh. 25. Mr. Shipman explained that his initial deposition statement was in the context of information customarily provided to Mr. Farley by the SONOPCO Project executives and he thought at the time of his 1994 deposition that Messrs. Hairston and McDonald were officers of SCS as well as GPC and APC and, as such, reported to Mr. Farley with respect to SCS matters. Tr. 1965-67, 1983-85, 1993-95 (Shipman).

In Issue 15, Intervenor contends that GPC failed to tell the NRC, during a December 1988 inspection of the corporate offices in Birmingham, Alabama, that Mr. Farley was involved with the SONOPCO Project as CEO of the SONOPCO Project, and failed to inform the NRC about Mr. McDonald's "reporting relationship" to Mr. Farley.¹¹⁶ Intervenor claims that: (1) Mr. Farley reported to Mr. Addison and The Southern Company Management Council which served as a board of directors for the SONOPCO Project; (2) Mr. Farley was involved with the operation and management of The Southern Company's nuclear plants, presiding over weekly staff meetings; and (3) GPC's letter of December 29, 1988, to NRC continued to mislead the NRC about Mr. Farley's role by stating that, "as shown on FSAR Figures 13.1.1-2 and 13.1.1-3, the Executive Vice President, the Senior Vice President-Nuclear Operations and the Vice President-Nuclear do provide line management direction for the operation of the Plant."

The record shows that Mr. Farley was President of APC during the December 1988 inspection, and he did not become Executive Vice President of The Southern Company and SCS until March 1, 1989. Farley at 1. The announcement that he would be the CEO of Southern Nuclear upon its incorporation was not made until March 1989. Farley at 11; Tr. 1723 (Hairston).

Intervenor's assertion that Mr. Farley presided over weekly staff meetings designated as "Farley staff meetings" is not supported by the hearing record. Although SONOPCO Project staff meetings were held beginning in November 1988, Mr. Farley did not attend these meetings until he relocated to the SONOPCO Project offices, after his election to Executive Vice President of The Southern Company and SCE, effective March 1, 1989, and he provided no management oversight or direction at those meetings. Farley at 21; McDonald at 21; Hairston at 24. Consistent with providing support services to the SONOPCO Project and his future position as CEO of Southern Nuclear, Mr. Farley's attendance was to keep abreast of system plant developments and, as Executive Vice President-Nuclear of The Southern Company, the meetings enabled him to provide periodic reports to The Southern Company Board of Directors. McCoy at 17-18; Farley at 11, 21; McDonald at 21; Tr. 1341-42 (McDonald), Tr. 1848-51 (Farley), Tr. 1989-90 (Shipman); McCrary Deposition at 38. The fact that Mr. Farley was kept informed and periodically briefed The Southern Company Board

¹¹⁶ From December 19 through 21, 1988, the NRC conducted an inspection of the corporate organization, responsibilities, and functions of SONOPCO at Birmingham, Alabama, during Phase I of the Southern Nuclear transition (Inspection Report Nos. 50-321/88-41, 50-366/88-41, 50-424/88-60, 50-425/88-77, 50-348/88-33, and 50-364/88-33) and observed, in Part 3 of the report, that:

In preparation for combining the management of Vogtle, Hatch, and Farley into one organization, GPC has reorganized and moved the corporate nuclear operations to Birmingham. . . . Currently, the Executive Vice President and Senior Vice President for Nuclear operations are officers of both GPC and APC. . . . The Vice Presidents for each of the three projects (Vogtle, Hatch, and Farley) report to the Senior Vice President of Nuclear Operations.

of Directors does not warrant the conclusion that he was part of the management structure for the Vogtle facility or exercised control over its operation or its other licensed activities.

The hearing record does not support Intervenor's assertion that Mr. Hairston's letter of December 29, 1988, that referenced organizational charts shown in FSAR Chapter 13, misled the NRC about Mr. Farley's role in the operation of the Vogtle facility (*see also* Issues 4, 5, and 7 herein). Mr. Farley had no such role.

Accordingly, there is no basis to conclude that the NRC was misled during its December 1988 inspection or by subsequent submittals regarding the organization in control of GPC's licensed activities.

In Issue 16, Intervenor contends that during a July 25, 1989 meeting with the NRC, GPC failed to accurately portray the actual configuration of the SONOPCO organization by not revealing that Mr. Farley had management responsibility over the Vogtle facility. Since the record does not support that Mr. Farley had management responsibility over the Vogtle facility, this contention is not substantiated.

In Issue 24, Intervenor alleges that GPC omitted from the Vogtle Emergency Plan any discussion of Farley's management functions and responsibilities as they related to the Corporate Emergency Plan described in Appendix 7 of the Vogtle Emergency Plan (Revision 12, effective April 1990). Intervenor's bases for this allegation are that (1) the Vogtle emergency procedures demonstrate that Mr. Farley had an emergency plan responsibility because he was listed in the On-Call Project Manager's telephone list as "Georgia Power Corporate Management"; and (2) Messrs. McDonald, Hairston, and McCoy as well as the rest of the corporate emergency organization were controlled from a practical standpoint by Mr. Farley.

Mr. McCoy testified that Revision 12 (dated April 1990) of the Corporate Emergency Plan accurately indicated that Mr. Farley had no role in the Corporate Emergency Organization, and that Mr. Farley was not part of the "Senior Corporate Management" identified in the Corporate Emergency Notification Tree (Figure C-1 of the Corporate Emergency Plan for the Vogtle Electric Generating Plant, Revision 12). McCoy at 18-19; *see also* Tr. 1597 (McCoy); Supplemental Prefiled Testimony of C. Kenneth McCoy, ff. Tr. 1560, "McCoy Supplemental," at 1. Even though Mr. Farley was accurately identified as Executive Vice President-Nuclear of The Southern Company, his name was listed under the heading "Georgia Power Corporate Management" in the On-Call Project Manager's telephone list. The heading was incorrect and, beginning in 1991, the section was renamed "Corporate Management" and included the designated title for each individual. McCoy Supplemental at 1; *see also* Tr. 1574-76, 1588-89 (McCoy).

The On-Call Project Manager's telephone list does not identify who is to be called in the case of a significant event at the Vogtle facility, is not part of a procedure, and is not intended to be used by the On-Call Project Manager (corporate) to identify who is to be notified in the event of an emergency. Administrative procedure VNS-EP-04, entitled "Duties of the On-Call Project Manager" (GPC Exh. 9), identifies who is to be notified by the On-Call Project Manager, in what order,¹¹⁷ and Mr. Farley was not required to be notified by the On-Call Project Manager as a part of the emergency call-out procedures.¹¹⁸ McCoy Supplemental at 2-3; Tr. 1580-92 (McCoy).

The record does not support Intervenor's assertion that Messrs. McDonald, Hairston, McCoy, and the rest of the corporate emergency organization in Birmingham, Alabama, were controlled by Mr. Farley. Messrs. McDonald and McCoy both testified that there was no attempt by Mr. Farley to control the operation of the Vogtle facility and that line management authority over licensed activities at the Southern Nuclear offices was very clear — through Mr. McCoy to Mr. Hairston, Mr. McDonald, and Mr. Dahlberg. McCoy at 19; McDonald at 25. GPC's response to the March 20, 1990 Vogtle SAE also demonstrates that Mr. Farley did not participate in the emergency response, but only listened to discussions regarding the event consistent with his need to know information. Tr. 1825-29 (Farley).

Accordingly, the allegation in Issue 24 is not supported. The hearing record does not support that Mr. Farley had emergency plan responsibilities indicative of a control over GPC's nuclear facilities or that he exercised control over GPC managers and personnel involved with GPC's emergency response. Therefore, the claim that Mr. Farley was omitted from the Vogtle emergency plan in order to mislead the NRC is unwarranted.

In summary, Intervenor's assertion that Mr. Farley functioned as the *de facto* Chief Executive Officer of the SONOPCO Project is not supported by the record. Mr. McDonald did not report to Mr. Farley regarding GPC licensed activities. The items cited do not demonstrate that Mr. Farley exercised control over licensed activities at GPC's nuclear facilities during his involvement in the SONOPCO Project. Rather, the record shows that GPC controlled the daily operations of the Vogtle facility in accordance with a chain of command extending from the Vogtle General Manager, through the Vice President of the

¹¹⁷ If a significant event occurred at the Vogtle facility, Administrative Procedure VNS-EP-04, as it existed in 1990, required that the appropriate GPC corporate management be notified and briefed on the emergency. If any one of those to be notified were not available, the On-Call Project Manager would go to the next person up the line. On occasions, Mr. McCoy was unable to reach Mr. McDonald or Mr. Hairston, and he called Mr. Dahlberg. McCoy Supplemental at 3-4.

¹¹⁸ The administrative procedure did not *require* that Mr. Farley be contacted for significant events at the Vogtle facility, but in practice, both Mr. Farley and Mr. Dahlberg would be called. *Id.*

Vogtle facility, through the Senior Vice President–Nuclear Operations, through the Executive Vice President–Nuclear Operations, to the President and CEO of GPC. A Nuclear Operations Overview Committee of the GPC Board of Directors conducted periodic reviews of the regulatory and operational performance of GPC’s nuclear plants.

B. Establishing and Implementing Nuclear Policy Decisions

Intervenor’s Statement of Issues and the petition, as supplemented, include allegations that Mr. Farley controlled the Vogtle facility based upon his involvement with establishing and implementing nuclear policy decisions. (Issues 1, 9, 15, 17, and 20; October 1, 1990 Supplement at 4).

In Issue 15, Intervenor claims that, in 1987, Messrs. Addison and Farley met privately and agreed Mr. Farley would serve as “chief executive of Southern Company’s nuclear division” and decided to locate Southern Nuclear in Birmingham without the knowledge of senior GPC officials.

Intervenor’s assertion that Messrs. Addison (CEO of The Southern Company) and Farley agreed in 1987 that Mr. Farley would become the chief executive of a Southern Company nuclear operating subsidiary is not supported by the hearing record. Although he did not recall the exact date, Mr. Addison believed that his discussions with Mr. Farley about Mr. Farley heading the Southern Nuclear Operating Company occurred “when the decision was made to go forward.” Addison Deposition at 36-37. The hearing record shows that Mr. Addison did not make the decision unilaterally, that Mr. Farley was elected to the position of President and CEO of Southern Nuclear by the Board of Directors, which included GPC’s CEO (Mr. Dahlberg) and GPC’s Executive Vice President–Nuclear Operations (Mr. McDonald) after Southern Nuclear was incorporated on December 17, 1990. Hairston at 37. The fact that Mr. Addison, the CEO of the holding company, discussed with a senior officer the possibility of that officer heading a new subsidiary, does not violate any Commission regulation and does not support a conclusion that Mr. Farley directed GPC licensed activities.

There is no basis in the record to conclude that Messrs. Addison and Farley decided where SONOPCO would be located, or that this information was withheld from GPC management. While Mr. Farley told Mr. Addison that he would consider heading up Southern Nuclear if the corporate offices were in Birmingham, Mr. Addison discussed the merits of the location with GPC, the issue was examined by task forces, and Southern Nuclear was located in Birmingham, Alabama, due to its proximity to the engineering support staff and the economics of that location. Addison Deposition at 80-81, 83; Tr. 1821, 1823 (Farley). Mr. Thomas McHenry, the GPC Manager–Nuclear Support, represented GPC on the implementation task force, and Mr. H.G. “Grady” Baker, GPC Senior Executive Vice President, was on the steering committee. Tr. 1331

(McDonald). Mr. Farley believed that the decision as to location was made by the Board of Directors in May 1988. Tr. 1822-23 (Farley).

In Issues 15 and 17, Intervenor alleges that by 1989 the Southern Company Management Council began functioning as the SONOPCO Project Board of Directors. Intervenor states in Issue 17 that (1) GPC's April 1, 1991 response to the petition falsely stated that The Southern Company Management Council was not involved in operating issues pertaining to GPC's nuclear plants;¹¹⁹ and (2) the functioning of the Management Council was omitted from the April 1 response and the FSAR.

The record shows that there was no Board of Directors for the SONOPCO Project and no Board of Directors for Southern Nuclear until it was incorporated at the end of 1990. Tr. 1773-75 (Farley); Farley (ff. Tr. 1061) at 13-14; Dahlberg at 8. Individuals who later became members of the Board of Directors of Southern Nuclear informally discussed the status of efforts to form Southern Nuclear, and other issues of common interest, as representatives of The Southern Company Management Council. Farley at 13-14.

Mr. McCoy testified that The Southern Company Management Council is not described in the Vogtle FSAR because the Council is not the licensee of the Vogtle facility or an organization with responsibilities regarding the operation of the Vogtle facility. The Southern Company Management Council only reviewed GPC's budget in connection with The Southern Company's obligations to its stockholders. McCoy at 16. Neither 10 C.F.R. § 50.33 nor 10 C.F.R. § 50.34 requires that such budget review activities be included in an FSAR. Thus, there was no misrepresentation to the NRC and Intervenor's allegations in Issues 15 and 17 are without merit.

In Issue 1, Intervenor asserts that key negotiations between GPC and Oglethorpe Power Corporation were conducted by Mr. Farley.

Mr. Farley testified that he conducted certain negotiations with Mr. Stacey of Oglethorpe Power Corporation at the request of Mr. Dahlberg, but the major part of the negotiations were through Mr. Grady Baker and Mr. Fred Williams. Farley at 33; *see also* Dahlberg at 11-12. Mr. Williams confirmed that he was in charge of negotiating the agreement, and that in his view, Mr. Farley merely provided Oglethorpe Power Corporation information and comfort about setting

¹¹⁹ Intervenor mischaracterizes GPC's April 1 response. The response stated, at 4:

The Southern System Management Council provides a forum for the exchange of information among subsidiary companies that will aid the Companies' daily operations, it reviews system performance and it provides strategic and policy guidance to the system. However, day-to-day management of policy and operating issues pertaining solely to an individual subsidiary company is the exclusive responsibility of the subsidiary company's CEO.

Intervenor offered no evidence that showed the statement to be inaccurate.

up nuclear operating companies. Tr. 2482-83 (Williams). Thus, Mr. Farley's participation does not indicate control of GPC licensed activities.

Intervenor claims in Issue 1 that Mr. Farley reviewed data requests and testimony before the Georgia Public Service Commission in support of GPC's 1989 rate case. Mr. Farley testified that he had no responsibility for GPC's rate case and did not direct Mr. McDonald's activities related to the 1989 rate case. Tr. 1803 (Farley); Farley at 34-35. Mr. Farley's monitoring of data requests to make sure that the SONOPCO Project was providing expeditious support (Tr. 1803-11 (Farley)) does not reflect control over licensed activities.

In Issue 9, Intervenor alleges that GPC's April 1, 1991 response to the petition falsely stated that the resolution of a dispute between Messrs. Dwight Evans (GPC Executive Vice President-External Affairs) and McDonald by Mr. Dahlberg's direction to McDonald regarding the presentation of performance indicators to the Georgia Public Service Commission was evidence of the reporting relationship and indicative of who was in control of nuclear operations at the Vogtle and Hatch facilities. Intervenor claims that this statement is false because Mr. McDonald, after an August 10, 1989 meeting, did not follow Mr. Dahlberg's instructions, and Messrs. McDonald and Farley reviewed and approved testimony that did not include alternative performance indicators.

The hearing record does not support Intervenor's assertion that Mr. McDonald did not follow Mr. Dahlberg's instructions. Messrs. McDonald and Dahlberg both testified that a decision was made at the August 10, 1989 meeting to be prepared to propose alternative performance standards, if necessary, and that this strategy was carried out in the handling of the 1989 rate case. Prefiled Testimony of A. William Dahlberg, III, ff. Tr. 1061, "Dahlberg," at 17; McDonald at 15-17; Tr. 1102-22, 1137-41 (Dahlberg); Tr. 1504 (McDonald). Mr. Farley received copies of the draft testimony to be submitted to the Georgia Public Service Commission, but he neither approved nor disapproved it. Farley at 34. He was in agreement with Mr. Dahlberg's decision that GPC should be prepared to propose alternative performance standards, if necessary. Tr. 1108-09 (Dahlberg). Such actions do not indicate control of nuclear operations or budget policy.

In Issue 20, Intervenor claims that in its April 1, 1991 response to the petition (at 12, Attachment I), GPC inaccurately states that Mr. Farley did not create the outage philosophy¹²⁰ for the Vogtle facility. Intervenor asserts that the response is inaccurate because (1) Mr. Farley was involved in the establishment of the outage philosophy at the Vogtle facility, (2) Mr. McCoy referred to Mr. Farley's

¹²⁰ As used here, "outage philosophy" refers to outage scheduling. Specifically, the "philosophy" was to use an "optimum" schedule — a schedule without the inclusion of time for contingencies. McCoy at 14-15.

role as indicated by an audio tape (Tape No. 236) recorded in August 1990,¹²¹ and Mr. Farley testified during a deposition that “Farley staff meetings” were held every week.

The record shows that GPC’s April 1, 1991 response to the petition was accurate because Mr. McDonald established and implemented the outage philosophy and Mr. Farley was not involved in overseeing the establishment of the outage philosophy. McDonald at 13; Tr. 1518-20 (McDonald); McCoy at 14; Farley at 30. Mr. McCoy’s statements on Tape No. 236 referred only to “discussions” about the outage philosophy that included Mr. Farley, and do not show that Mr. Farley set, established, directed, or created the outage philosophy at the Vogtle facility. McCoy at 14. Mr. Farley testified that he (1) did not direct the operating philosophy and other executive matters concerning operation of the Vogtle facility in the weekly staff meetings, (2) did not have any authority to control, and (3) did not attempt to exercise any control over management decisions affecting licensed activities or personnel matters concerning the Vogtle facility. Farley at 22. Moreover, Mr. Mosbaugh admitted that he had no personal knowledge to support his claim that the outage philosophy came from Mr. Farley. Tr. 2129-35.

Accordingly, the hearing record does not support Intervenor’s allegation in Issue 20 that GPC’s April 1, 1991 statement is inaccurate or that Mr. Farley controlled operation of the Vogtle facility by establishing or implementing the Vogtle outage or other operational philosophy.

In summary, the hearing record shows that nuclear policy decisions for the Vogtle facility were established and implemented by GPC, and there was no evidence that Mr. Farley established the outage philosophy or any other operational policies for the Vogtle facility. Mr. Farley’s limited involvement in a 1989 rate case matter before the Georgia Public Service Commission (i.e., his review of draft testimony regarding alternative performance standards) does not indicate any control of GPC’s nuclear operations or licensed activities. Intervenor also provided no information that The Southern Company Management Council acted as the SONOPCO Project board of directors until the Project was incorporated.

C. Employing, Supervising, and Dismissing Nuclear Personnel

In his Statement of Issues and the petition, as supplemented, Intervenor asserts that Mr. Farley exercised control over the Vogtle facility because he (1) selected and approved GPC’s management staff; (2) reviewed nuclear personnel in 1989 as evidenced by GPC Management Council’s exclusion of nuclear personnel

¹²¹This issue is also raised in the October 1, 1990 Supplement to the Petition at 4. Petitioners claim that Mr. McCoy’s taped statement, that the outage philosophy was created by Mr. Farley and others, supports their assertion that Vogtle project management assumed that Mr. Farley, not Mr. Dahlberg, controlled Vogtle’s operations.

from its 1989 companywide review of management; (3) decided that Mr. Michael Barker, a GPC employee, would not be transferred from the SONOPCO Project to the Nuclear Operations Contract Administration (NOCA) group in Atlanta; (4) prepared Mr. McDonald's annual performance appraisal; and (5) implemented changes in Vogtle personnel evaluations and pay. (Issues 1, 6, 8, 14A, 14B, 15, 19, 21, 27, and 28; October 1, 1990 Supplement to Petition at 1-3.)

The hearing record fails to support Intervenor's allegation (Issues 6 and 15; October 1, 1990 Supplement to Petition at 1-2) that Mr. Farley selected and approved GPC management staff. The decision to select the individual officers responsible for GPC's nuclear operations was made by GPC management with the approval of the GPC Board of Directors. GPC's Vice President, Grady Baker, and not Mr. Farley, recommended that Messrs. McDonald and Hairston become officers of GPC. Mr. Farley's involvement in the selection of Messrs. McDonald and Hairston was limited to concurring as President of APC that they could take on the additional responsibilities associated with managing GPC's nuclear facilities. Farley at 25-26.

Mr. Farley's involvement in the hiring of Mr. McCoy consisted of discussing Mr. McDonald's proposal to hire him. McDonald at 10-11; Farley at 25-26; McCoy at 5-6; Tr. 1349-50 (McDonald) and Tr. 1727 (Hairston). GPC's CEO, Mr. Robert W. Scherer, interviewed Mr. McCoy before he was appointed, and the GPC Board of Directors subsequently appointed Mr. McCoy to his current position. McCoy at 1, 5, and 6.

Mr. Farley was involved in the selection of Messrs. McCrary and Long as Vice Presidents in SCS. As President of APC, Mr. Farley was consulted on the appointments of Messrs. McCrary and Long because the Farley nuclear facility was being supported by the SONOPCO Project and SCS officers. Hairston at 24. Mr. Farley was a member of a selection committee, including GPC and APC representatives, to make recommendations for the Vice President of Administrative Services position.¹²² Mr. McDonald and Mr. Jack Causey of GPC were also members. Tr. 1276 (McDonald).

Thus, the hearing record supports the conclusion that Mr. Farley did not make decisions regarding the hiring of any of the officers reporting to Messrs. McDonald and Hairston. Mr. Farley's limited involvement with SCS officers within the SONOPCO Project (such as Messrs. McCrary and Long) does not appear inappropriate since the SONOPCO Project and its SCS officers were also providing support to the APC nuclear plant.

Intervenor's assertion (Issues 1 and 14A) that the GPC Management Council's exclusion of nuclear personnel from its 1989 companywide review of management was evidence that nuclear operations were reviewed by Mr. Farley was

¹²² Typical of the selection process for SCS senior personnel, no selection committee was convened for Mr. Long because his functions and position in the SONOPCO Project were similar to his position in SCS. Farley at 22-24.

not supported by the hearing record. Mr. Dahlberg testified that the nuclear management was not included because the nuclear officers had just been reviewed as part of the recent formation of the SONOPCO Project. Tr. 1185-88 (Dahlberg).

Intervenor asserts (Issue 8) that GPC's August 24, 1994, response to a Licensing Board question¹²³ was inaccurate because it failed to identify the NOCA group as an organization that had oversight responsibilities within GPC, failed to state that SONOPCO Project personnel refused to cooperate with NOCA, and that SONOPCO personnel, including Mr. Farley, interfered with the operation, staffing, and existence of NOCA.¹²⁴

The hearing record indicates that NOCA never performed the type of oversight functions identified by the Board's question. NOCA did not perform any oversight function regarding licensed activities and the people assigned to NOCA were not qualified to perform oversight of licensed activities. Tr. 2565-76, 2579, 2588-89, 2596 (McCoy); Tr. 1238 (Dahlberg); McCoy Rebuttal at 3. While NOCA was, in part, formed by Mr. Dahlberg to monitor the performance of GPC's nuclear plants, it was later determined that its data-gathering function duplicated activities of SONOPCO Project personnel reporting directly to Mr. Dahlberg. Dahlberg at 13; Tr. 1193 (Dahlberg).

Mr. Hobby, who was General Manager of NOCA, testified that employees in the SONOPCO Project refused to cooperate in supplying him information regarding the plants, and prevented him from hiring the employees needed to perform NOCA's intended function. Mr. McDonald viewed NOCA as an impedance in the GPC chain of command and admitted that he did not cooperate with NOCA because he felt Mr. Hobby was attempting to act as an intermediary between Mr. McDonald and Mr. Dahlberg. Tr. 1483 (Dahlberg); *see also* Tr. 1485 (McDonald). Mr. McDonald's concern as a GPC official regarding the GPC chain of command does not constitute transfer of control of licensed activities at GPC nuclear facilities. Furthermore, Mr. Hobby lacked any personal knowledge that Mr. Farley directed or otherwise influenced Mr. McDonald's actions regarding NOCA. Tr. 2352-57 (Hobby).

Intervenor's claim that Mr. Farley interfered with the staffing of NOCA by deciding that Mr. Michael Barker, a GPC employee, could not be transferred from the SONOPCO Project to the NOCA group in Atlanta (Issue 8; October 1, 1990 Supplement to Petition at 3-4), was not substantiated. Mr. Hobby admitted that only GPC employees attempted to prevent him from interviewing

¹²³ Question 2 in the Board's Memorandum and Order, dated May 25, 1994, asked what organizational units or executive personnel of GPC had any form of oversight activity over the SONOPCO Project, such as "managerial control, audits, investigation, personnel, quality assurance or control, or root cause assessments."

¹²⁴ Petitioners assert that Mr. William Evans, a GPC Corporate Concerns investigator, told Mr. Hobby that Mr. Farley would be "making the call" as to whether Mr. Hobby could interview a SONOPCO candidate for the NOCA performance engineer position. October 1, 1990 Supplement to Petition at 3-4.

Mr. Michael Barker for the NOCA performance engineer position. Tr. 2360-61 (Hobby). The hearing record shows that Messrs. Hairston and McDonald, both GPC officers, opposed this transfer because they believed that the NOCA position had been assigned an inflated rating. Tr. 1737-38 (Hairston); Tr. 1490-94 (McDonald). Mr. Barker discussed his transfer directly with Mr. Dahlberg. Tr. 1222-23 (Dahlberg). Neither Mr. Farley nor Mr. Grady Baker could recall any discussion of Mr. Barker on May 5, 1989, with Mr. Dahlberg regarding Mr. Hobby's proposed additions to the NOCA staff. Tr. 1759-60, 1820-21 (Farley); Baker DOL Deposition at 41. Mr. Hobby's belief about Mr. Farley's interference was based on information from individuals who did not attend the May 5, 1989 meeting. Hobby at 41; Evans Deposition at 17-18. Accordingly, the assertion that Mr. Farley "made the call" is not supported by the record.

Mr. Farley did tell Mr. Dahlberg on or about May 5, 1989, that some organizations in The Southern Company system, such as NOCA, were duplicative (Farley at 32-33; Tr. 1756 (Farley)), but Mr. Dahlberg came to the same conclusion without Mr. Farley's input. Dahlberg at 13; Tr. 1228 (Dahlberg); Tr. 2461, 2497-2504 (Williams). Thus, Mr. Farley's action did not convey a command, or constitute control, over GPC personnel matters.

Intervenor's claim (Issue 1) that Mr. Farley prepared Mr. McDonald's annual performance appraisal was not substantiated. The record shows that Mr. McDonald's annual performance appraisal was prepared by Messrs. Harris and Dahlberg, the respective CEOs of APC and GPC. Although Messrs. Harris and Dahlberg gave Mr. Farley a chance to comment on the review, Mr. Farley did not know what was finally done. Tr. 1861-62 (Farley).

The record does not substantiate Intervenor's claims (Issues 1 and 6) that Mr. Farley implemented changes in personnel evaluations and pay with respect to Vogtle nuclear operations. The record shows that Mr. Farley did not implement changes to personnel evaluations or pay policy for Vogtle nuclear operations personnel. Mr. Farley explained the new Southern Company systemwide policies and answered questions on them. Farley at 31. This involvement was appropriate for his position as a Southern Company officer and did not constitute control over licensed activities of GPC's nuclear facilities.

As an Executive Vice President of The Southern Company, Mr. Farley addressed nuclear plant employees to brief them on the systemwide changes being made to the incentive pay programs of all of the operating companies. At that time, he also polled employees about any concerns they had with their employment situation. Such systemwide activities are typically performed by a representative of The Southern Company. McDonald at 17-18. These activities do not constitute improper control of GPC personnel or NRC-licensed activities.

In summary, the record does not show that Mr. Farley controlled GPC nuclear facilities by employing, supervising, and dismissing nuclear personnel,

or that GPC provided inaccurate information to the NRC regarding Mr. Farley's involvement with personnel matters.

D. Controlling Costs

In his Statement of Issues, Intervenor alleged that Mr. Farley's control of GPC nuclear facilities is shown through budget and personnel pay matters in that (1) Southern Nuclear, its predecessor, and The Southern Company controlled GPC's nuclear budget since November 1988; (2) Mr. Farley implemented changes in personnel evaluations and pay for Vogtle nuclear operations personnel; and (3) the GPC Management Council did not review GPC's 1990 nuclear operating budget. Intervenor asserts that inaccurate and incomplete information was provided to the NRC regarding GPC's control of budget and personnel pay matters. (Issues 1, 6, 12, 14A, 14B, and 17.)

Intervenor alleged in Issue 6 (*see also* Issues 1 and 12) that GPC's budget had been under the control of Southern Nuclear since November 1988, and thus the March 28, 1991 Vogtle FSAR amendment revising Chapter 13 inaccurately states that (1) the GPC Executive Vice President–Nuclear Operations reports to GPC's President and CEO with respect to all matters concerning budgets, and (2) Southern Nuclear matters are currently limited to operational support activities.

Intervenor's allegation regarding budget control is based upon his opinion that GPC's 1990 budget was approved by Mr. Farley and later by Mr. Addison over Mr. Dahlberg's objection. Testimony of a number of witnesses about GPC's 1990 budgeting process, and subsequent nuclear budgets, shows that GPC retained control of its nuclear budgets. GPC's 1990 (and later) nuclear budgets were reviewed by the Presidents of APC (Mr. Harris), GPC (Mr. Dahlberg), SCS (Mr. Franklin), The Southern Company (Mr. Addison), The Southern Company Executive Vice President–Nuclear (Mr. Farley), the Executive Vice President–Nuclear Operations of GPC and APC (Mr. McDonald), probably the Senior Vice President–Nuclear Operations of GPC and APC (Mr. Hairston), probably the nuclear plant project Vice Presidents (Messrs. McCoy, Beckham, and Woodard), and probably the SONOPCO Project Assistant Comptroller (Mr. Gilbert). Dahlberg at 9. The SONOPCO group presented the 1990 budgets recommended by Messrs. Hairston and McDonald for all three GPC nuclear facilities to Mr. Addison and his staff during a December 1989 meeting in Birmingham, Alabama. Mr. Addison then visited each of the operating groups and received a report on their budgets from Mr. McDonald, Mr. Hairston, and the project vice presidents. Farley at 28-29; Tr. 1392-94, Tr. 1405-06 (McDonald). The proposed budgets for the three nuclear facilities were then submitted to the operating companies, APC and GPC. Mr. Dahlberg received, from the GPC Management Council, the portion reflecting GPC's nuclear plants

for incorporation into the overall GPC budget and for approval. Budget approval was then given by GPC's CEO for the GPC capital and operating budgets, and by the GPC Board of Directors for the capital budget. After approval by GPC, the total GPC budget was submitted to The Southern Company. Dahlberg at 9; Tr. 1240-41 (Dahlberg); McDonald at 14-15. GPC Management Council reviewed the 1990 GPC nuclear budgets, as part of the total GPC budget, before they were approved by Mr. Dahlberg. The capital budget was also approved by the GPC Board of Directors. Dahlberg at 10.

Mr. Farley's involvement was limited to reviewing the budgets as an Executive Vice President of The Southern Company and advising Mr. Addison, who was responsible for the review of all operating company budgets. Dahlberg at 10; Tr. 1779-83, 1795 (Farley). Mr. Dahlberg determined whether the 1990 budget was acceptable. Farley at 27. Mr. Addison had never, however, approved or disapproved GPC's budget over Mr. Dahlberg's objection. Dahlberg at 11.

The review of budgets of subsidiaries by holding companies (e.g., The Southern Company) to ensure that the budgets of the operating companies were reasonable and appropriate is not unusual or indicative of a transfer of control.¹²⁵

Accordingly, the hearing record does not support Intervenor's assertion that Southern Nuclear controlled GPC's budget. Therefore, there is no support for Intervenor's claim that GPC inaccurately stated that (1) the GPC Executive Vice President—Nuclear Operations reports to GPC's President and CEO with respect to all matters concerning budgets, and (2) Southern Nuclear matters are currently limited to operational support activities. The record supports a conclusion that Southern Nuclear matters are limited to operational support activities.

Intervenor asserts in Issue 14A that GPC's April 1, 1991 response to the petition is false in stating that the GPC Management Council functioned as a policy-setting body and made corporate resource allocation decisions because, in late 1989, the GPC Management Council did not participate in the review of GPC's 1990 nuclear operating budget. The hearing record, however, showed that Intervenor's assertion was incorrect in that the GPC Management Council did review the 1990 nuclear budget as part of the total GPC budget review before approval by Mr. Dahlberg. See Tr. 1396-98, 1403; Dahlberg at 10.

Intervenor claims in Issue 14B that in the April 1, 1991 response to the petition, GPC misrepresents that Mr. McDonald reported periodically to the

¹²⁵ The review of GPC's budget by The Southern Company Management Council in connection with The Southern Company's obligations to its stockholders is not an activity that need be described in the Vogtle FSAR, and its omission does not warrant the conclusion that GPC's April 1, 1991 response to the petition was inaccurate as Intervenor asserts in Issue 17.

GPC Management Council regarding matters such as budgets and organizational goals.

Mr. McDonald testified that he reported to the GPC Management Council on nuclear operating matters, including budget matters, with the qualification that "reported" meant "provided budgets for their review."¹²⁶ Organizationally, he reported only to the GPC CEO. McDonald at 14. In view of Mr. McDonald's testimony, the hearing record does not support a conclusion that GPC's April 1, 1991 response was inaccurate.

In summary, the hearing record does not support a conclusion that GPC misrepresented its budgets affecting the operation of GPC licensed facilities. There is no indication in the hearing record that the particular process GPC used to develop its budget is dispositive to Intervenor's assertion that Mr. Farley, The Southern Company, or SONOPCO Project controlled the operation of the Vogtle facility. Rather, the record shows that GPC was responsible for the costs of the Vogtle facility. After review by GPC's Management Council, the operating and capital budgets were approved by GPC's President and CEO, and the capital budget was also approved by the GPC Board of Directors. The record does not support the conclusion that Messrs. Farley and Addison approved GPC's nuclear budgets. As an Executive Vice President of The Southern Company, Mr. Farley was involved in reviewing the nuclear budgets as part of the normal process for preparing annual budgets in the Southern system. Given The Southern Company's holding company status, Mr. Addison's involvement in reviewing and providing guidelines and requirements for adequate earnings and reasonable capital needs was appropriate.

II. OTHER ALLEGED INACCURACIES COMMUNICATED TO NRC

Intervenor's Statement of Issues and the petition contain assertions that GPC managers provided inaccurate or incomplete information to the NRC when describing its organization and plans to form Southern Nuclear, and when responding to the petition. The alleged misrepresentations or omissions regard statements about (1) the Vogtle chain of command, (2) Mr. Dahlberg's

¹²⁶ Meeting minutes show that Mr. McDonald participated in Management Council meetings about the 1989 and 1990 budgets on September 23 and October 14, 1988, and presented organizational goals for the Vogtle and Hatch facilities during a December 7, 1988 meeting. Intervenor Exh. 135 (meeting minutes) at 27, 29-30, 42-43. Mr. McDonald attended a July 25, 1989 meeting during which the 5-year capital budget targets were approved, and the schedule for budget reviews, including Management Council review and Mr. Addison's review, was agreed upon. Intervenor Exh. 135 at 71-73. The Management Council also considered nuclear budgets during meetings on November 6 and 14, and December 4, 1989. Intervenor Exh. 135, at 90, 93-96, 97 (capital budget), 98, 104-16 (nuclear update).

relationship with Vogtle site management, (3) Mr. Farley's responsibilities as Executive Vice President–Nuclear of The Southern Company, (4) the 1989 title of Mr. Dahlberg, (5) SONOPCO Project's control over the Vogtle facility since November 1988, (6) the composition of the GPC Management Council, and (7) the title held by Mr. Farley in 1988. (Issues 1, 2, 12, 13, 18, 19, and 26-28.)

The hearing record regarding the alleged illegal license transfer issue does not support that GPC concealed an unauthorized role of Mr. Farley or a *de facto*, unauthorized organization for control of GPC nuclear facilities.

In Issue 1 (*see also* Section 2.206 Petition § III.2; and July 8, 1991 Supplement § III), Petitioners stated that GPC misled the Commission about the chain of command from the Vogtle Project's Plant Manager (i.e., the General Manager) to its CEO before the NRC issued the operating license for the facility.

On March 30, 1989, the Commissioners met to discuss and possibly vote on the full-power operating license for Vogtle Unit 2. Commissioner Carr expressed concern about the hierarchy between the Vogtle Plant Manager and the Chief Executive Officer (CEO), noting that it "looked to me like he was a long way from the CEO." Mr. R.P. McDonald, GPC Executive Vice President–Nuclear Operations, responded that (1) he (Mr. McDonald) reported to Mr. A. William Dahlberg, the GPC CEO; (2) that Mr. Ken McCoy, Vice President of Vogtle, reported to Mr. McDonald; and (3) that Mr. George Bockhold, then Vogtle General Manager, reported directly to Mr. McCoy. At the conclusion of the meeting, the Commissioners voted unanimously in favor of the license, and the license was issued the following day.

On May 1, 1989, Mr. W.G. Hairston, III, Senior Vice President for Nuclear Operations, sent the NRC a letter of correction of the transcript, noting that Mr. McDonald had "inadvertently left out the Senior Vice President of Nuclear Operations. The organization is as described on figures 13.1.1-1 and 13.1.1-2 of the Vogtle Final Safety Analysis Report."

The Petitioners claim that Mr. McDonald knowingly made false statements to the NRC Commissioners in the presence of Messrs. Dahlberg, McCoy, and Bockhold during his response to then Commissioner Carr in that he "eliminated one entire level of management between the plant manager and the CEO." Moreover, the Petitioners asserted (Petition at 8) that:

Messrs. Dahlberg, McCoy and Bockhold should have known that Mr. McDonald's statements were false and should have brought this to the immediate attention of the Commission and otherwise corrected the record *before* the Commission acted on the Vogtle full-power license request.

In its Response to the Petition of April 1, 1991, GPC noted that the Commission had been apprised of the Company's organization before the meeting on March 30, 1989, including the Senior Vice President position, by

an amendment to the Vogtle FSAR that was submitted November 23, 1988. The amendment described the reporting chain as being from Mr. McCoy to Mr. Hairston to Mr. McDonald. GPC's Response also indicated that the NRC had reviewed the organizational structure in December 1988 and issued an inspection report.¹²⁷ In the inspection report, the NRC stated that the vice presidents of the Farley, Hatch, and Vogtle facilities reported to the Senior Vice President, who reported to the Executive Vice President, and that the organization for Vogtle was consistent with the Vogtle FSAR amendment submitted in November 1988.

In its April 1, 1991 Response, GPC also noted that, during the March 30 meeting, Commissioner Rogers stated that he had reviewed the Company's organizational chart during his visit to the plant site. In addition, GPC noted that it had submitted the letter of correction to the transcript approximately 2 weeks after receiving the NRC transcript.

The NRC Staff has reviewed this issue and concludes that Mr. McDonald's reply to then Commissioner Carr was inaccurate in that the transcribed record clearly contradicted other documents of record, including the FSAR and NRC inspection reports. The inaccuracy was material in that the reply (1) was in direct response to the Commissioner's stated concern regarding an organizational structure in which the plant manager appeared to be "a long way from the CEO," (2) could have influenced the Commission's decision, and (3) could have been considered by the Commission in reaching its decision.

There was no apparent motive for Licensee and its employees to attempt to deliberately mislead the Commissioners since the Licensee had previously provided correct information, and NRC Staff members were present who knew the correct information.¹²⁸ The NRC Staff does not view Mr. McDonald's inaccurate statement or omission as intentional or significant in that it is unlikely the statement would have caused the Commission to reach a different decision. No enforcement action was taken regarding the omission of Mr. Hairston in the organizational structure.

In summary, while inaccurate information was initially given to the Commissioners, it appears to have been inadvertent, it was corrected by the Licensee upon discovery, and the NRC Staff was already aware of the correct information. Under the NRC's Enforcement Policy (NUREG-1600), unsworn oral statements that are unintentionally inaccurate are not normally acted upon unless they involve significant information by a licensee official. While the Licensee should have corrected the material omission either during or immediately following the meeting, further action regarding this omission is not warranted due to its mi-

¹²⁷NRC Inspection Report Nos. 50-321/88-41, 50-366/88-41, 50-424/88-60, 50-425/88-77, 50-348/88-33, and 50-364/88-33, dated February 7, 1989.

¹²⁸Mr. John Rogge, the NRC's Senior Resident Inspector for the Vogtle facility at the time, attended the meeting with the Commissioners in Washington, DC, and testified during the Phase I hearing that at that meeting he was aware that Mr. Hairston was in the Vogtle chain of command. Tr. 2731 (Rogge).

nor significance and because no information other than the Petitioners' opinion exists to support the position that the omission was intentional.

Intervenor also alleges (*see* Issue 1) that GPC falsely stated during the March 30, 1989 meeting with NRC that Mr. Dahlberg had a "personal hands on" relationship with the management at the plant site. The meeting transcript (Intervenor Hearing Exh. 17), at page 5, indicates that Mr. Dahlberg described GPC's upper management as being accessible. The record shows that Mr. Dahlberg visited plants periodically and the Vogtle facility at least twice in 1989, and was involved in nuclear operations. His "hands-on" management style referred to his oversight, his daily communications with the nuclear management, his plant visits, and his willingness to take calls periodically from the site. Intervenor Hearing Exh. 32, at 4, 15; McCoy at 6-7; Tr. 1153-59 (Dahlberg). Therefore, the record does not support the allegation that the statement was inaccurate.

In Issue 2, Intervenor states that Mr. Hairston's letter of May 1, 1989, to the NRC correcting the Unit 2 full-power license hearing transcript was inaccurate in asserting that an attached FSAR Figure 13.1.1-1 as amended November 23, 1988, accurately depicted the corporate management structure for the Vogtle facility since the figure did not portray Mr. Farley's role, and indicated that Mr. McDonald (the "Executive Vice President-Nuclear Operations" position) reported to Mr. Scherer ("Chairman and CEO"), rather than to Mr. Dahlberg ("President").

In December 1988, Mr. Scherer relinquished his position as CEO and Mr. Dahlberg became CEO, but not Chairman. Thus, Intervenor is correct inasmuch as FSAR Figure 13.1.1-1 had not been updated to reflect this change of title. Moreover, the figures attached to the May 1, 1989 letter should have shown the Executive Vice President-Nuclear reporting to the "President and CEO," which was Mr. Dahlberg's correct title.

Mr. Hairston testified during the transfer hearing that the only purpose of the May 1, 1989 letter was to correct Mr. McDonald's omission of Mr. Hairston's role during the Unit 2 full-power hearing. Mr. Hairston and others did not notice the outdated title in the CEO box. Hairston at 29.

Messrs. Allenspach and Rogge, who participated in NRC Staff's review of the organizational structure for the full-power licensing of Vogtle Unit 2 and the related inspection of the organizational structure in December 1988, testified that the focus of the NRC Staff's review of the organization in control of the Vogtle facility was at Mr. Hairston's level and lower and they attached no particular significance to the organizational structure represented at levels above Mr. Hairston. Tr. 2678-80, 2698.

There is no evidence that Mr. Hairston's explanation regarding the outdated title in the CEO box was inaccurate or that the NRC was misled in any significant manner by this oversight. In addition, as discussed in Section III.B.1.a of this Director's Decision, Mr. Farley was not in the Vogtle chain of command.

In summary, while Intervenor is correct that FSAR Figure 13.1.1-1 did not accurately reflect Mr. Dahlberg's title of "President and CEO," the error was not a significant factor in the NRC Staff evaluation of the information, and there is no evidence that it misled the NRC. The record does not support Intervenor's assertion that the figure is also inaccurate because it failed to reflect Mr. Farley's role in the control of the Vogtle facility.

In Issue 12, Intervenor claims that in the April 1, 1991 response to the petition, GPC misstated Mr. Farley's responsibilities as Executive Vice President-Nuclear as including:

(1) overseeing the formation of Southern Nuclear, (2) acting as spokesman for Southern Nuclear among chief executive officers of the other Southern Company affiliates,¹²⁹ and (3) representing the Southern Company on the national scene concerning generic nuclear power issues.

Mr. McDonald testified that this description is an accurate reflection of Mr. Farley's duties as described in Mr. McDonald's letter agreement dated April 24, 1989, with Mr. Franklin of SCS. The description is consistent with the NRC Staff's historical knowledge of Mr. Farley's activities and duties. The hearing record provides no substantive evidence to the contrary.

In Issue 13, Intervenor asserts that, because Mr. Hairston was a GPC Senior Vice President in April 1991 and had never been a member of the GPC Management Council, GPC's April 1, 1991 response to the petition falsely states that the "GPC Management Council is made up of all the Executive and Senior Vice Presidents of GPC."

Intervenor is correct with respect to Mr. Hairston and the error was admitted in the hearing testimony. McDonald at 13-14; Tr. at 1075-77 (Dahlberg); Tr. 1442-43 (McDonald). There is no evidence that the error was anything other than a simple oversight. The primary focus of the statement, that Mr. McDonald was on the Management Council and Mr. Farley was not, is correct. The NRC was not significantly misled by the error with respect to Mr. Hairston.

In Issue 28, Intervenor alleges that the April 1, 1991 GPC response to the petition falsely states that Mr. Farley's role in the selection of personnel for the SONOPCO Project was proper in that "Mr. Addison requested such assistance from Mr. Farley and such assistance fell within his duties as Executive

¹²⁹The response to the petition stated that this function refers to Mr. Farley's membership on the Southern System Management Council.

Vice President–Nuclear of The Southern Company.” Intervenor claims that this statement is false because the staffing selections were made in 1988 and Mr. Farley did not become Executive Vice President–Nuclear of The Southern Company until March 1, 1989.

Intervenor is correct. Mr. McDonald admitted that, technically, the April 1, 1991 response to the petition was inaccurate in stating that staffing selections made in 1988 were within Mr. Farley’s duties as Executive Vice President–Nuclear of The Southern Company since Mr. Farley had not yet assumed that position in 1988. McDonald at 12.

The error was not significant or intentional because the same page of the April 1, 1991 response (Intervenor Exh. 48, at 9) indicated Mr. Farley’s correct title in 1988, i.e., President of APC.

In Issue 26, Intervenor alleges that in a September 4, 1992 license amendment application, GPC omitted facts pertaining to the actual configuration and operation of the Vogtle facility in stating that in January 1991, Southern Nuclear began providing nuclear support services, technical services, and administrative services but omitting reference to the SONOPCO Project’s “control over the nuclear operations of plants Vogtle, Hatch, and Farley [which] began in November 1988” prior to Southern Nuclear’s incorporation. Mr. Hairston testified that the September 4, 1992 statement regarding Southern Nuclear was accurate and that Southern Nuclear was incorporated on December 17, 1990, and became effective January 1, 1991. Hairston at 46.

The license amendments application is consistent with information received by the NRC during the late 1990–early 1991 time frame and the NRC was well informed of the phased approach employed by GPC to establish a nuclear operating company through various meetings, inspections, and discussions. Intervenor provided no evidence that the services provided by SONOPCO Project from November 1988 until Southern Nuclear’s incorporation constituted control over operations or licensed activities for GPC or APC nuclear facilities. Accordingly, there is no evidence that the license amendments application of September 4, 1992, was inaccurate or misled the NRC.

In Issue 19, Intervenor states that in its October 3, 1991 response to the section 2.206 petition as revised July 8, 1991, GPC falsely states that (1) the selection process used in 1988 for the staffing of SONOPCO was not completed during the two-day meeting of SONOPCO Project executives, and (2) Mr. McDonald “never purported to give an unqualified or rigid top-down characterization of how the organization was staffed.”

Messrs. McCoy and McDonald testified that while a number of individuals were identified as the most likely candidates for positions within the SONOPCO Project during that two- or three-day meeting, the selection process continued beyond the meeting. McCoy at 16; McDonald at 11; Tr. at 1301 (McDonald).

Mr. McDonald testified that the selection process involved Mr. McCoy and Mr. J.T. Beckham (Vice President of the Hatch facility) starting at the top of the organization and, using a blank organization chart, identifying prospective candidates who were most qualified for positions in the organization. Selected managers then participated in selecting those individuals who would be working for them. He only recalled that they settled on the top tier during the meeting, although they may have penciled in other names, and the other candidates were shuffled around for a couple of weeks. Tr. at 1301, 1304-08 (McDonald).

Given Mr. McDonald's description of the selection process, the hearing record does not support the conclusion that the statement regarding GPC's October 3, 1991 statement is inaccurate or misleading.

In Issue 27, Intervenor alleges that GPC's October 3, 1991 response to the petition inaccurately states that Mr. McDonald's testimony concerning the selection of Messrs. McCrary and Long given in the *Yunker and Fuchko* DOL proceeding was not inconsistent with his testimony in the *Hobby* DOL proceeding.

During the licensing transfer hearing, Mr. McDonald testified that his answers were different, and were not contradictory, because the questions were different. In the *Yunker and Fuchko* proceeding, when he was asked who selected Messrs. McCrary and Long for their positions in the SONOPCO Project, he understood the question to be who was ultimately responsible for referring them to the Board of Directors, and he replied he was not sure but assumed it was the President of Southern Company Services. In the *Hobby* case, he was asked if he was "involved" in selecting them and, since he had been involved with recommending them, gave an affirmative reply. McDonald at 11-12.

In light of the differences in the questions posed, the evidence does not support the conclusion that GPC's response of October 3, 1991, is inaccurate.

III. CONCLUSION

The record shows that GPC provided some inaccurate or incomplete information to the NRC when describing its organization and plans to form Southern Nuclear, and when responding to the petition. This information involved (1) the omission of Mr. Hairston when Mr. McDonald described the Vogtle chain of command during a March 30, 1989 meeting; (2) a 1989 FSAR organizational chart showing the position of Mr. Dahlberg as "Chairman and CEO" rather than "President and CEO"; and (3) GPC's April 1991 written response to the petition indicating that the GPC Management Council included all Senior Vice Presidents (which was inaccurate because Mr. Hairston was not a member), and indicating Mr. Farley's title in 1988 to be Executive Vice President-Nuclear of The Southern Company (a position he did not assume until March 1, 1989).

This inaccurate or incomplete information was not significant in terms of NRC focus on nuclear operations and licensed activities or in the context of the overall correct information provided to the NRC, and did not mislead the NRC. Thus the inaccuracies and omissions are not sufficient to warrant NRC enforcement action or conclusions that (1) GPC concealed an unauthorized role of Mr. Farley or a *de facto*, unauthorized organization for control of GPC nuclear facilities; or (2) GPC lacks the requisite character and integrity to be a licensee.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF ENFORCEMENT

James Lieberman, Director

In the Matter of

**WESTINGHOUSE ELECTRIC
CORPORATION**
(Madison, Pennsylvania)

March 20, 1997

The Director, Office of Enforcement, has taken action with regard to a petition filed by Shannon Doyle requesting that the Commission take action with regard to Westinghouse Electric Corporation. The Petitioner requested that the Commission investigate allegations that Westinghouse willfully provided false information to the Department of Labor (DOL), institute a show-cause proceeding pursuant to 10 C.F.R. § 2.202, and/or impose a civil penalty upon Westinghouse. The Petitioner had asserted, as a basis for his request, that Westinghouse had failed to correct the DOL record and provided material false statements to the DOL Administrative Law Judge in a case arising under the Energy Reorganization Act. In denying the petition, the Director determined that the matter should be referred to the DOL Administrative Review Board for its consideration.

TECHNICAL QUALIFICATIONS: REQUIREMENTS

The NRC generally does not have specific requirements for qualification and training of health physics technicians.

NRC: JURISDICTION

The NRC and DOL have complementary responsibilities in the area of employee protection.

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

I. INTRODUCTION

On May 30, 1996, Mr. Shannon Doyle (Petitioner) filed a petition pursuant to 10 C.F.R. § 2.206 requesting that the Nuclear Regulatory Commission (NRC) take immediate action against Westinghouse Electric Corporation (Westinghouse). Specifically, the Petitioner requested that the NRC investigate allegations that Westinghouse has willfully provided false information to the Department of Labor (DOL), and institute a show-cause proceeding pursuant to 10 C.F.R. § 2.202 and/or impose a civil penalty upon Westinghouse.

As a basis for his request, the Petitioner asserted, among other things, that Westinghouse had failed to correct the record and, through its counsel, had provided material false statements to the DOL Administrative Law Judge (ALJ) in a case arising under the Energy Reorganization Act (ERA), 89-ERA-022. Specifically, the Petitioner asserted that Westinghouse: (1) "knowingly let remain the false impression of the Administrative Law Judge that registration with the National Registry of Radiation Protection Technologists (NRRPT) is a requirement for the holding of the position of health physics technician in the nuclear power industry"; and (2) "purposely maintained this false impression by providing through its counsel false material statements in maintaining that an NRRPT filing to the USNRC 'establishes that a passing score on the registration test is required for the position of health physics technician.'"

By a letter dated August 16, 1996, I informed the Petitioner that, pursuant to section 2.206, the petition had been referred to me. I also informed the Petitioner that his request for immediate action had been denied, but that as provided by section 2.206, action would be taken on his request within a reasonable time. To address the concerns in the petition, I also requested in my August 16, 1996 letter that the Petitioner provide further information supporting the petition. In addition, by a separate letter to Westinghouse dated August 16, 1996, I requested from Westinghouse a response to certain questions, including, among other things, whether testimony by Westinghouse before the DOL ALJ in this case asserted that registration with the NRRPT or a passing grade on an NRRPT registration examination was required before gaining employment with Westinghouse as a radiation technician.

By letter dated October 8, 1996, the Petitioner responded to my August 16th letter. By letter dated November 8, 1996, Westinghouse submitted its response to my August 16, 1996 letter.

II. DISCUSSION

Westinghouse is a contractor that provides services at various nuclear power plants that hold licenses from the NRC. Hydro Nuclear Services, Inc. (Hydro), was incorporated on January 23, 1985, as Westinghouse's nuclear decontamination service business, in part, providing workers to perform decontamination services at nuclear power plants. Hydro was a contractor for the Indiana & Michigan Power Company, which holds Facility Operating License Nos. DPR-58 and DPR-74, issued by the NRC pursuant to 10 C.F.R. Part 50 on March 30, 1976, and December 23, 1977, respectively. The licenses authorize the Licensee to operate the D.C. Cook Nuclear Power Plants in accordance with the conditions specified therein.

On December 9, 1988, the Petitioner filed a complaint with the DOL asserting that Hydro had violated section 210 of the Energy Reorganization Act (now section 211) when it failed to hire him as a decontamination technician to work at the D.C. Cook plants during an outage in the fall of 1988.

On March 30, 1994, the Secretary of Labor issued a Final Decision and Order in this case, 89-ERA-22, finding that Hydro had discriminated against the Petitioner.¹ Hydro petitioned the Court of Appeals for the Third Circuit for review of the Secretary's Final Decision and Order; however, on August 24, 1994, pursuant to a motion by the Secretary of Labor and Westinghouse, the court remanded the case to DOL for consideration of damages.

On December 14, 1994, a hearing on damages was held before a DOL ALJ. One of the issues raised at the hearing by the Petitioner was that he was entitled to damages for lost promotional opportunities as a result of his wrongful discharge. Specifically, he argued that he would have been promoted from decontamination technician to a position as health physics technician had he not been wrongfully discharged. With regard to this issue, on December 12, 1994, a deposition concerning the DOL complaint was taken. During the deposition, Mr. William Burns, Westinghouse Manager of Steam Generator Field Services and Field Readiness Operations, stated, in response to a question concerning the requirements for qualification to work as a health physics technician: "In the industry, the certain amount of hours would be given credit for, but there are also requirements of certain amount of education plus a national testing program to qualify as a radiation protection technician." (Tr. 17-18.) In addition, during the hearing, Mr. Burns, in response to questions concerning education or testing requirements to become a health physics technician, stated:

¹ On June 28, 1995, the NRC issued a Notice of Violation in Enforcement Action No. 95-080, to Westinghouse, categorizing the discrimination against Mr. Doyle as a Severity Level III violation.

Well, to be more or less board certified and receive certificates [of] education or testing, the National Registry of Radiation Protection Technicians semi-annually conduct [sic] testing seminars at the American Nuclear Society summer and winter meetings. The Health Physics Society also conducts certain amounts of school and testing to become a health physics technician, or a certified technician.

(Tr. 165.)

On March 28, 1995, Hydro filed "Post-Hearing Memorandum of Law Relating to the Assessment of Damages" in connection with the above matter. In this filing, Hydro stated, in part: "Doyle understood that to become a health physics technician, he had to log a certain number of hours of experience, pass a national test, and obtain the required educational background." (*Id.* at 25-26.)

On April 7, 1995, Hydro filed "Respondent's Proposed Findings of Fact and Conclusions of Law" concerning the above matter. In this filing, Hydro stated, in part: "Moreover, at no time during this job with Alabama Power did Doyle take or pass the national qualifying test needed for promotion to a board-certified health physics technician." (*Id.* at 2.)

On November 7, 1995, the ALJ issued his Recommended Decision and Order on Damages (Decision on Damages). In his Decision on Damages, apparently relying on the above, the ALJ stated, in part:

To establish lost promotions, Complainant must show: 1) that Complainant had the particular skills or other job-related qualifications required by Respondent to be promoted to health physics technician; 2) that the health physics technician position was in a line of progression upward from the decontamination technician position, that is, the decontamination technician would normally be promoted to health physics technician after some interval of acceptable performance; and 3) that the prerequisite service as a decontamination technician is not itself justified by business necessity aside from the skills or other qualifications to perform the health physics technician job. [Citation omitted.]

The Complainant has not fulfilled the first part of the analysis since he did not acquire the hours or the necessary passing grade on the health physics technicians exam.

(Decision on Damages at 17.)

Therefore, the ALJ denied the Petitioner's claim that he would have attained a position as health physics technician had he not been wrongfully discharged, and determined that the Petitioner was not entitled to damages for lost promotions.

Subsequently, the Petitioner appealed the ALJ's Decision on Damages and also attempted to supplement the record. In his appeal and motions to supplement the record, he argued that he was entitled to lost promotional benefits. As part of his Second Motion to Supplement the Record, the Petitioner submitted a filing by the NRRPT in a Petition for Rulemaking proposing an amendment to 10 C.F.R. Part 35, docketed by the NRC on November 24, 1995, purportedly to prove that the position of health physics technician did not require the passing of a national certification test.

On April 17, 1996, Hydro submitted a "Memorandum of Law in Opposition to Complainant Shannon T. Doyle's Second Motion to Supplement the Record." In this filing, Hydro stated, among other things: "At the damages hearing and at various depositions, . . . Mr. Burns clearly testified that in order to become a health physics technician, one was required to . . . (3) pass a national qualifying test." (*Id.* at 10.)

In his petition, the Petitioner indicates that these statements, which imply that passing a national qualifying test was required in order to obtain or hold the position of health physics technician at Westinghouse,² constitute the false statements provided by Westinghouse's counsel. In his October 8, 1996 response to my August 16, 1996 letter, the Petitioner further asserts that the ALJ was misled by Mr. Burns' testimony concerning schooling and testing requirements, which resulted in the ALJ's determination that natural progression would not have enabled him to attain the position of health physics technician.

Notwithstanding Hydro's position in its "Memorandum of Law in Opposition to Complainant Shannon T. Doyle's Second Motion to Supplement the Record," in its November 8, 1996 response to my August 16, 1996 letter, Westinghouse stated, in part:

No Westinghouse witness testified that NRRPT registration or passing an NRRPT registration exam was a prerequisite to gaining employment with Westinghouse as an HP [Health Physics] technician. In fact, the testimony is so general that it says nothing at all about specific Westinghouse or Hydro Nuclear hiring requirements or, for that matter, the specific requirements of any other employer.

In addition, Westinghouse asserted that its witness, Mr. Burns, provided the testimony concerning the NRRPT or similar requirements or certification. However, Westinghouse further asserted:

[T]aken in context, this testimony indicates only that HP technicians can and sometimes do obtain this type of board certification and that national organizations, such as the NRRPT, provide testing for individuals to obtain such certification. The inference can not be drawn from this testimony that such certification was an absolute prerequisite to employment as a HP technician at Westinghouse or elsewhere

(*Id.* at 4.)

² The NRC generally does not have specific requirements for qualification and training of health physics technicians.

III. ANALYSIS

It appears that Westinghouse, in its November 8, 1996 response to the NRC, characterized the evidence presented to DOL differently from that actually provided to the DOL in Westinghouse's submittals, as described above.

The NRC and DOL have complementary responsibilities in the area of employee protection.³ After considering the petition and the documents submitted by both the Petitioner and Westinghouse, I have determined that the petition raises matters that fall within the jurisdiction and authority of the DOL, rather than the NRC. For this reason, I have concluded that this matter should be referred to the DOL Administrative Review Board for its consideration.

IV. CONCLUSION

For the reasons set forth above, the petition is denied. In accordance with 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. 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

James Lieberman, Director
Office of Enforcement

Dated at Rockville, Maryland,
this 20th day of March 1997.

³ As noted in Section II, the NRC has taken enforcement action for the underlying violation of the applicable Commission discrimination regulation.

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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APPENDIX A — Witnesses (unpublished)

APPENDIX B — List of Exhibits (unpublished)

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.

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

⁴In 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).

⁵The 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).

⁶The 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) 50-160/87-01⁹ (GANE Exh. 21).

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 violations¹⁰ 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 examples).

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

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

¹³The 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.

¹⁴Although 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 programmatic 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 management¹⁶ 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

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

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. *Millsbaugh 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

²²The 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).

²³This 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. 2511-12 (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.

²⁶The 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 (GANExh. 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 (GANExh. 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 (GANExh. 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, slipshod 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

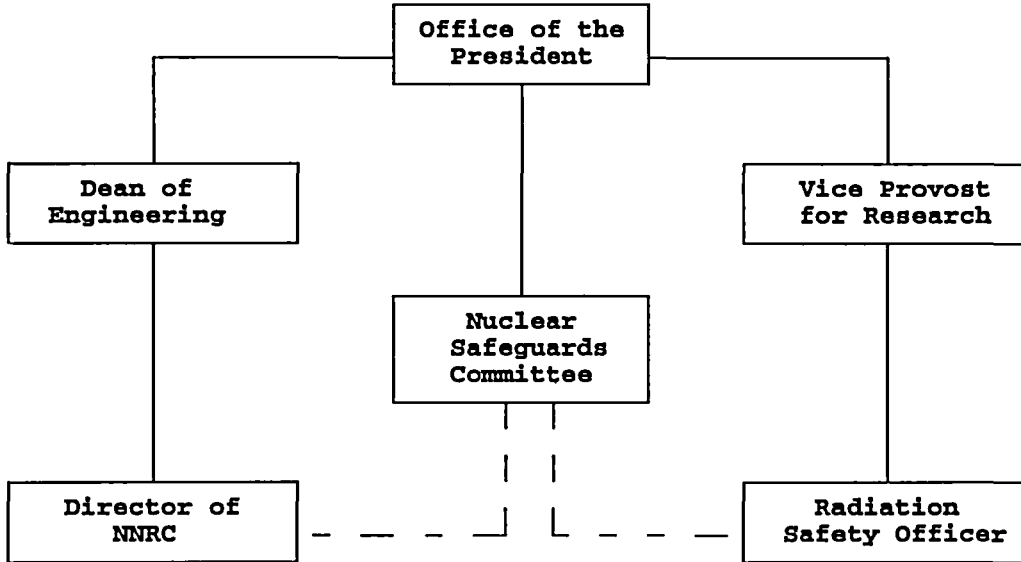


Figure 1. Derived from Tsoulfanidis, ff. Tr. 1939, at Exh. GT-2.

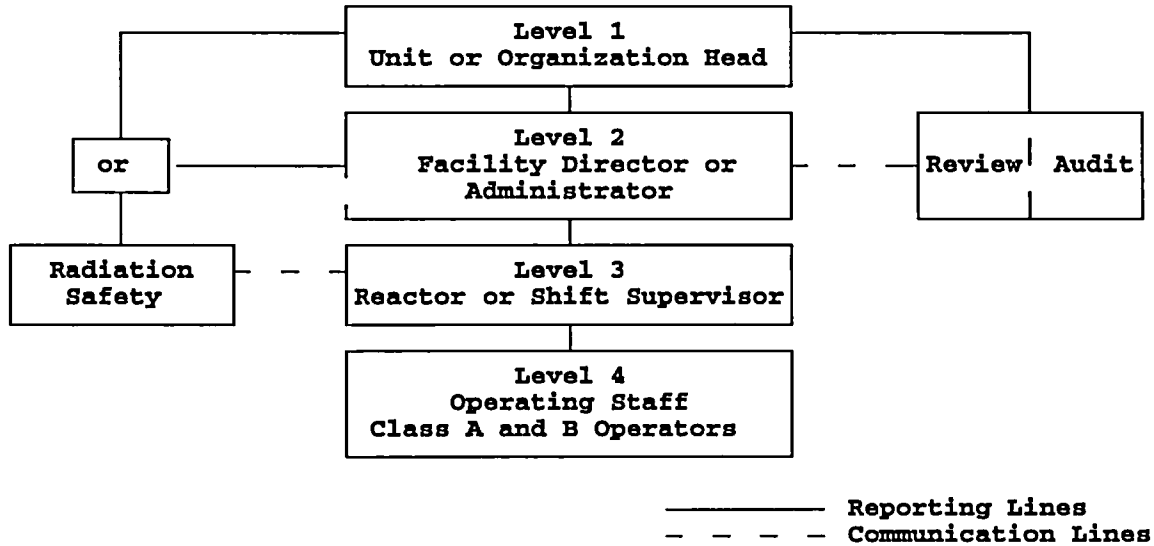


Figure 2. ANSI/ANS-15.1-1990, derived from Attachment 2 to Testimony of Staff Panel C, ff. Tr. 3171.

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,

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.

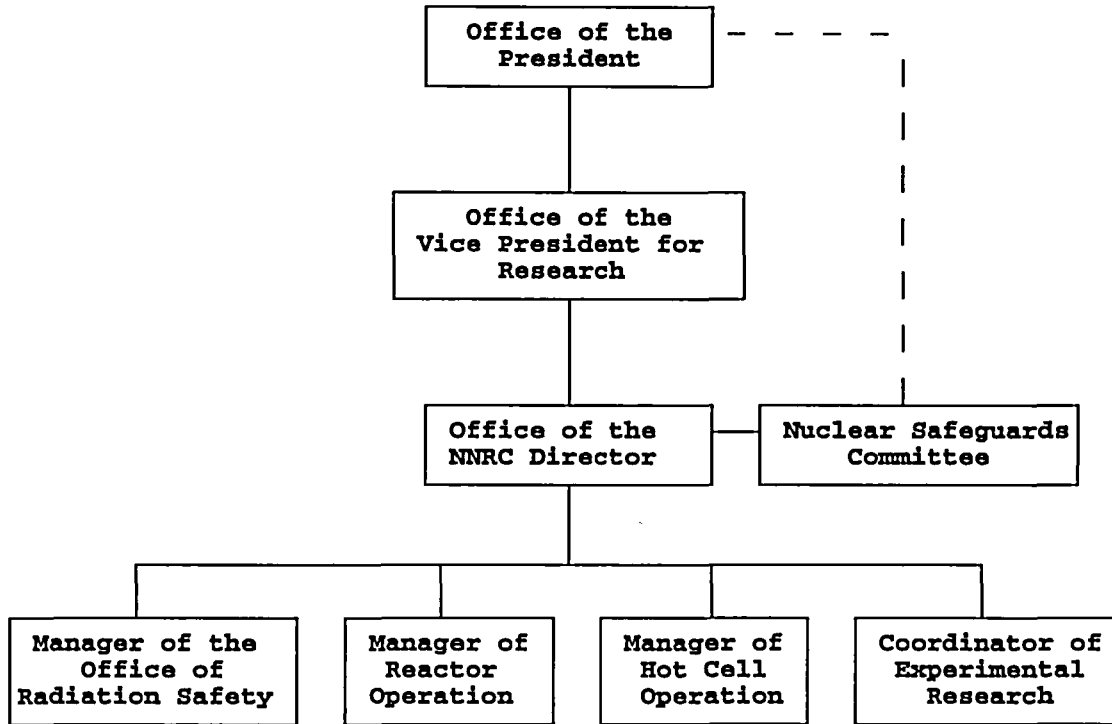


Figure 3. Derived from GT Exh. 6, NNRC Reorganization Chart.

(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:

³⁷The 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 MORS 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],

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 public 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

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

¹Specifically, 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

²Letter 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

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 OCNCS 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 OCNCS 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 OCNCS, 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 26, 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

**WISCONSIN ELECTRIC POWER
COMPANY**
(Point Beach Nuclear Plant,
Units 1 and 2)

Docket Nos. 50-266
50-301
72-5

CONSUMERS POWER COMPANY
(Pallsades 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(*f*), provides as follows:

(*f*) **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(*f*) 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.

¹The 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(d),

²The 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.

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

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 baghouse¹ 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 limits⁴ by disposing of all stored material in a

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-

⁵To 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

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 Millstone¹ 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

²The 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.

³The 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

⁴Section 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 procedure 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.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 55-20726-SP

RALPH L. TETRICK
(Denial of Application for Reactor
Operator License)

May 20, 1997

The Commission remands to the Presiding Officer the issue whether Mr. Tetrick correctly answered Question 63 of his written Senior Operator examination, and directs the Presiding Officer to reconsider expeditiously his prior negative ruling in light of new information submitted to the Commission. The Commission also grants a temporary stay of both the Presiding Officer's Initial Decision and his order denying reconsideration of the Initial Decision (LBP-97-2, 45 NRC 51 (1997), and LBP-97-6, 45 NRC 130 (1997)).

MEMORANDUM AND ORDER

On February 28, 1997, the Presiding Officer issued an Initial Decision in this proceeding, concluding that Ralph L. Tetrick, who is currently a reactor operator at the Turkey Point Nuclear Generating Plant (Units 3 and 4), had answered correctly seventy-eight out of ninety-eight valid questions on his Senior Reactor Operator (SRO) written examination. This ruling resulted in Mr. Tetrick's score being changed to 79.59%. The Presiding Officer then rounded Mr. Tetrick's revised score of 79.59 to the nearest integer, 80, thereby giving him a passing grade on the written examination. LBP-97-2, 45 NRC 51 (1997).

The NRC Staff filed a Motion for Reconsideration challenging the Presiding Officer's decision to "round up" the score. The Presiding Officer denied the NRC Staff's motion. LBP-97-6, 45 NRC 130 (1997). The Staff then filed with the Commission both a request for stay and a petition for review of LBP-97-2 and LBP-97-6, again challenging the Presiding Officer's decision to "round up" Mr. Tetrick's test score. In response, Mr. Tetrick asserted that, if the Commission reviews the Presiding Officer's decisions on the "rounding" issue, it should also examine whether the Presiding Officer was correct in ruling that Mr. Tetrick had answered Question 63 of the SRO examination incorrectly.¹

In a recent letter submitted by the NRC Staff to the Commission, dated May 1, 1997, the utility's Vice-President at Turkey Point has stated that he believes Mr. Tetrick's answer to Question 63 is a correct one. The Staff maintains otherwise. The matter appears to turn ultimately on the interpretation of language in a number of technical documents, some of which may not be in the record. This issue is, at bottom, a technical one on which we are unwilling to reverse or affirm the Presiding Officer without further factual and technical inquiry.

We therefore remand in its entirety the issue of Question 63 to the Presiding Officer and direct him to reconsider expeditiously his prior ruling in light of the utility's May 1st letter. "In Commission practice the [Presiding Officer], rather than the Commission itself, traditionally develops the factual record in the first instance." *Georgia Institute of Technology* (Georgia Tech Research Reactor, Atlanta, Georgia), CLI-95-10, 42 NRC 1, 2 (1995). *Accord Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 255 (1996).

We will defer a ruling on the "rounding up" issue, which remains pending before us, until after disposition of the remand. In light of our remand and the still-pending "rounding up" issue, we grant a temporary stay of LBP-97-2 and LBP-97-6. The Staff may withhold issuance of the Senior Reactor Operator license to Mr. Tetrick pending further order of the Commission.

¹That question reads as follows:

Plant conditions:

- Preparations are being made for refueling operations.
- The refueling cavity is filled with the transfer tube gate valve open.
- Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5, CNTMT SUMP HI LEVEL are in alarm.

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

- a. Verify alarms by checking containment sump level recorder and spent fuel level indication.
- b. Sound the containment evacuation alarm.
- c. Initiate containment ventilation isolation.
- d. Initiate control room ventilation isolation.

The only issue before us on appeal regarding Question 63 is whether Mr. Tetrick's answer of "a" is also correct. (Everyone agrees that answer "b" is correct.)

IT IS SO ORDERED..

For the Commission²

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 20th day of May 1997.

² Commissioner Diaz was not available for the affirmation of this Order. Had he been present, he would have approved the Order.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

**REGENTS OF THE UNIVERSITY
OF CALIFORNIA
(Indemnity Claim)**

May 29, 1997

The Commission denies the Regents' claim for the NRC's payment of attorney's fees and expenses incurred in the Regents' defense of two private tort suits against it (subsequently settled) for alleged harm caused by radioactive releases from the NRC-licensed Argonaut nuclear test reactor at the University of California at Los Angeles (UCLA).

The Commission finds that section 170 of the Atomic Energy Act (known as the Price-Anderson Act) bars the NRC's payment of licensee legal expenses incurred in connection with settlements. Furthermore, the Commission finds that even if it were permitted to pay such expenses under the Act, it would not approve the claim because by statute and under the Indemnity Agreement the Regents should have timely notified the NRC at the point where governmental indemnity arose and should have sought NRC approval of the settlement of the tort cases.

**ATOMIC ENERGY ACT (AEA): INTERPRETATION OF
SECTION 170h (PRICE-ANDERSON ACT)**

The Price-Anderson Act is best understood as barring Commission payment of licensee legal expenses incurred in connection with settlements. 42 U.S.C. § 2210(h).

NRC: CONSIDERATION OF INDEMNITY CLAIMS

The Commission cannot authorize expenditures of government money without express statutory authority or in the face of a statutory prohibition against such payments. 31 U.S.C. §§ 1341, 1350.

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

Section 170h of the AEA appeared in the original 1957 Price-Anderson Act. It provides the authority for the Commission, when it anticipates making indemnity payments for public liability claims, to collaborate with an indemnified person, approve payments of claims, take charge of such action, and settle or defend any such action.

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The 1975 Hathaway Amendment altered section 170h of the AEA by providing that a Commission-approved settlement "shall not include expenses in connection with the claim incurred by the person indemnified."

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The 1988 Price-Anderson Act amendments loosened restrictions on government payment of legal costs and modified several of the Hathaway Amendment provisions, but did not alter section 170h in any respect; therefore, the bar against indemnifying a licensee's expenses in settlements remains in place.

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The Commission believes that a lawsuit that is dismissed voluntarily after a negotiated arrangement in which a licensee, among other things, forfeits any right to seek costs from plaintiff qualifies as a "settlement" and not a "dismissal."

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The fact that a specific provision of the Price-Anderson Act other than section 170h was modified by the 1988 Amendments to contemplate government

payment of licensee legal costs in some situations does not mean that Congress repealed section 170h by implication.

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The Price-Anderson Act contemplates that at the point where governmental indemnity arises in a public liability claim, the licensee will offer the government the opportunity to take over defense of the claims and manage the lawsuit. 42 U.S.C. § 2210(h).

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

By statute, a licensee is required both to notify the NRC that it has reached the point where government indemnification payments will be required under a public liability claim and to seek NRC's approval of the settlement of such a claim.

ATOMIC ENERGY ACT (AEA): INTERPRETATION OF SECTION 170h (PRICE-ANDERSON ACT)

The Price-Anderson Act provides for indemnification of expenses incurred defending claims against licensees, not reimbursement for expenses incurred in presenting claims to the government.

DECISION

I. INTRODUCTION

In a series of letters beginning on January 17, 1996, the Regents of the University of California have demanded that the Commission pay \$91,375.22 in indemnification for attorneys' fees and expenses incurred in defending two private tort suits against the Regents.¹ The Regents seek indemnification under

¹The Regents' initial letter, dated January 17, 1996, demanded NRC payment of \$76,102.26. More recently, in a letter dated January 31, 1997, the Regents amended their claim to include an additional \$15,272.96 in legal costs, an amount that apparently reflects attorneys' fees and costs the Regents have incurred in pursuing their indemnity claim with the NRC. The Regents' submissions do not make clear who bears the risk of loss in the event that the NRC rejects the indemnity claim. That presumably is a matter of contract among the Regents, their private insurer, and the law firm that has handled this matter.

section 170 of the Atomic Energy Act, 42 U.S.C. § 2210 (known as the Price-Anderson Act), and under their indemnity agreement with the Commission executed pursuant to that Act.

The two underlying tort suits, known as the *Miller* and *Redisch* cases, sought damages for harm to plaintiffs' persons allegedly caused by releases of radioactivity during normal operations of the NRC-licensed Argonaut nuclear test reactor at the University of California at Los Angeles (UCLA) between 1979 and 1984. By late October 1996, the Regents had settled both cases, which therefore were never tried or decided on the merits. The settlements resulted in the payment of no damages to plaintiffs. Under their terms, plaintiffs voluntarily dismissed their lawsuits, and the Regents relinquished all rights to seek legal costs from plaintiffs.

Under the Price-Anderson Act and under the Commission's indemnity agreement with the Regents, the Commission agreed to indemnify the Regents for "public liability" exceeding \$250,000 when such liability arises from a "nuclear incident." See section 170k, 42 U.S.C. § 2210(k). The Regents' January 17, 1996 claim for indemnity asserted that expenses incurred in defending the *Miller* and *Redisch* cases exceeded the \$250,000 threshold by roughly \$76,000. The Regents' private insurer apparently paid the first \$250,000 in legal costs.

In a letter dated August 6, 1996, the Commission's Office of the General Counsel advised lawyers for the Regents that it was disinclined to recommend payment of the indemnity claim. More than 6 months later, on January 31, 1997, the Regents replied and asked that their claim be presented directly to the Commission.

After reviewing the factual background of the Regents' indemnity claim, the relevant provisions of the Price-Anderson Act, and the Regents' letters and submissions to the NRC detailing their claim, we have decided to deny it — for two independent reasons. First, the Price-Anderson Act is best understood as barring Commission payment of licensee legal expenses incurred in connection with settlements. See section 170h, 42 U.S.C. § 2210(h). Second, even if we were able to construe the Act to permit Commission payment of such expenses as a general matter, we would not approve an indemnity payment in this case because the Regents failed to give the Commission reasonable notice of the extent of their expenses in time for the Commission to take protective measures. See *id.* Some of the expenses also appear unreasonably excessive or insufficiently related to defense of the underlying tort suits.

We detail the reasons for our decision below. We issue our decision as a formal opinion because the Regents specifically requested Commission consideration of their indemnity claim, and because our views may shed some light on seldom invoked provisions of the Price-Anderson Act.

II. DISCUSSION

The Commission plainly cannot authorize expenditures of government money without express statutory authority or in the face of a statutory prohibition against such payments. Both the Constitution (the Appropriations Clause, art. 1, § 9, cl. 7) and federal statute (31 U.S.C. §§ 1341, 1350) impose this restriction on Commission expenditures. *See Office of Personnel Management v. Richmond*, 496 U.S. 414, 424-30 (1990). Under the related “sovereign immunity” doctrine (*id.* at 432), a claimant may not pursue monetary relief against the government absent authority “unequivocally expressed in statutory text.” *Lane v. Pena*, 116 S. Ct. 2092, 2096 (1996).

This background law requires the Commission to scrutinize the Regents’ claim against the public treasury in this case with great care. We cannot discern the clear authority necessary to pay the claim. Nor would we find the claim otherwise payable even if we were able to answer the authority question differently.

I. Authority to Pay

Contrary to the Regents’ view, we believe that section 170h of the Atomic Energy Act provides the governing law. That section appeared in the original 1957 Price-Anderson Act and to this day provides the authority for the Commission to collaborate with an indemnified person, approve payments of claims, appear through the Attorney General on behalf of the person indemnified, take charge of such action, and settle or defend any such action. Section 170h further provided, in its original form, that a settlement “may include reasonable expenses in connection with the claim incurred by the person indemnified.”²

Section 170h has had only one substantive alteration. That came in 1975 as part of a series of changes presented as an amendment by Senator Hathaway. Senator Hathaway’s aim was (at least in part) to ensure that government indemnity money ended up in the hands of victims of nuclear incidents, and was not diverted to attorney’s fees and other costs. *See generally* Damage Claims Under the Atomic Energy Act, 1 U.S. Op. OLC 157 (1977).

The Hathaway Amendment altered a number of the Act’s provisions, including section 170h, which as revised provided that a Commission-approved settlement “shall *not* include expenses in connection with the claim incurred by the person indemnified” (emphasis added). “Therefore,” concluded the Comptroller General in a 1980 opinion, “the Act must be interpreted as follows: the gov-

²See H.R. Rep. No. 296, 85th Cong., 1st Sess. 23 (1957) (noting that the expenses “could include reasonable attorney’s fees incurred by the person indemnified in examining any claims”).

ernment will not indemnify a person for his legal expenses.” See “Interpretation of Price-Anderson Act,” File B-197742, 1980 WL 16980, at *4 (C.G.).

In 1988 amendments to the Price-Anderson Act, after revisiting the legal costs issue in cognizant committees, Congress loosened the across-the-board restrictions on government payment of legal costs and modified several of the Hathaway Amendment provisions, *but did not alter section 170h in any respect*. This leaves in place the section 170h bar against indemnifying a licensee’s expenses in settlements and prevents the Commission from paying the legal expenses incurred by the Regents in settling the *Miller* and *Redisch* cases. Congress may have assumed that licensees’ own insurance would be adequate to cover legal costs in such cases. See *Damage Claims Under the Atomic Energy Act*, 1 U.S. Op. OLC at 158 & n.3 (discussing legislative history of Hathaway Amendment).

The Regents argue that section 170h does not apply here because the *Miller* and *Redisch* lawsuits in actuality were dismissed, not settled. We find this argument wholly unpersuasive. The documents the Regents themselves have provided us show plainly that the two cases were dismissed voluntarily and only after the parties reached a negotiated arrangement in which the Regents, among other things, forfeited any right to seek costs from plaintiffs. By any standard, this qualifies as a “settlement.”

The Regents’ only other argument is that the section 170h bar must give way because it is less “specific” than another provision, section 170k, which applies to educational institutions and appears to contemplate government payment of licensee legal costs in some situations.³ As noted above, the “legal costs” language currently found in section 170k (and in other Price-Anderson Act provisions) dates from the 1988 Amendments that modified some aspects of the 1975 Hathaway Amendment but made no changes in section 170h. Standard principles of statutory construction prevent us from assuming that Congress repealed section 170h by implication. *Watt v. Alaska*, 451 U.S. 259, 266-67 (1981). On the contrary, we are obliged to give effect to all statutory provisions.

³ Section 170k’s applicability here is far from crystal clear by its own terms. That provision establishes that the Commission shall indemnify educational licensees “from public liability in excess of \$250,000 for nuclear incidents,” and says that the “aggregate indemnity” in connection with each nuclear incident may not exceed \$500,000,000, “including such legal costs as are approved by the Commission.” But in this case the aggregate indemnity limit was never approached. And no public liability payment was made, much less one in excess of \$250,000. By definition, “public liability” does not include legal costs; by contrast, licensees’ own “financial protection” is defined as including damages and legal costs. See sections 11k, 11w, 42 U.S.C. §§ 2014(k), (w). For educational institutions the financial protection requirement was waived and instead the requirement for exceeding \$250,000 in public liability was established as the trigger for governmental indemnity. See section 170k, 42 U.S.C. § 2210(k).

Id. See Bennett v. Spear, 117 S. Ct. 1154, 1166 (1997).⁴ We cannot, therefore, accept the Regents' invitation simply to ignore the section 170h prohibition.

We see no basis, in sum, for disregarding section 170h's apparent prohibition against paying licensee legal expenses incurred in settling cases. The Regents themselves have offered us none. We therefore decline to approve their indemnity claim.

2. *Prior Notice and Reasonableness of Indemnity Claim*

Even if section 170h did not bar Commission reimbursement of licensee legal costs in settled cases, as we think it does, we would not approve payment of the Regents' indemnity claim in this case. The Price-Anderson Act, and the NRC's indemnity agreement with the Regents, indisputably contemplate Commission "approval" of claims for legal costs. Such a right of approval implies Commission review for reasonableness. Here, we cannot find the Regents' claim reasonable.

a. As a matter of procedure, the Price-Anderson Act contemplates that at the point where governmental indemnity arises, here at the \$250,000 threshold, the licensee will offer the government the opportunity to take over defense of the claims and manage the lawsuit. *See* section 170h, 42 U.S.C. § 2210(h). One purpose of this provision, presumably, is to allow the government to take over representation or active management of the case with a view toward minimizing public expenses.

Here, a series of letters from counsel for the Regents did alert the NRC Staff to the existence of the *Miller* and *Redisch* cases, and to the possibility of exceeding the \$250,000 limit. But the Regents' letters also indicated that plaintiffs' merits claims were insubstantial and that the case would be "tendered" to the NRC if expenses reached the \$250,000 limit. *See, e.g.*, Letter dated August 10, 1995. No "tender" ever occurred until the two cases ended, after the Regents had exceeded the \$250,000 limit by nearly \$80,000. The lack of timely tender prejudiced the NRC.

Eight days before the parties agreed on the settlement in *Redisch*, with the *Miller* suit having already been dismissed, the Regents' insurer sent the NRC a letter reporting \$28,534.08 in remaining "available financial protection" from the private insurer and indicating that tender to NRC was expected "in the very near future since [the *Redisch* case] is still unresolved." *See* Letter from Bohner, dated October 18, 1995. But it now appears that in actuality the Regents' law

⁴ Our reading of section 170h does not nullify the "legal costs" authorization found in section 170k or in other provisions of the Price-Anderson Act. Those provisions remain applicable in the absence of a settlement. Moreover, even in connection with a settlement, the Commission could approve payment of *plaintiffs'* legal costs. *See* section 11jj, 42 U.S.C. § 2014(jj). Section 170h simply prevents Commission payment of licensees' legal costs in settling a case.

firm at that time already had incurred additional billable hours amounting to more than \$30,000 and already had paid out additional expenses in excess of \$20,000 (many apparently incurred much earlier). In other words, the Regents already had entirely consumed and substantially exceeded the \$28,534 that supposedly remained as “available financial protection.”

Thus, if the Regents were correct that their legal expenses were payable by the NRC after \$250,000 (*but see* note 3, *supra*), they had reached an appropriate tender time and passed it before they negotiated the *Redisch* settlement. By statute, they not only ought to have notified the NRC but they also should have sought NRC approval of the settlement. *See* section 170h, 42 U.S.C. § 2210(h). As part of the settlement, however, and without NRC approval, they relinquished any right to claim legal costs against plaintiffs or monetary sanctions under Rule 11 of the Federal Rules of Civil Procedure. Had the NRC been given presettlement notice that the \$250,000 limit had been reached, it might have insisted on some recompense from plaintiffs or their lawyers for the substantial expenses their insubstantial lawsuit had caused. The government almost surely would have limited any further expenditures by the private lawyers.

Even the Regents’ letter reporting termination of the case indicated that there still remained \$3,654.94 of the insurance money. That letter suggested only that “some expense in excess” of \$250,000 might be expected. *See* Letter dated December 6, 1995. By then, of course, there was no case for the government to take over and no opportunity to minimize government costs. In addition, when read in conjunction with the prior letter’s reference to \$28,000 in remaining financial protection, the close-out letter’s language raised no expectation of more than a *de minimis* exceeding of the \$250,000 limit. The NRC therefore was quite surprised a few weeks later, when counsel for the Regents demanded \$76,000 from the Commission. The substantial excess, one-third again over the insurance amount, apparently occurred in some measure because of late-arriving bills for earlier-performed services.

In these circumstances, the government was not given a timely opportunity to take over these cases and minimize public costs. The Regents have since suggested that the NRC Staff ought to have been aware that experts’ fees would be high and that pretrial preparation would be expensive; however, the people in the best position to make that assessment were the defendants’ counsel themselves. The Regents’ correspondence did not call attention to the apparently lengthy lag time between incurring obligations for expenses and notification of them as expenditures. And, as we stressed above, the Regents did not make its tender in time for the NRC to monitor and approve the ultimate settlement or otherwise to take action in an attempt to minimize the potential costs to the U.S. government.

In short, given the Regents' failure to timely tender the case to the NRC, we do not find it reasonable for the government to pick up the bill for the Regents' expenses.

b. In addition, some of the expenses incurred by the Regents in reaching and exceeding the \$250,000 limit appear questionable substantively. To begin with, we see no basis in the Price-Anderson Act to approve the Regents' claim for approximately \$15,000 in attorney's fees and costs incurred *after* termination of the underlying tort suits, apparently as part of the Regents' effort to persuade the NRC to make indemnity payments. See note 1, *supra*. The Act provides for indemnification of expenses incurred defending claims against licensees, not reimbursement for expenses incurred in presenting claims to the government.

The Regents' fee claim raises a number of additional questions. For example, the billing records' descriptions of law firm hours are often vague and insufficiently segregated as to tasks as well as being chronologically out of order — with significant expenses for billed hours appearing considerably later than previous invoices represented as being “for services rendered through” a specified date. Moreover, the billing records indicate that counsel incurred substantial expenses on matters not directly related to defense of the tort cases, such as correspondence with the insurer-client and organizing what were apparently disorganized UCLA files. Finally, the records show that high-priced law firm partners, rather than associates or paralegals, conducted such fairly mundane tasks as document and privilege reviews and also that they traveled extensively to meet with experts rather than conduct conferences by telephone, at significantly less expense.

The Regents might be able to provide adequate answers to some or all of our substantive questions. But we need not resolve these questions definitively in view of our decision on other grounds not to pay Price-Anderson Act indemnity in this case.

III. CONCLUSION

For the foregoing reasons, the Commission declines to approve the Regents' indemnity claim.

For the Commission

JOHN C. HOYLE
Secretary of the Commission

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Thomas S. Moore, Chairman
Richard F. Cole
Frederick J. Shon

In the Matter of

Docket No. 70-3070-ML
(ASLBP No. 91-641-02-ML)
(Special Nuclear Material License)

LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center)

May 1, 1997

In this Final Initial Decision in the combined construction permit–operating license proceeding for the Claiborne Enrichment Center, the Licensing Board (1) determines that a thorough NRC Staff investigation of the facility site selection process is essential to determine whether racial discrimination played a role in that process, thereby ensuring compliance with the nondiscrimination directive contained in Executive Order 12898; (2) resolves in favor of the Intervenor portions of the contention concerning the adequacy of the Staff’s treatment in the final environmental impact statement of the impacts of relocating the parish road connecting the African American communities of Forest Grove and Center Springs and the economic impacts of the facility on properties in those communities; and (3) denies the Applicant’s requested authorization for a license.

NEPA: ENVIRONMENTAL JUSTICE

On February 11, 1994, the President issued Executive Order 12898, 3 C.F.R. 859 (1995), titled “Federal Actions to Address Environmental Justice in Minority

Populations and Low-Income Populations,” and an accompanying Memorandum for the Heads of All Departments and Agencies, 30 Weekly Comp. Pres. Doc. 279 (Feb. 14, 1994). The President’s memorandum states that the Executive Order is designed “to focus Federal attention on the environmental and human health conditions in minority communities and low-income communities with the goal of achieving environmental justice” and “to promote nondiscrimination in Federal programs substantially affecting human health and the environment.”

NEPA: ENVIRONMENTAL JUSTICE

As an independent regulatory agency the NRC is not mandatorily subject to Executive Order 12898. Nevertheless, on March 31, 1994, the then Chairman of the Commission wrote the President stating that the NRC would carry out the measures in the Executive Order. By voluntarily agreeing to implement the President’s environmental justice directive, the Commission has made it fully applicable to the agency and, until that commitment is revoked, the President’s order, as a practical matter, applies to the NRC to the same extent as if it were an executive agency. The NRC is obligated, therefore, to carry out the Executive Order in good faith in implementing its programs, policies, and activities that substantially affect human health or the environment.

NEPA: ENVIRONMENTAL JUSTICE

Although Executive Order 12898 does not create any new rights that the Intervenor may seek to enforce before the agency or upon judicial review of the agency’s actions, the President’s directive is, in effect, a procedural directive to the head of each executive department and agency that, “to the greatest extent practicable and permitted by law,” it should seek to achieve environmental justice in carrying out its mission by using such tools as the National Environmental Policy Act.

NEPA: ENVIRONMENTAL JUSTICE

Pursuant to the President’s order, there are two aspects to environmental justice: first, each agency is required to identify and address disproportionately high and adverse health or environmental effects on minority and low-income populations in its programs, policies, and activities; and second, each agency must ensure that its programs, policies, and activities that substantially affect human health or the environment do not have the effect of subjecting persons and populations to discrimination because of their race, color, or national origin.

NEPA: ENVIRONMENTAL JUSTICE

It is clear that Executive Order 12898 directs all agencies in analyzing the environmental effects of a federal action in an EIS required by NEPA to include in the analysis, “to the greatest extent practicable,” the human health, economic, and social effects on minority and low-income communities.

NEPA: ENVIRONMENTAL JUSTICE

In using the term human health and environmental “effects” in Executive Order 12898 and the accompanying memorandum the President’s order tracks the regulations of the Council on Environmental Quality (“CEQ”) that define “effects” to include both direct and indirect effects and states that “[e]ffects includes ecological (such as the effects on natural resources and on the components, structures, and functioning of affected ecosystems), aesthetic, historic, cultural, economic, social, or health, whether direct, indirect, or cumulative.” 40 C.F.R. § 1508.8(b).

NEPA: ENVIRONMENTAL JUSTICE

Executive Order 12898 does impose duties on the NRC because the Commission has undertaken to carry out the President’s directive, but no party to an agency proceeding has a remedy with regard to the manner in which the agency carries out its commitment to the President to implement Executive Order 12898.

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FINAL INITIAL DECISION
(Addressing Contention J.9)

This Final Initial Decision addresses the remaining contention — environmental justice contention J.9 — filed by the Intervenor, Citizens Against Nuclear Trash (“CANT”), in this combined construction permit–operating license proceeding. The Applicant, Louisiana Energy Services, L.P. (“LES”), seeks a 30-year materials license to possess and use byproduct, source, and special nuclear material in order to enrich uranium using a gas centrifuge process at the Claiborne Enrichment Center (“CEC”). The Applicant plans to build the CEC on a 442-acre site in Claiborne Parish, Louisiana, that is immediately adjacent to and between the unincorporated African-American communities of Center Springs and Forest Grove, some 5 miles from the town of Homer, Louisiana.

There is no serious dispute between the parties regarding the essential facts concerning the site location and area demographics. Claiborne Parish is in northern Louisiana and lies along the southern border of Arkansas. The proposed CEC site is located in the approximate center of the parish some 50 miles northeast of Shreveport, Louisiana. The site, called the LeSage property, is a rough approximation of a square and the CEC will occupy the center 70 acres of the site. The LeSage property is currently bisected by Parish Road 39 (also known as Forest Grove Road) running north and south through the property.

Immediately to the north of the site, Parish Road 39 crosses State Road 9 that runs in a northeasterly direction from the town of Homer 5 miles away. The community of Center Springs, roughly centered on the Center Springs Church, lies along State Road 9 and Parish Road 39 and is located approximately 0.5 kilometer (about 0.33 mile) to the north of the LeSage property. The community of Forest Grove, again very roughly centered on the Forest Grove Church, lies approximately 3.2 kilometers (about 2 miles) south of the site along Parish Road 39 (and other intersecting unnamed local roads). The Forest Grove Community runs south along Parish Road 39 to where Parish Road 2 crosses State Road 2 that runs in an easterly direction from the town of Homer. The two community churches, which share a single minister, are approximately 1.1 miles apart, with the LeSage property lying between them.

The community of Forest Grove was founded by freed slaves at the close of the Civil War and has a population of about 150. Center Springs was founded around the turn of the century and has a population of about 100. The populations of Forest Grove and Center Springs are about 97% African American. Many of the residents are descendants of the original settlers and a large portion of the landholdings remain with the same families that founded the communities. Aside from Parish Road 39 and State Road 9, the roads in Center Springs or Forest Grove are either unpaved or poorly maintained. There are no stores, schools, medical clinics, or businesses in Center Springs or Forest Grove. The Intervenor's evidence was undisputed that from kindergarten through high school the children of Center Springs and Forest Grove attend schools that are largely racially segregated. Many of the residents of the communities are not connected to the public water supply. Some of these residents rely on groundwater wells while others must actually carry their water because they have no potable water supply.

Although none of the parties put in any specific statistical evidence on the income and educational level of the residents of Forest Grove and Center Springs, the 1990 United States Bureau of the Census statistics in the record show they are part of a population that is among the poorest and most disadvantaged in the United States. Claiborne Parish is one of the poorest regions of the United States with a total population in 1990 of 17,405 and a racial makeup of 53.43% white and 46.09% African American. Over 30% of the parish population live below the poverty level with over 58% of the black population and 11% of the white population living below the poverty line. Per capita income of the black population of Claiborne Parish is only 36% of that of the white population, compared to a national average of 55%. Over 69% of the black population of Claiborne Parish earn less than \$15,000 annually, 50% earn less than \$10,000, and 30% earn less than \$5,000. In contrast, among whites in the parish, 33% earn less than \$15,000 annually, 21.5% earn less than \$10,000, and 6.5% earn less than \$5,000. In Claiborne Parish, over 31% of blacks live in households in which there are no motor vehicles and over 10% live in households that lack complete plumbing. Over 50% of the African-American households in the parish have only one parent, 58% of the black population have less than a high school education, including almost 33% of the parish black population over 24 years old that has not attained a ninth grade education.

The Intervenor's environmental justice contention is grounded in the requirements of the National Environmental Policy Act of 1969, 42 U.S.C. § 4321 *et seq.* ("NEPA"). As originally filed, the contention essentially asserts that the negative economic and sociological impacts of closing Parish Road 39 connecting the minority communities to make way for the plant and placing the facility in the midst of a rural black community of over 150 families have not been appropriately considered in the Applicant's Environmental Report ("ER").

Further, the contention claims that the siting of the CEC follows a national pattern of siting hazardous facilities in minority communities and that no steps to avoid or mitigate the disparate impact of the CEC on this minority community have been taken.

With this Final Initial Decision addressing contention J.9, all of the issues in the licensing proceeding will have been addressed. The history of this proceeding may be found in three previous decisions. See LBP-96-7, 43 NRC 142 (1996); LBP-96-25, 44 NRC 331 (1996); LBP-97-3, 45 NRC 99 (1997). Suffice it to say that the three earlier Partial Initial Decisions decided all of the Intervenor's other health, safety, safeguards, environmental, financial qualification, and decommissioning funding contentions in the proceeding. Like a number of the other contentions in this proceeding, the Intervenor's environmental justice contention J.9 presents questions of first impression in NRC licensing proceedings.

I. ENVIRONMENTAL JUSTICE CONTENTION

A. Contention J.9

In its entirety, the Intervenor's contention J.9 asserts that the Applicant's Environmental Report does not adequately describe or weigh the various environmental, social, and economic impacts and costs of operating the CEC. In support of the contention, it then states:

BASIS: NEPA requires the NRC to fully assess the impacts of the proposed licensing action, and to weigh its costs and benefits. LES' Environmental Report contains a brief "benefit-cost analysis" that is improperly slanted in favor of the benefits of the project, and contains little discussion of the potentially significant impacts and their environmental and social costs. The discussion is inadequate with respect to the following issues:

. . . .

9. The proposed plant will also have negative economic and sociological impacts on the minority communities of Forest Grove and Ce[nter] Springs. Forest Grove Road, which joins the two communities, must be closed in order to make way for the proposed plant, which would lie between them. If the road is closed off, it will cause hardships to families who use the road, residents who car-pool to work, school transportation, sports-related activities that involve children living in both communities, and church services that are divided between the two communities.

Moreover, the ER does not reflect consideration of the fact that the plant is to be placed "in the dead center o[f] a rural black community consisting of over 150 families." The proposed siting of the CEC in a minority community follows a pattern noted in a 1987 study by the United Church of Christ, "Toxic Wastes and Race In the United States, A National Report on the Racial and Socio-Economic Characteristics of Communities With Hazardous Waste Sites." The study found that "[r]ace proved to be the most significant among variables tested in association with the location of commercial hazardous waste facilities. This represented

a consistent national pattern." It also found that "In communities with one commercial hazardous waste facility, the average minority percentage of the population was twice the average minority percentage of the population in communities without such facilities (24 percent vs. 12 percent)." The ER does not demonstrate any attempts to avoid or mitigate the disparate impact of the proposed plant on this minority community. [Citations and footnotes omitted.]

In opposing the admission of the contention before the Licensing Board, the Applicant argued that CANT's "allegations are premised on speculation" and that the Intervenor had provided "no support for the proposition that closing off Forest Grove Road and building the plant will have negative impacts on the two communities." LBP-91-41, 34 NRC 332, 353 (1991). The NRC Staff did not oppose the admission of the contention. The Licensing Board, as then constituted, admitted contention J.9 ruling that "CANT has identified an issue with sufficient basis and specificity to meet the requirements of [10 C.F.R. § 2.714(b)(2)]." *Id.* As in the case of several of the Intervenor's other contentions that were heard in this proceeding, CANT contention J.9, which was required by the Commission's Rules of Practice to be filed before the issuance of the environmental impact statement ("EIS"), is phrased only in terms of a challenge to the Applicant's ER. *See* LBP-96-25, 44 NRC at 337-38. Nevertheless, the Intervenor's contention necessarily encompasses the Staff's later-filed final environmental impact statement and all parties in their evidentiary presentations on contention J.9 included evidence on all aspects of the issues. *See id.*; 10 C.F.R. § 2.714(b)(2)(iii).

Further, as indicated in the earlier decisions in this proceeding, the Commission's Rules of Practice, 10 C.F.R. § 2.732, provide that the Applicant has the burden of proof in the proceeding. Therefore, in order for the Applicant to prevail on each contested factual issue, the Applicant's position must be supported by a preponderance of the evidence. *See* LBP-96-7, 43 NRC at 144-45. As LBP-96-25 indicates, however, where environmental and NEPA issues are involved, care must be taken in applying the Commission's general burden of proof rule because the NRC, not the Applicant, has the burden of complying with NEPA. Accordingly, because the Commission's regulations require the Applicant to file an environmental report and prescribe its contents, the Applicant has the burden on contentions, or portions of contentions like J.9, asserting deficiencies in the ER. Similarly, because the Staff is ultimately responsible for preparing the EIS required by NEPA, the Staff generally has the burden on contentions, or portions of contentions like J.9 that are taken to assert deficiencies in the FEIS. Additionally, because the Staff relies extensively upon the Applicant's ER in preparing the EIS, when the Applicant becomes a proponent of a particular challenged position set forth in the EIS, the Applicant, as such a proponent, also has the burden on that matter. *See* LBP-96-25, 44 NRC at 338-39.

Finally, we reiterate the additional NEPA obligations the Commission placed upon the Licensing Board in the hearing notice. The Commission directed the Board to determine whether the Staff's environmental review conducted pursuant to 10 C.F.R. Part 51 was adequate and whether the agency had complied with the requirements of section 102(2)(A), (C), and (E) of NEPA. In addition, the Commission instructed the Board independently to consider the cost-benefit balance among the conflicting factors contained in the record of the proceeding. *See* 56 Fed. Reg. 23,310 (1991). As we noted previously in LBP-96-25, 44 NRC at 339, "[a]lthough obviously related, these obligations placed upon us by the Commission to ensure the agency's compliance with NEPA are independent of the parties' burdens with respect to the Intervenor's environmental contentions."

B. Executive Order 12898

Subsequent to the admission of the Intervenor's contention J.9 and the Staff's issuance of the draft EIS, on February 11, 1994, the President issued Executive Order 12898, 3 C.F.R. 859 (1995), and an accompanying Memorandum for the Heads of All Departments and Agencies, 30 Weekly Comp. Pres. Doc. 279 (Feb. 14, 1994). The President's order, titled "Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations," contains a number of provisions but two are most pertinent here. In subsection 1-101 under the heading "Agency Responsibilities," the President directs that

[t]o the greatest extent practicable and permitted by law . . . each Federal agency shall make achieving environmental justice part of its mission by identifying and addressing, as appropriate, disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority populations and low-income populations in the United States.

3 C.F.R. at 859. Further, in section 2.2, the President orders that

[e]ach Federal agency shall conduct its programs, policies, and activities that substantially affect human health or the environment, in a manner that ensures such programs, policies, and activities do not have the effect of excluding persons (including populations) from participation in, denying persons (including populations) the benefits of, or subjecting persons (including populations) to discrimination under, such programs, policies, and activities, because of their race, color, or national origin.

Id. at 861. The President's directive also contains a number of general provisions. In subsection 6-604, the President requests that independent agencies comply with the provisions of the order. *See id.* at 863. Finally, subsection 6-609 states that the order is intended to improve the internal management of the

executive branch and that it does not create any substantive or procedural rights in any person or create any right of judicial review. *See id.*

The President's memorandum accompanying the order states that the Executive Order is designed "to focus Federal attention on the environmental and human health conditions in minority communities and low-income communities with the goal of achieving environmental justice" and "to promote nondiscrimination in Federal programs substantially affecting human health and the environment." 30 Weekly Comp. Pres. Doc. at 279. To accomplish these goals, the Presidential memorandum specifically states that, in conducting analyses required by NEPA, "[e]ach Federal agency shall analyze the environmental effects, including human health, economic and social effects, of Federal actions, including effects on minority communities and low-income communities." *Id.* at 280.

It is the NRC's position that, as an independent regulatory agency, the NRC is not mandatorily subject to Executive Order 12898. Nevertheless, on March 31, 1994, the then Chairman of the Commission wrote the President stating that the NRC would carry out the measures in the Executive Order. In furtherance of this agency commitment, the NRC has participated in the Interagency Working Group on Environmental Justice created by the Executive Order and the NRC has drafted an environmental justice strategy as called for by the President's order.

Although Executive Order 12898 does not create any new rights that the Intervenor may seek to enforce before the agency or upon judicial review of the agency's actions, the President's directive is, in effect, a procedural directive to the head of each executive department and agency that, "to the greatest extent practicable and permitted by law," it should seek to achieve environmental justice in carrying out its mission by using such tools as the National Environmental Policy Act. Pursuant to the President's order, there are two aspects to environmental justice: first, each agency is required to identify and address disproportionately high and adverse health or environmental effects on minority and low-income populations in its programs, policies, and activities; and second, each agency must ensure that its programs, policies, and activities that substantially affect human health or the environment do not have the effect of subjecting persons and populations to discrimination because of their race, color, or national origin. Thus, whether the Executive Order is viewed as calling for a more expansive interpretation of NEPA as the Applicant suggests¹ or as merely clarifying NEPA's longstanding requirement for consideration of the impacts of major federal actions on the "human" environment as the Intervenor

¹ Applicant's Proposed Findings of Fact and Conclusions of Law (May 26, 1995) at 223-24 [hereinafter App. P.F.].

argues,² it is clear the President's order directs all agencies in analyzing the environmental effects of a federal action in an EIS required by NEPA to include in the analysis, "to the greatest extent practicable," the human health, economic, and social effects on minority and low-income communities.³

By voluntarily agreeing to implement the President's environmental justice directive, the Commission has made it fully applicable to the agency and, until that commitment is revoked, the President's order, as a practical matter, applies to the NRC to the same extent as if it were an executive agency. The NRC is obligated, therefore, to carry out the Executive Order in good faith in implementing its programs, policies, and activities that substantially affect human health or the environment. Further, because NRC licensing actions are activities that substantially affect human health and the environment, the Executive Order is applicable to the licensing of the CEC.

Thus, in carrying out the additional obligation the Commission has placed upon us in the hearing order (i.e., to ensure that the Staff's environmental review is adequate and in compliance with section 102(2)(A), (C), and (E) of NEPA), we necessarily also must ensure agency compliance with the President's environmental justice directive. Hence, contrary to the Applicant's assertion,⁴ Executive Order 12898 does impose duties on the NRC because the Commission has undertaken to carry out the President's directive, but no party to this proceeding has a remedy with regard to the manner in which the agency carries out its commitment to the President to implement Executive Order 12898.

C. Witnesses and Exhibits

Before turning to the substance of the environmental justice issues before us, we first briefly detail the witnesses and exhibits that were presented by the parties. Consistent with the Commission's burden-of-proof rule and in accordance with the stipulation of the parties, the Applicant presented its case first, followed by the Intervenor, and then the Staff. In support of its position on contention J.9, the Applicant presented the prefiled direct testimony of Peter G. LeRoy, the Licensing Manager of the CEC, and the prefiled testimony of a panel of witnesses consisting of B. William Dorsey, William H. Schaperkotter, Larry Engwall, Jesse B. Swords, and Peter G. LeRoy. Although the Applicant's

² Citizens Against Nuclear Trash's Proposed Reply Findings of Fact and Conclusions of Law Regarding Contention J.9 (June 26, 1995) at 2-3 [hereinafter CANT R.F.].

³ In using the term human health and environmental "effects" in Executive Order 12898 and the accompanying memorandum, the President's order tracks the regulations of the Council on Environmental Quality ("CEQ") that define "effects" to include both direct and indirect effects and states that "[e]ffects includes ecological (such as the effects on natural resources and on the components, structures, and functioning of affected ecosystems), aesthetic, historic, cultural, economic, social, or health, whether direct, indirect, or cumulative." 40 C.F.R. § 1508.8(b). See also 40 C.F.R. § 1508.14.

⁴ App. P.F. at 227.

witnesses appeared as a single panel, the two sets of testimony are separately numbered and appear bound in the record one after the other. (LeRoy fol. Tr. 840; Dorsey et al. fol. Tr. 840.)

Mr. LeRoy was responsible for compiling the information in the Applicant's ER and several ER amendments on the potential environmental, economic, and sociological impacts associated with the CEC. (LeRoy at 1-2 fol. Tr. 840.) He also had primary responsibility for the preparation of section 7 of the ER that describes the CEC site selection process, although Mr. LeRoy had no direct involvement in the siting process, having first become involved with the CEC in July 1989. (*Id.* at 1; Dorsey et al. at 5-6 fol. Tr. 840.)

Mr. Dorsey is employed by Fluor Daniel, Inc.,⁵ as Director of Siting and Consulting Services, a position he has held since 1974. In that capacity, he is responsible on a worldwide basis for coordinating, directing, and performing consulting services for industrial clients in all areas of project development, including feasibility studies, site location analyses, and management consulting. From approximately March 1987 through November 1989, he provided services under contract to one or more of the original participants of the venture that subsequently became LES as a site selection consultant and he directed and had overall responsibility for the site selection process for the CEC. Mr. Dorsey has earned a BA degree in economics and an MBA degree and he has more than 25 years of experience in site selection for industrial facilities and has been involved in hundreds of siting projects while at Fluor Daniel. (Dorsey et al. at 1-2, 5 & Attach. 1 fol. Tr. 840.)

Mr. Schaperkotter, who also is employed by Fluor Daniel, Inc., reported to Mr. Dorsey at the beginning of the CEC site selection process. He holds a BS degree in business administration and an MBA degree and he served as Manager of Facility Siting and Consulting Services from 1984 through 1988. During this time, he supervised dozens of site selection projects for industrial facilities and, from the spring of 1987 until the end of 1988 when he was promoted and transitioned out of his position, he had principal operational responsibility for the siting of the CEC. He also was involved in the preparation of section 7 of the ER in 1990. (Dorsey et al. at 2-3, 6 & Attach. 2 fol. Tr. 840.)

At the time of the hearing, Mr. Engwall was employed by Fluor Daniel, Inc., as an Operations Coordinator. He has earned a BS degree in engineering and an MBA degree. From approximately March 1989 to January 1990, he worked in the Facility Siting and Consulting Services Group. In April 1989 he was assigned principal operational responsibility for the siting of the CEC

⁵ Fluor Daniel, Inc., is involved in the LES project as the parent corporation of Claiborne Fuels, Inc., the sole general partner of the Delaware limited partnership, Claiborne Fuel, L.P., which is a LES general partner. Fluor Daniel, Inc., is, in turn, a wholly owned subsidiary of Fluor Corporation. (Dorsey et al. at 11 fol. Tr. 840.) See LBP-96-25, 44 NRC at 379.

and concluded his involvement with the CEC in November 1989. Before Mr. Engwall began work on the CEC project, he received several weeks of training in site selection. After completing the CEC site selection, he worked on several other site selection projects and then moved into other areas at Fluor Daniel. (Dorsey et al. at 3, 6 & Attach. 3 fol. Tr. 840; Intervenor's Exhibit I-RB-56, at 9-10.)

Mr. Swords is employed by Duke Engineering and Services, Inc., as an Engineering Manager.⁶ He holds a BS degree in engineering and has approximately 16 years of experience in the nuclear industry, including 4 to 5 years of experience in site selection for nuclear facilities. In the last stages of the CEC siting process, from June 1989 until November 1989, he provided technical site selection services with regard to the physical evaluation of specific sites under contract to LES. He also was involved in drafting section 7 of the ER in 1990. (Dorsey et al. at 4, 6 & Attach. 4 fol. Tr. 840.)

The prefiled direct testimony of the Applicant's witnesses was admitted pursuant to a pretrial stipulation of the parties and without further objection at the hearing. (Tr. 840.) Because the Applicant did not offer these witnesses as experts and, in light of the parties' admissibility stipulation, the Board did not rule at the hearing on the qualifications of these witnesses as experts. Obviously, however, as the LES official responsible for compiling the information in the ER on the site selection process and on the various impacts associated with the CEC, Mr. LeRoy was qualified to testify concerning that information. Additionally, we find that, as participants in the CEC site selection process, Mr. Dorsey, Mr. Schaperkotter, and Mr. Swords are qualified to testify concerning that process and also are qualified by knowledge and experience to testify as experts on site selection for industrial facilities. Further, we find that, as a participant in the process, Mr. Engwall is qualified to testify concerning that process but we do not find him qualified as an expert on industrial facility site selection.⁷

In support of its contention J.9, the Intervenor presented the testimony of Dr. Robert D. Bullard, Ware Professor of Sociology at Clark Atlanta University. (Bullard at 1 fol. Tr. 853.) He holds an MA degree in sociology from Clark

⁶ Duke Engineering and Services, Inc., is a subsidiary of Duke Power Company (Swords Tr. 953) which, in turn, is a LES general and limited partner. See LBP-96-25, 44 NRC at 380.

⁷ Pursuant to a stipulation of the parties the following Applicant exhibits were admitted into evidence relating to contention J.9: Applicant's Exhibit 16, LES letter to NRC dated March 30, 1992 (with attachment A containing response to NRC request for additional information) (App. Exh. 16); Applicant's Exhibit 18, Letter dated December 8, 1994, from Robert L. Draper, Winston & Strawn, Washington, D.C., to Diane Curran, Harmon, Curran, Gallagher & Spielberg, Takoma Park, Maryland (with enclosure of 1990 U.S. Census data for Homer, Louisiana) (App. Exh. 18); Applicant's Exhibit 19, Copies of Claiborne Enrichment Center "Community Newsletter" (App. Exh. 19); Applicant's Exhibit 20, State of Louisiana Air and Water Permits for LES (App. Exh. 20); Applicant's Exhibit 23, Market Search Corporation, Louisiana Quality of Life Survey (July 1989) (App. Exh. 23); Applicant's Exhibit 24, Market Search Corporation, Louisiana Quality of Life Survey (Sept. 1990) (App. Exh. 24); Applicant's Exhibit 25, LES letter to NRC dated September 29, 1994 (with enclosures containing ER Revision 17, SAR Revision 20, and License Application Revision 10) (App. Exh. 25). (Tr. 981-82.) Previously, the Applicant's ER, Applicant's Exhibit 1(h), which is relevant to contention J.9, was admitted into evidence. (Tr. 31.)

Atlanta University and a PhD in sociology from Iowa State University. Dr. Bullard has worked, conducted research, lectured, and written prolifically in the areas of urban land use, housing, community development, industrial facility siting, and environmental quality for more than 15 years and his scholarship and activities have made him one of the leading experts on environmental justice. He currently serves on the United States Environmental Protection Agency National Justice Advisory Council. Of the many works he has written, Dr. Bullard's book *Dumping in Dixie: Race, Class and Environmental Quality* (Westview Press 1990) has become a standard text in the environmental justice field. He also authored *Confronting Environmental Racism: Voices from the Grassroots* (South End Press 1993) and *Unequal Protection: Environmental Justice and Communities of Color* (Sierra Club Books 1994). Most recently he co-edited *Residential Apartheid: The American Legacy* (UCLA Center for Afro-American Studies Publications 1994). (*Id.* at 1-2; Intervenor's Exhibit I-RB-48.)

The Intervenor offered Dr. Bullard's prefled direct testimony as his expert opinion on contention J.9 and that of an expert in socioeconomic impact analysis. (Tr. 843-44.) His direct testimony was admitted pursuant to a stipulation of the parties and without further objection at the hearing. (Tr. 853.) We find that Dr. Bullard is qualified by education, knowledge, and experience to testify as an expert on the issues involved in contention J.9.⁸

⁸ Pursuant to a stipulation of the parties the following Intervenor exhibits were admitted into evidence relating to contention J.9: Intervenor's Exhibit I-RB-48, Vita of Robert D. Bullard (I-RB-48); Intervenor's Exhibit I-RB-49, Executive Order 12898, "Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations" (Feb. 11, 1994) and accompanying Memorandum for the Heads of All Departments and Agencies (Feb. 11, 1994) (I-RB-49); Intervenor's Exhibit I-RB-50, EPA Draft Environmental Justice Strategy for Executive Order 12898 (Jan. 1995) (I-RB-50); Intervenor's Exhibit I-RB-51, NRC Draft Strategic Plan — Environmental Justice (undated) (I-RB-51); Intervenor's Exhibit I-RB-52, Comment of Eula Mae Malone, Center Springs community, on scoping of EIS (I-RB-52); Intervenor's Exhibit I-RB-53, Handwritten map of Center Springs and Forest Grove communities prepared by Norton Tompkins (1992) (I-RB-53); Intervenor's Exhibit I-RB-54, Letter dated June 25, 1991, from Charles J. Haughney, Chief, Fuel Cycle Safety Branch, NRC, to LES, Attention W. Howard Arnold (I-RB-54); Intervenor's Exhibit I-RB-55, Portions of deposition of William S. Schaperkottter (Dec. 21, 1994) (I-RB-55); Intervenor's Exhibit I-RB-56, Portions of deposition of Larry Engwall (Jan. 26, 1995) (I-RB-56); Intervenor's Exhibit I-RB-57, Portions of deposition of B. William Dorsey (Dec. 21, 1994) (I-RB-57); Intervenor's Exhibit I-RB-58, Map and Analysis, "Poor Households as Percent of Total County Households — 1989, Thirteen Southern States," Southern Regional Council, Voting Rights Programs (Aug. 1993) (I-RB-58); Intervenor's Exhibit I-RB-59, Map and Analysis, "Black Population as Percent of Total County Population — 1990 and Congressional Districts, Eleven Southern States," Southern Regional Council, Voting Rights Programs (Sept. 1993) (I-RB-59); Intervenor's Exhibit I-RB-60, Letter dated November 2, 1994, from Robert L. Draper, Winston & Strawn, Washington, D.C., to Diane Curran, Harmon, Curran, Gallagher & Spielberg, Takoma Park, Maryland (I-RB-60); Intervenor's Exhibit I-RB-61, "CEPP, Centrifuge Enrichment Plant Project, Site Selection," Larry Engwall, Project Manager (May 17, 1989) (I-RB-61); Intervenor's Exhibit I-RB-62, Letter dated July 30, 1990, from A.M. Segrest, Manager, Projects and Administration, Duke Engineering & Services, Inc., to R.D. Belprez, Fluor Daniel, Inc. (with attachment) (I-RB-62); Intervenor's Exhibit I-RB-63, Fluor Daniels, "Site Recommendation Report for the Centrifuge Enrichment Plant Project" (Aug. 1989) (I-RB-63); Intervenor's Exhibit I-RB-64, Memo to File from Peter G. LeRoy (June 13, 1990) (I-RB-64). (Tr. 853.)

Additionally, the following Intervenor exhibits that were not subject to the parties' admissibility stipulation were admitted into evidence without objection or, in the case of I-RB-68, after the Applicant withdrew its objection:

(Continued)

In support of its position on contention J.9, the Staff presented the testimony of Merri L. Horn, Dr. Ibrahim H. Zeitoun, and Harry Chernoff. (Horn et al. fol. Tr. 904.) Ms. Horn is an environmental engineer in the Enrichment Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards. She holds a BS degree in physics and an MS degree in environmental engineering and she is the Environmental Project Manager for the CEC license application. (*Id.* at 1 & Attach. 1.) Dr. Zeitoun is employed by Science Applications International Corporation ("SAIC") as a Senior Environmental Analyst and he has earned both an MS degree and a PhD in fisheries biology. He is the SAIC project manager for the NRC contract to prepare the EIS for the CEC and has over 20 years of experience in directing and supporting multidisciplinary programs and projects in the areas of waste management, energy, and the environment. (*Id.* at 1 & Attach. 2.) Mr. Chernoff is also employed by SAIC as a Senior Economist and he has over 15 years of experience in energy economics, research and development program analysis, energy cost modeling, policy and regulatory analysis, and socioeconomics. He has earned a BS degree in economics and an MBA degree and he participated in preparing the EIS for the CEC. (*Id.* at 1 & Attach. 3.)

Pursuant to the pretrial stipulation of the parties and without further objection at the hearing, the prefiled direct testimony of the Staff witnesses was admitted. (Tr. 904.) We find that Ms. Horn, as the Staff's primary regulator with regard to the environmental impact analysis in the FEIS, and Dr. Zeitoun and Mr. Chernoff, as participants in the preparation of the FEIS for the CEC, are qualified to testify on the matters raised in their prefiled testimony.⁹

II. DISCRIMINATION ELEMENT OF ENVIRONMENTAL JUSTICE

Although the Intervenor's contention was filed before the President issued Executive Order 12898, CANT's contention J.9 is aimed at two concerns that are components of the Executive Order as well. Contention J.9 essentially asserts

Intervenor's Exhibit I-RB-65, LES Site Selection Files, "Numerical Listing (1-58) of Potential Sites" (I-RB-65); Intervenor's Exhibit I-RB-66, LES Site Selection Files, 4" x 8" Louisiana topographical map listing potential sites (32091-A1-TM-100) (1982) (I-RB-66); Intervenor's Exhibit I-RB-67, 1990 U.S. Bureau of the Census Data for Claiborne Parish, Louisiana (I-RB-67); Intervenor's Exhibit I-RB-68, Population by Race Living Within One Mile of LES Candidate Sites derived from U.S. Bureau of the Census PL 94-171 data on CD-ROM and TIGER/Line files (I-RB-68); Intervenor's Exhibit I-RB-69, Map, Claiborne Parish, 1990 Enterprise Zones (Oct. 1994) (I-RB-69). (Tr. 845, 853, 883, 987.)

⁹ Without objection, Staff Exhibit 3, Letter dated March 10, 1995, from Maria E. Lopez-Otin, NRC Environmental Justice Coordinator, to Kathy Aterno, Chair, Environmental Justice Subcommittee for Policy and Coordination, U.S. Environmental Protection Agency (with enclosure of final NRC Environmental Justice Strategy) (Staff Exh. 3), was offered into evidence by the Staff and admitted. (Tr. 1006.) Previously, the Staff's FEIS, Staff Exh. 2, which is relevant to contention J.9, was admitted into evidence. (Tr. 501.)

that the Applicant's ER and the Staff's FEIS have not adequately weighed the negative economic and sociological impacts on the minority communities of Forest Grove and Center Springs caused by closing Forest Grove Road that now joins them and placing the facility in the midst of these communities — a siting practice that follows a national pattern of locating hazardous facilities in minority communities. Further, the contention asserts that there has been no attempt to avoid or mitigate the disparate impact of the facility on this minority community. Thus, the Intervenor's contention has the same general focus as the President's environmental justice directive: disproportionate impacts on a minority population and racial discrimination.

Indeed, all parties apparently agree that the CEC will affect residents of a low-income minority populated community and that consideration of the environmental justice implications of the project is warranted. Similarly, all parties presented evidence on these factors with respect to contention J.9. In this Part II, therefore, we consider the discrimination aspect of environmental justice with respect to the Applicant's site selection process, a process that both contention J.9 and the Intervenor's expert witness charge was racially biased.

A. The CEC Siting Process

The site selection process that ultimately led to the selection of the LeSage property as the site for the CEC began in the first half of 1987 and, after several stops and starts, concluded in the fall of 1989. (Dorsey et al. at 5-6, 12, 22, 25 fol. Tr. 840.) The process took place before the Applicant, Louisiana Energy Services, L.P., was formed in 1990 and was conducted by employees of Fluor Daniel, Inc., under contract to one or more of the original venturers in the project that subsequently became partners in LES. (*Id.* at 10-11.) Representatives of the original participants in the venture comprised the Steering Committee that, *inter alia*, oversaw the selection process, participated in formulating the various site selection criteria, and acted upon the recommendations of Fluor Daniel. (*Id.* at 13, 16, 21.)¹⁰

The CEC siting process consisted of a number of phases and the Applicant's description of the siting process is set forth in the Applicant's ER. (App. Exh. 1(h), at 7.1-1 to -11.) The Staff's recitation of the siting process in the FEIS reproduces that set forth in the ER. (Staff Exh. 2, at 2-3 to -20.) A second description of the siting process is contained in Intervenor's Exhibit I-RB-63, Fluor Daniel's "Site Recommendation Report for the Centrifuge Enrichment Project" (Aug. 1989). That August 24, 1989 report, prepared by Mr. Engwall

¹⁰ Even though LES had not yet been formed at the time the CEC site was selected, all parties nevertheless refer to the site selection process as though LES conducted it. For ease of reference, we generally follow that convention, recognizing that it is technically inaccurate.

and submitted to the Steering Committee by Fluor Daniel, is the report that the Steering Committee had before it in making the final site selection. Clearly, as the Applicant's witnesses testified, the Fluor Daniel report was the principal document in the site selection process and a key document factored into the description of the site selection process in section 7 of the Applicant's ER. (Dorsey et al. at 44, 48 fol. Tr. 840.) For current purposes, it suffices to note that, although similar, the description of the site selection process contained in the Applicant's ER and the Fluor Daniel Report do not reflect identical phases for the selection process or the same site selection criteria or even the same number of criteria for the various phases of the selection process. We recognize that some of these differences are significant; however, to minimize confusion, we refer to the phases of the process used in the ER, which also appear in the FEIS and were used in the testimony of the Applicant's and the Intervenor's witnesses.

The CEC site selection process began with a coarse screening of the forty-eight contiguous states to identify a region of the United States for the facility. This Coarse Screening Phase applied various selection criteria involving the service area of sponsoring electric utilities, transportation distances, and seismic and severe storm factors. In October 1987, the siting consultants recommended northern Louisiana to the Steering Committee as the regional location for the facility and the Steering Committee adopted this recommendation. (Dorsey et al. at 10, 21 fol. Tr. 840; App. Exh. 1(h), at 7.1-2 to -5.)

Because of a hold on the project, it was not until the spring of 1988 that the site selection consultants conducted what the ER labels a two-phase intermediate screening process to select the most suitable host community. (Dorsey et al. at 15, 22; App. Exh. 1(h), at 7.1-5.) In Intermediate Phase I, communities across northern Louisiana within 45 miles of Interstate 40 were solicited with the assistance of the Louisiana Department of Economic Development. The candidate communities were asked to nominate potential sites based on a set of criteria that, *inter alia*, indicated the proposed facility was a chemical plant. In answer to the solicitation, 21 communities in 19 parishes with over 100 sites responded and expressed an interest in hosting the project. (Dorsey et al. at 11, 15, 24, 28; App. Exh. 1(h), at 7.1-5 to -6.)

According to the ER, during Intermediate Phase I, the site selection personnel then visited each of the communities and, applying a second set of criteria, reduced to nine the number of candidate communities. (App. Exh. 1(h), at 7.1-6.) Actually, however, during the spring and summer of 1988, only Mr. Schaperkotter visited nineteen of the twenty-one communities and met with or spoke with representatives of the other two communities. Specifically, he spoke by telephone with the mayor of Farmerville and eliminated that community. He also met in Shreveport with members of a regional economic development group representing Claiborne Parish and the town of Homer and learned that they were

busy pursuing another project at that time. Using reconnaissance-level data, Mr. Schaperkotter eliminated twelve communities for failing to meet one or more of the Intermediate Phase I criteria, leaving nine candidate host communities of the original twenty-one communities. (Dorsey et al. at 25, 28-30 fol. Tr. 840.) Although Mr. Schaperkotter did not visit Homer or any site in Claiborne Parish, the ER indicates Homer was one of the remaining nine candidate communities. (App. Exh. 1(h), at 7.1-6 & Fig. 7.1-6b.)

The purpose of the second phase of intermediate screening was to select a host community from the nine communities still under consideration. (Dorsey et al. at 25 fol. Tr. 840; App. Exh. 1(h), at 7.1-6.) When Mr. Schaperkotter left the siting group at Fluor Daniel in late 1988, he had completed most of the work for Intermediate Phase I. The project was again dormant until the spring of 1989 when Mr. Engwall was assigned principal operating responsibility for what the ER describes as Intermediate Phase II. (Dorsey et al. at 32-33 fol. Tr. 840.)

During this phase, Mr. Engwall scored the remaining nine candidate communities against another set of criteria that had been refined and expanded from those used in the first intermediate phase. (*Id.* at 22-23, 34-35.) In ranking the candidate communities he employed the Kepner-Tregoe ("K-T") method of decisional analysis. The K-T decisional analysis method is a widely used means for comparing alternatives on the basis of multiple criteria using a ten-point weighted scoring system in which criteria are divided into those that must be met ("musts") and those that are desirable ("wants"), with the wants weighted according to relative importance.¹¹ (*Id.* at 34; App. Exh. 1(h), at 7.1-6.) Further, in applying each "want" criterion to an alternative, the top rated alternative for that criterion always gets a ten and each of the other alternatives is compared relative to the best one. (Engwall Tr. 947.)

When assigned to the project in April 1989, Mr. Engwall visited a number of the communities previously visited by his predecessor to learn more about Mr. Schaperkotter's evaluative process. His visits included several communities that had been eliminated in Intermediate Phase I because they had expressed a renewed interest or proposed additional sites. Mr. Engwall also visited each of the nine remaining candidate communities, including Homer, which he visited for the first time on May 22, 1989. (Dorsey et al. at 26 fol. Tr. 840; Engwall Tr. 936.) In every community, Mr. Engwall viewed nominated sites and, according to his report to the Steering Committee, half of the fifteen criteria he applied were related to community characteristics and the other half were site specific. (I-RB-63, at 20.) In any event, as long as there was at least one site in each community meeting the established criteria the community remained in contention. (Dorsey

¹¹ In referring to K-T decisional analyses in the ER, the Applicant references Charles H. Kepner & Benjamin B. Tregoe, *The New Rational Manager*, Princeton Research Press (1981). (App. Exh. 1(h), at 7.1-12.)

et al. at 35 fol. Tr. 840.) Mr. Engwall assigned values for the nine communities, in consultation with Mr. Schaperkotter and Mr. Dorsey. (*Id.* at 36.) Based on Mr. Engwall's scoring, Homer was the highest rated community, with Winnsboro the runner up. (App. Exh. 1(h), at 7.1-8.) The Steering Committee then selected Homer as the host community. On June 9, 1989, the then Senator of Louisiana, Bennett Johnson, came to Homer and announced that it had been selected as the CEC host community. (Bullard at 57 fol. Tr. 853.)

After selecting Homer as the host community, the ER states that a fine screening process, in two phases, was employed to obtain the three most preferred sites from the six sites nominated by Homer community leaders. (App. Exh. 1(h), at 7.1-9.) In what the ER describes as Fine Screening Phase I, Mr. Engwall scored each of the six sites using the K-T decisional analysis against another set of criteria developed in conjunction with the Steering Committee. (Dorsey et al. at 39 fol. Tr. 840; App. Exh. 1(h), at 7.1-9.) Although eleven sites in Claiborne Parish were initially nominated by community leaders, five sites were immediately dropped by Mr. Engwall for failing to meet the selection criteria and only six sites were seriously considered and scored. (Engwall Tr. 944.) On the basis of the K-T analysis, the LeSage site was top rated and recommended for selection, pending confirmatory onsite studies. The second and fourth rated sites, the Emerson and Prison sites, respectively, also were carried to the next phase as alternatives to the LeSage property. The third most preferred site, the Baptist Children's Home site, was dropped for failing to meet the mandatory low flood risk criterion. (App. Exh. 1(h), at 7.1-10.)

During Fine Screening Phase II the three remaining sites were examined in more detail to select a final site. At this juncture, Mr. Swords, an engineer, joined the siting process. (Dorsey et al. at 39, 41 fol. Tr. 840.) A number of technical criteria relating to, *inter alia*, the cost of site work and grading, preliminary geotechnical evaluation, and the cost of providing electric power to the site were added to the criteria used in the first phase of fine screening. Again using K-T decisional analysis, Mr. Engwall apparently scored the three sites, with the LeSage property receiving the highest rating, followed by the Emerson site, and then the Prison site. (*Id.* at 39; App. Exh. 1(h), at 7.1-10 & Fig. 7.1-9.) The Applicant's ER notes that "[a]ll three properties are adequate sites for locating the CEC and relatively indistinguishable in their environmental characteristics." (App. Exh. 1(h), at 7.1-11.) Because it was the highest rated site, however, the site selection consultants, in August 1989, recommended the LeSage property to the Steering Committee. (Dorsey et al. at 39; I-RB-63, at ES-1.) On November 3, 1989, the selection of the LeSage property was publicly announced. (App. Exh. 1(h), at 9.5-9.)

B. The Parties' Positions

All parties presented evidence on the question whether race was a consideration in the selection of the site for the CEC. In sum, the Applicant and the Intervenor took diametrically opposed positions, while the Staff took the position it found nothing in the Applicant's ER to indicate that racial considerations were a factor in the site selection.

1. The Applicant

All of the Applicant's witnesses on contention J.9 testified in their prefiled direct testimony that the CEC site selection process was not racially biased or based on racial considerations. Although not directly involved in the siting process but with primary responsibility in the year after the LeSage site had been selected for preparing section 7 of the Applicant's ER, the LES Licensing Manager, Mr. LeRoy, stated that he was unaware of any instance in which, or evidence that, the race or color of any individual or group of individuals was a factor in any decision regarding the siting of the CEC. Similarly, he stated he had no knowledge that the siting of the CEC involved any intent to discriminate against the communities of Forest Grove and Center Springs on the basis of race or socioeconomic status. (LeRoy at 33-34 fol. Tr. 840.) Further, he testified that, in his judgment, the site selection process was not biased in any regard. (Tr. 951.)

In like vein, the Fluor Daniel consultants that oversaw and conducted the site selection process, Messrs. Dorsey, Schaperkotter, and Engwall, and Mr. Swords, the Duke Engineering and Services, Inc., engineer who was involved in the technical analysis for Fine Screening Phase II, together stated that the racial mix or racial makeup of the local population was not considered as a site selection criterion. (Dorsey et al. at 24 fol. Tr. 840.) Together these witnesses also stated that they were unaware of any instance in which, or evidence that, the race or color of any individual or any group was a factor in any decision concerning the siting of the facility. Further these witnesses together stated that the siting of the CEC did not involve any intent to discriminate against the communities of Forest Grove or Center Springs on the basis of race or socioeconomic status. (*Id.* at 48-49.) Finally, each of these witnesses testified that, in his judgment, the site selection process was not biased in any regard. (Tr. 951.)

2. The Intervenor

Intervenor witness Dr. Bullard in his prefiled direct testimony stated that, in his opinion, the process for selecting the CEC site was, among other things, biased and that racial considerations were a factor in the site selection process.

(Bullard at 39, 43 fol. Tr. 853.) Dr. Bullard based his conclusion that the CEC siting process was racially discriminatory on four major points. According to Dr. Bullard, the first factor and the most significant indication that institutionalized racism played a part in the site selection, was the fact that, at each progressively narrower stage of the site selection process, the level of poverty and African Americans in the local population rose dramatically, until it culminated in the selection of a site with a local population that is extremely poor and 97% African American. (*Id.* at 43.) Specifically, Dr. Bullard stated:

This progressive trend, involving the narrowing of the site selection process to areas of increasingly high poverty and African American representation, is also evident from an evaluation of the actual sites that were considered in the Intermediate and Fine Screening stages of the site selection process. At my request, the American Civil Liberties Union of Virginia performed an analysis, using census track data, of the percentage of black population within a one mile radius of 78 of the 79 sites that LES claims it seriously considered as candidate sites.¹²¹ The ACLU's analysis shows that the aggregate average percentage of black population for a one mile radius around all of the 78 sites examined (in 16 parishes)¹²² is 28.35%. When LES completed its initial site cuts, and reduced the list to 37 sites within nine communities (parishes), including Homer, the aggregate percentage of black population rose to 36.78%. When LES then further limited its focus to six sites in Claiborne Parish, the aggregate average percent black population rose again, to 64.74%. The final site selected, the "LeSage" site, has a 97.1% black population within a one-mile radius.

¹²¹Because LES' site selection documentation is so contradictory, it is difficult to determine how many sites were actually considered at any particular point in time by LES. However, counsel for LES stated in discovery that an undated document entitled "Numerical listing (1-58) of potential sites" [I-RB-65], and a "Huge topo map — 1982 Bastrop/Louisiana — Mississippi (32091-[A])-TM-100" [I-RB-66] provide the most comprehensive listing of sites that were considered. See letter from Robert L. Draper to Diane Curran (November 2, 1994) identifying [these exhibits] as providing the most comprehensive listing of sites that received serious consideration in the site selection process. [I-RB-60]. Based on these documents, the ACLU was able to identify, by description and/or map location, 79 candidate sites. Because one of these sites, the Armistead Cagean site, was identified on the list of 58, but was not clearly identified on the map, it was not considered in the analysis.

¹²²The twenty sites that were not identified on the list of 58 sites were placed in the appropriate parish by map location for computation purposes, rather than attempting to associate each unidentified site with a particular community. An exception to this was made for Homer, where six sites that were not included in the list of 58 sites were all identified in the draft and final EIS as being considered connected with the town of Homer.

(*Id.* at 46-47.) The tabulation of the ACLU analysis was received in evidence as Intervenor's Exhibit I-RB-68.

The second point showing discrimination according to Dr. Bullard, is LES' application in Fine Screening Phase I of the "low adjacent population within a 2-mile radius" criterion in a biased and discriminatory manner in connection with the LeSage and Emerson sites to protect the white, middle class lifestyle on Lake Claiborne next to the Emerson site. (Bullard at 44, 51-52 fol. Tr. 853.) Relying on Mr. Engwall's deposition testimony (I-RB-56, at 105-06), Dr. Bullard testified that, as the principal person responsible for site selection process

at this stage involving winnowing the six Homer sites to three, Mr. Engwall initially evaluated and scored the low population criterion for the LeSage site based upon an “eyeball assessment.” As Mr. Engwall described this process, he drove along the road through Forest Grove and every now and then he drove up a dirt road where he saw “a small cluster of houses” and “boarded up houses.” From this survey, Mr. Engwall concluded that in this area there were “maybe ten people living there at most.” (I-RB-56, at 105-06; Bullard at 52 fol. Tr. 853.) Dr. Bullard further testified that it did not appear Mr. Engwall drove through Center Springs at all. As a result of this survey, Mr. Engwall gave the LeSage site a “low population” score of 9 out of a maximum of 10 and, when multiplied by the “want” weight of 8, it yielded a weighted score of 72. (Bullard at 52 fol. Tr. 853.)

Dr. Bullard declared that, in fact, there are 150 people living in Forest Grove and 100 in Center Springs. According to Dr. Bullard, had Mr. Engwall taken the most basic measures to assess population levels, such as consulting aerial photographs or county land records or talking to inhabitants of Forest Grove, he would not have rendered this African American population essentially invisible or taken the condition of the housing as empirical evidence of the number of people living there. (*Id.* at 52.)

Next, Dr. Bullard asserted, Mr. Engwall compounded the problem by using invalid and biased considerations in comparing the population level of the LeSage site to that of the Emerson site. The Emerson site, which was the overall second highest rated site in Fine Screening Phase I, was given a “low population” score of 7, yielding a significantly lower weighted score of 56. Again relying on Mr. Engwall's deposition testimony (I-RB-56, at 102, 105, 108-10), Dr. Bullard asserted that the Emerson site score also was based on Mr. Engwall's observations from driving around the site, which led him to conclude that between 50 and 100 people actually lived there. Yet when asked what he saw that caused him to score the site a seven, Mr. Engwall answered “[p]robably the proximity to the lake.” Mr. Engwall went on to explain that “[w]e just felt opinion-wise people would probably not want this plant to be close to their pride and joy of their lake where they go fishing.” (I-RB-56, at 109; Bullard at 53 fol. Tr. 853.) The significance of the lake, Dr. Bullard asserted, also was emphasized a few pages earlier in his deposition when Mr. Engwall testified that the Emerson site was rated neutral to slightly negative because

[i]t was right on the edge of this lake. This lake is a very nice lake. This lake is the pride and joy of this part of Louisiana, nice boating, nice homes along the lake. It was felt that an industrial facility real close to that lake would not be in keeping with the existing usage, which was nice homes, vacation and fishing, hunting. (I-RB-56, at 102.)

Based on Mr. Engwall's deposition testimony, Dr. Bullard concluded it was clear that quality of life considerations improperly affected Mr. Engwall's scoring of the low population criterion for the Emerson site given that, at this stage of the evaluation process, there were no site specific criterion related to quality of life. He further maintained that Mr. Engwall's biased judgment on the quality of life concern regarding the desirability of avoiding the lakeside site where white, middle class people lived was directly related to the relative scoring of the low population criterion. Dr. Bullard asserted that the total effect of Mr. Engwall's actions was to discriminate against the Forest Grove and Center Springs communities because their residents' lifestyle and socioeconomic status were on a much lower plane. (Bullard at 54-55 fol. Tr. 853.)

The third factor Dr. Bullard testified about was racial discrimination inherent in the Fine Screening Phase I criterion of not siting the facility within at least 5 miles of institutions such as schools, hospitals, and nursing homes. (*Id.* at 13, 43-44.) He asserted that by its own terms, this criterion is inherently biased toward the selection of sites in minority and poor areas because these areas generally lack institutions such as schools, hospitals, and nursing homes that are the focus of this criterion. Dr. Bullard stated that even though Forest Grove and Center Springs are 5 miles from the nearest town, there are no schools, hospitals, or medical facilities of any kind or, for that matter, any other service institution in either community. He stated that, while it is not necessarily inappropriate to attempt to site a hazardous facility in an area that is far from these institutions, this criterion cannot be applied equitably unless the process is enlightened by consideration of the demographics of the affected population. Otherwise, he stated, disadvantaged populations will invariably be favored as hosts for more hazardous facilities as is evidenced by the fact that minority communities already host a disproportionate share of prisons, half-way houses, and mental institutions. (*Id.* at 13.)

The fourth and final point, according to Dr. Bullard, was the use of various community support criteria in the selection process that had the effect of discriminating against the people of Forest Grove and Center Springs. He testified that during the siting process LES relied upon the opinions of Homer, a community 5 miles from the actual host community. This was inappropriate, he concluded because Homer stood to minimize the risks and maximize the benefit to itself by placing the facility a good distance from its own residents. In contrast, the actual host communities of Forest Grove and Center Springs were never informed of the siting decision until it was too late for the residents to affect the selection process. (*Id.* at 13-14.)

This was particularly significant, Dr. Bullard testified, because the principal criteria for site selection were support from the community and opinion leaders in the community. Indeed, LES considered it of primary importance that the facility

should be located in a locale where it would be considered a community asset.¹² Dr. Bullard testified, however, that, despite the importance of such community support, LES did not even recognize the existence of Forest Grove and Center Springs as communities, let alone consult their leaders. Instead, LES defined the “community” as Homer, a town 5 miles away whose government contains no representation from Forest Grove or Center Springs. Further, he declared that the concept of community leadership, which was key to the assessment of community support in the selection process was biased toward consultation with individuals who, rather than having an interest or stake in the welfare of Forest Grove or Center Springs, instead stood to benefit from imposing the risks of the facility on these neighboring communities while the community of Homer reaped the benefits. According to Dr. Bullard, the groups of community leaders with whom LES met and with whom it consulted to form its opinion of “community support,” “active and cohesive community leadership” and “community leader preferences,” were dominated by the Claiborne Parish Industrial Development Foundation — on which Forest Grove and Center Springs have no representatives — and elected officials from the towns of Homer and Haynesville, rather than Forest Grove and Center Springs. Thus, Dr. Bullard concluded that a facially neutral site selection process was perverted to give certain communities the discretion to decide who should accept the adverse impacts of the proposed facility. (*Id.* at 47-51.)

3. *The NRC Staff*

In chapter 2, section 2.3.1, of the FEIS at the end of its description of the LES site selection process, the Staff concludes that “the LES approach for selecting the site was reasonable.” (Staff Exh. 2, at 2-19.) Thereafter, in chapter 4, section 4.2.1.7.4, titled “Environmental Justice,” the Staff states, *inter alia*, that it considered environmental justice from the perspective of whether there is evidence LES selected the CEC site based on racial considerations. It states that, although many comments on the draft environmental impact statement alleged that LES deliberately chose the site because it is in an African American community, none cited any specific evidence to support the charge. In the FEIS, the Staff asserts that based on its review of the public comments and the LES description of the site selection process, it concluded that “[t]he LES process

¹² As evidence of the importance of this factor, Dr. Bullard noted that in Intermediate Phase II when the field had been narrowed to nine communities, “local support” was a criterion that had the highest possible scoring weight of 10. Similarly, he observed that, in both Intermediate Phases I and II, “active, cohesive community leadership” was evaluated and in Phase II (where K-T analysis was used for the first time) that criterion was given a “want” weight of 10. Finally, he indicated that, although at the Fine Screening stage when LES was choosing among the six Homer sites community support was no longer considered because it was deemed already to have been established in the selection of Homer, in choosing among the six sites, LES nonetheless gave a “want” weight of 10 to “community leader preferences.” (Bullard at 47-48 fol. Tr. 853.)

appears to be based solely on business and technical considerations” and it found “no specific evidence that racial considerations were a factor” in the process. (*Id.* at 4-34.)

In their prefiled direct testimony, the Staff’s witnesses, Ms. Horn and Dr. Zeitoun, reiterated the findings in the FEIS and stated that the LES site selection criteria “appeared to be objectively applied in each phase of the selection process; and none of the criteria appear to be based on racial considerations.” (Horn et al. at 12 fol. Tr. 904.) The Staff witnesses further testified, however, that “[t]he Staff did not conduct a detailed evaluation of the site selection process. The Staff did not evaluate each individual criterion and make a determination if that particular criterion was appropriate. The Staff only considered the information provided in the Environmental Report.” (*Id.*) Finally, Ms. Horn and Dr. Zeitoun reiterated that “[b]ased on the information in the Environmental Report, the Staff did not see any evidence that racial considerations were a factor in the site selection process.” (*Id.*)¹³

C. Licensing Board Determination

The nondiscrimination component of Executive Order 12898 requires that the NRC conduct its licensing activities in a manner that “ensures” those activities do not have the effect of subjecting any persons or populations to discrimination because of their race or color. 3 C.F.R. at 861. In the FEIS and in its prefiled direct testimony, the Staff stated that it sought to determine whether race played a role in the CEC site selection process by reviewing the information in the Applicant’s ER. In taking this action, the Staff necessarily recognized the agency’s obligation under the nondiscrimination component of the President’s environmental justice directive to make sure the site selection process conducted by the original venturers in what subsequently became the LES project was free from racial discrimination.

In the circumstances presented in this licensing action, however, by limiting its consideration to a facial review of the information in the Applicant’s ER, the Staff has failed to comply with the President’s directive. As we discuss more fully below, a thorough and in-depth investigation of the Applicant’s siting process by the Staff is essential to ensure compliance with the President’s nondiscrimination directive if that directive is to have any real meaning. Moreover, such a thorough Staff investigation is needed not only to comply with Executive

¹³In its proposed findings dealing with the site selection process, the Staff suggests that we approach the issue by “looking at the question of whether the selection process was overtly racist.” NRC Staff’s Proposed Findings of Fact and Conclusions of Law in the form of a Partial Initial Decision Regarding Contentions B, J, K, and Q (May 26, 1995) at 57.

Order 12898, but to avoid the constitutional ramifications of the agency becoming a participant in any discriminatory conduct through its grant of a license.

Racial discrimination in the facility site selection process cannot be uncovered with only a cursory review of the description of that process appearing in an applicant's environmental report. If it were so easily detected, racial discrimination would not be such a persistent and enduring problem in American society. Racial discrimination is rarely, if ever, admitted. Instead, it is often rationalized under some other seemingly racially neutral guise, making it difficult to ferret out. Moreover, direct evidence of racial discrimination is seldom found. Therefore, under the circumstances presented by this licensing action, if the President's nondiscrimination directive is to have any meaning a much more thorough investigation must be conducted by the Staff to determine whether racial discrimination played a role in the CEC site selection process.

Before turning to a discussion of the evidence in this proceeding, we wish to emphasize that our determination that the Staff's limited review of the description of the siting process set out in the ER was inadequate and that the Staff now must undertake a thorough investigation, is not intended as a criticism of the Staff. The obligations imposed upon the Staff by the Commission's commitment to the President to implement the provisions of the Executive Order are new to the agency. Because this agency's primary responsibilities historically have dealt with technical concerns, investigating whether racial discrimination played a part in a facility siting decision is far afield from the Staff's past activities. Indeed, because racial discrimination questions have not previously been involved in agency licensing activities, this is an area in which the Staff has little experience or expertise. Nevertheless, if the President's directive is to have any meaning in this particular licensing action, the Staff must conduct an objective, thorough, and professional investigation that looks beneath the surface of the description of the site selection process in the ER. In other words, the Staff must lift some rocks and look under them.

Substantial evidence presented by the Intervenor in this proceeding demonstrates why it is imperative that the Staff conduct such a thorough investigation. As we have noted, direct evidence of racial discrimination is rare. Nonetheless, the Intervenor's evidence, the most significant portions of which are largely un rebutted or ineffectively rebutted, is more than sufficient to raise a reasonable inference that racial considerations played some part in the site selection process such that additional inquiry is warranted. In so stating, we do not make specific findings on the current record that racial discrimination did or did not influence the site selection process. When stripped of its abundant irrelevant chaff, the record is simply inadequate, objectively viewed, to reach any conclusion with the requisite degree of confidence. A finding that the selection process was tainted by racial bias is far too serious a determination, with potentially longlasting consequences, to render without the benefit of a thorough and

professional Staff investigation aided by whatever outside experts as may be necessary. Additionally, the Applicant, because of the allocation of the burden of proof in the adjudicatory process and the nature of this particular subject matter, is, to some extent, in the position of proving a negative. Thus, in this instance any finding that racial considerations either did or did not play a part in the site selection process should be made only after the Staff has undertaken a complete and systematic examination of the entire process.

Looking to the record of this proceeding, the Intervenor's statistical evidence presented by Dr. Bullard and set out in Intervenor's Exhibit I-RB-68, shows that as the site selection process progressed and the focus of the search narrowed, the level of minority representation in the population rose dramatically. See *supra* p. 386. The Intervenor's analysis did not include one of the seventy-nine seriously considered proposed CEC sites because it was not clearly identified on the large map on which the siting consultants had marked the proposed sites. (Bullard at 46 n.121 fol. Tr. 853; see I-RB-66.) Of the remaining seventy-eight proposed sites, however, the Intervenor's analysis reveals that the aggregate average percentage of black population within a 1-mile radius of each of the sites across sixteen parishes is 28.35%. After the initial site cuts reduced the list to thirty-seven sites in nine parishes, including the sites in Claiborne Parish, the aggregate percentage of black population rose to 36.78%. Then, when the search narrowed to the six sites in Claiborne Parish, the aggregate average percent of black population increased to 64.74%. Ultimately, the process culminated in a chosen site with a black population of 97.1% within a 1-mile radius of the LeSage site, which is the site with the highest percent black population of all seventy-eight examined sites. (Bullard at 46-47 fol. Tr. 853; I-RB-68, at 2-4.) This statistical evidence very strongly suggests that racial considerations played a part in the site selection process. It does not, of course, rule out all possibility that race played no part in the selection process. Nonetheless, the Intervenor's statistical evidence clearly indicates that the probability of this being the case is unlikely. Certainly, the possibility that racial considerations played a part in the site selection cannot be passed off as mere coincidence.

For its part the Applicant did not attempt to rebut the Intervenor's statistical analysis with any statistical evidence of its own or present any witness challenging the statistical validity of the Intervenor's evidence.¹⁴ Rather, Mr. LeRoy,

¹⁴ Although at the hearing the Applicant did not challenge the Intervenor's statistical evidence with any statistical evidence or witnesses of its own, the Applicant, in its proposed findings (App. P.F. at 319 n.199), argues that it has no way of knowing whether the Intervenor's statistical data are correct and whether the site locations on which they are based were properly identified.

After having its initial objection sustained, Applicant withdrew its objection to Intervenor's Exhibit I-RB-68 (Tr. 883) so that exhibit was admitted into evidence. Thus, it is too late now for procedural arguments challenging that evidence. Further, as the Intervenor's exhibits show, the map used by the Intervenor to locate each of the proposed sites (I-RB-66) was turned over by the Applicant to the Intervenor during discovery from the Applicant's

(Continued)

the LES Licensing Manager, although not directly involved in the actual siting process, stated that the siting process was not biased in any way and that he was not aware of any instance in which, or evidence that, the race or color of any individual or group was a factor in any siting decision. (LeRoy at 33 fol. Tr. 840; Tr. 951.) He also testified that it was only coincidence that the selection process ended with a site that has a black population of 97.1% within a mile radius of it. (Tr. 965.) The three Fluor Daniel siting consultants, Messrs. Dorsey, Schaperkotter, and Engwall gave similar testimony, as did Mr. Swords, the Duke Engineering and Services, Inc., engineer involved in the last phase of the selection process. (Dorsey et al. at 48-49 fol. Tr. 840; Tr. 951.)

As we have already observed, we would not expect instances of racial discrimination to be admitted. Instances of racial bias are often rationalized in ways that avoid the question, so that a person can state, with conviction, that he or she did not discriminate even when objective evidence suggests otherwise. In so stating, it is not our intent to impugn the integrity of the Applicant's witnesses. Rather, our point is simply that this and similar testimony of the Applicant's witnesses does not adequately rebut the Intervenor's statistical evidence.¹⁵

In response to an inquiry from the Licensing Board on the statistical probability of coincidentally selecting a site that is 97.1% black within a one-mile radius from among the seventy-eight proposed CEC sites, Mr. Dorsey did testify that because of the selection criteria of a large site size and a low population area "the odds are very high that that is going to happen no matter where you go. It may not be 97-." (Tr. 966.) Mr. Dorsey then added that, if you are in Louisiana or Mississippi or some other states in this part of the country, "[i]t is simply the make-up of the rural areas within that region." (Tr. 967.) In this regard, Mr. LeRoy added that "[t]he rural population of Claiborne Parish, I believe, is about 60 percent African American." (Tr. 968.)¹⁶ Yet, at least with respect to Claiborne Parish (on which the record contains considerable data),

own site selection files. (I-RB-60.) The 79 proposed CEC sites marked on the map were placed there by the Fluor Daniel siting consultants during the selection process, not by the Intervenor, so the Applicant's complaint that it does not know how Dr. Bullard located the sites is well wide of the mark. Moreover, Dr. Bullard's prefiled direct testimony containing the methodology and results of the statistical analysis was served on the Applicant by overnight mail on February 24, 1995, so it had that information for well over 2 weeks before Dr. Bullard testified on March 16, 1995. Accordingly, the Applicant's post-hearing objections are without merit.

¹⁵The Applicant also argues that to accept as evidence of racial discrimination the Intervenor's testimony that at each progressive stage of the selection process the level of minority population rose dramatically, "would be to suggest that any attempt to build a facility in the vicinity of Forest Grove and Center Springs or similar communities is inherently racially discriminatory" (App. P.F. at 322) and "as a matter of law would deprive communities such as Forest Grove and Center Springs of the opportunity even to be considered as the site for a project." (App. P.F. at 323.) We do not agree. Any conclusion that the site selection process was racially biased necessarily would be an ultimate determination of fact based on the specific site selection process applied in this proceeding. If such a finding were made, it would not be a determination "as a matter of law" and it most certainly would not deprive depressed minority communities of the opportunity for future improvement.

¹⁶Interestingly, in the portion of his deposition admitted into evidence, Mr. Engwall testified that 90% to 95% of the entire population of Claiborne Parish lived in Homer and Haynesville, the two urban centers in the parish. (I-RB-56, at 104, 107.)

the record before us does not support the Applicant's assertion that the odds are very high that, because of the high percentage of blacks in the rural population, the black population around any rural site inevitably would be markedly higher than the racial makeup of the parish at large or the racial makeup of the rural population.¹⁷

In addition to this statistical evidence, the Intervenor presented additional evidence indicating that racial considerations played a role in the CEC site selection process. Based on Mr. Engwall's deposition testimony, Dr. Bullard also testified that, with respect to the LeSage and Emerson sites, Mr. Engwall applied the low population criterion during the Fine Screening Phase of the site selection process in a biased and discriminatory manner to protect the white, middle class lifestyles on Lake Claiborne next to the Emerson site. *See supra* pp. 386-88. (Bullard at 51-55 fol. Tr. 840.) A thorough and careful reading of all the parts of Mr. Engwall's deposition admitted in evidence clearly supports Dr. Bullard's assertion that racial and economic-based quality of life considerations influenced Mr. Engwall's scoring of the Emerson site. (I-RB-56 at 108-09, 102.) Overall, Dr. Bullard's testimony fairly recites and reasonably characterizes Mr. Engwall's deposition testimony on this point. At a minimum, that deposition testimony raises a strong inference that race and economic status played a role in the scoring of the two sites.

Moreover, Dr. Bullard's testimony on this matter was not persuasively and effectively rebutted. Mr. Schaperkotter testified that LES did not apply the low population criterion in a biased matter. (Tr. 929.) But Mr. Schaperkotter had left the project prior to that time. Instead, at the Fine Screening Phase of the site selection process, it was Mr. Engwall who had primary operational responsibility for the project and it was Mr. Engwall who visited and scored the LeSage and Emerson sites.

¹⁷ The record shows that the population of Louisiana is 30.8% African American. (Bullard at 45 fol. Tr. 840; I-RB-59.) Drawing on census data, the FEIS states that the population of Claiborne Parish is 17,405 and that 53.43% of the population is white and 46.09% black. (Staff Exh. 2, at 3-102 to -103.) Thus, there are slightly more than 8000 African Americans in Claiborne Parish. Although no party introduced census figures on the urban-rural breakdown of the population of Claiborne Parish or the racial makeup of that breakdown, that information can be reasonably derived from other record evidence. There are only two urban areas in Claiborne Parish, Homer and Haynesville, although there are numerous rural enclaves. The census data in Applicant's Exhibit 18 on Homer, the largest town in the parish, shows a black population of 2346 or 56.5% of the total population of 4152. (App. Exh. 18, at 16.) The radial sector map and corresponding population table in the Applicant's ER (App. Exh. 1(h), at Fig. 2.2-6 & Table 2.2-9) indicates that the population of Haynesville is approximately 3000. Hence, the total urban population of Claiborne Parish is approximately 7000 and the rural population is approximately 10,400. Therefore, approximately 60% of the total population of Claiborne Parish lives in rural areas. Even assuming the entire black population of the parish outside of Homer resides in rural areas and that no blacks live in Haynesville, the second urban center in the parish, the maximum percentage of blacks in the rural population would be less than 55%. Making the reasonable assumption that one-third of the population of Haynesville is black, then the rural black population of the parish is approximately 45% and thus essentially the same as the racial makeup of the parish population. In light of these population figures derived from the evidentiary record for Claiborne Parish, it is not at all apparent that the rural black population of the parish creates a situation where the "odds are very high" that any rural site in the parish would have a surrounding black population that is much higher than the racial makeup of the parish at large or the racial makeup of the rural black population.

Even more troubling, however, is Mr. Engwall's attempted revision at the hearing of his deposition testimony regarding how he assessed the population of the LeSage and Emerson sites that was neither credible nor convincing. At his deposition, Mr. Engwall no less than seven times testified under oath that he performed his evaluation of the population of the LeSage and Emerson sites by driving through the area and performing a visual or "eyeball" assessment. (I-RB-56 at 106; *id.* at 102-08.) Indeed, he even asked his questioner, Intervenor's counsel, "How else are you going to do it?" and indicated that, in his site selection training prior to his work on the CEC project, he learned to evaluate population by driving around and looking. (I-RB-56 at 106.) In his rebuttal testimony at the hearing, however, Mr. Engwall testified that although he had said that at his deposition, he later was looking through the siting files and saw a map that he recalled using to gather information on the proximity of houses near the Emerson and LeSage sites. He also declared that he remembered taking an airplane flight around three or four sites to get an idea of the population levels. He then stated it was this later information that he used in scoring the sites for the Kepner-Tregoe analyses (Tr. 931-32.)

The marked difference in Mr. Engwall's testimony on this matter from the time of his deposition to the time of trial causes us seriously to doubt the credibility of this revised explanation. Further, his demeanor at the hearing in responding to his counsel's question and the substance of his response, in particular the generality of that response, convince us that Mr. Engwall's earlier deposition testimony is a more accurate accounting of the process he used to gauge and score the population of the LeSage and Emerson sites.¹⁸ In the same vein, Mr. Engwall's attempt in his rebuttal testimony (Tr. 933) to distance himself from his earlier deposition testimony regarding the low population scoring for the Emerson site and his view that the proposed CEC facility was not compatible with the land uses around Lake Claiborne was neither credible nor persuasive.¹⁹ Accordingly, we find that this specific example of the application of a site selection criterion raises a reasonable inference, which was

¹⁸ For example, Mr. Engwall did not otherwise identify the "map" from the siting files that he "used to gather information on the proximity of houses near each one of the sites" (Tr. 932) nor was it introduced into evidence.

¹⁹ In its proposed findings, the Applicant suggests that Dr. Bullard provided no basis for his conclusion that the lakeside community around Lake Claiborne is white, middle class. (App. P.F. at 310 n.189.) Dr. Bullard's areas of expertise, however, include land use and minority housing (I-RB-48) and he testified that "it is very simple to tell who lives where. Given the demographics of the parish, given the nature of Forest Grove and residential segregation in this parish, it is fairly simple to look at the numbers and the charts and tell who lives where." (Tr. 874.) The Applicant presented no evidence of any kind that the residential community around Lake Claiborne was not a white, middle class area and that Dr. Bullard was incorrect in his description. Indeed, in light of the Bureau of the Census statistics in Intervenor's Exhibit I-RB-67 on the household incomes of white and black households in Claiborne Parish (I-RB-67 at 10), it is reasonably inferred that the "very nice lake" with "nice homes along the lake" that the Applicant's witness, Mr. Engwall, described (I-RB-56 at 102) are not the homes of Claiborne Parish African Americans.

not effectively rebutted by the Applicant, that racial bias played a part in the selection process.²⁰

To summarize, the Intervenor's statistical evidence and its evidence concerning the application of the low population criterion stand as significant probative evidence in the current record that racial considerations played a part in the site selection process. This evidence demonstrates that a thorough Staff investigation of the site selection process is needed in order to comply with the President's nondiscrimination directive in Executive Order 12898. The Intervenor did provide other evidence concerning the inherent racial bias in the fine screening criterion of siting the facility 5 miles from institutions such as schools, hospitals, and nursing homes and evidence on the manner in which various community opinion and support criteria in the selection process discriminated against the minority communities of Forest Grove and Center Springs. This evidence is, at most, only indirectly indicative that racial considerations played a part in the site selection process. Nevertheless, when coupled with the Intervenor's statistical evidence and its evidence concerning the application of the low population

²⁰ In his rebuttal testimony, Mr. LeRoy testified that prior to the hearing he had a house count performed that confirmed Mr. Engwall's scoring for the Emerson and LeSage sites. He stated that this drive-by survey showed approximately 140 houses within a 2-mile radius of the Emerson site and approximately 70 houses for the LeSage site. (Tr. 932.)

There are several reasons why Mr. LeRoy's testimony does not rebut effectively the inference of racial discrimination in the application of the population scoring criterion. That count has no real relevance to the quality of life considerations about the incompatibility of the proposed CEC facility with the white, middle class homes on the lake that we have found improperly influenced Mr. Engwall's scoring of the Emerson site relative to the LeSage site. In any event, using a house count instead of an actual population enumeration for determining the population around the LeSage site and that portion of Forest Grove within 2 miles of the Emerson site does not provide accurate information because the use of the standard multiplier of 2.8 persons per household undercounts minority households and yields totally unrealistic results. (Bullard Tr. 988-89.) Additionally, the Applicant's ER states that 50% of the houses located on Lake Claiborne within 5 miles of the LeSage site are not permanent residences. (App. Exh. 1(h), at 2.2-2.) Therefore, it appears that some significant portion, if not all, of those houses are included in Mr. LeRoy's house count. Hence, that house count does not reliably establish the population around the LeSage and Emerson sites.

Finally, in an effort to bolster its low population scoring defense, the Applicant argues that Intervenor's Exhibit I-RB-68 showing the population within 1 mile of the LeSage site as 138 and the population within 1 mile of the Emerson site as 393 effectively confirms the low population scoring of the two sites. Because the fine screening stage low population criterion is a 2-mile radius, the presence of a good portion of Lake Claiborne within 2 miles of the Emerson site precludes any accurate conclusion from the 1-mile radius figures. In sum, none of the evidence in the current record provides an accurate or reliable figure of the population within 2 miles of the Emerson and the LeSage sites. The record does clearly establish, however, that Mr. Engwall's count of 10 people for the LeSage site and 50 to 100 people for the Emerson site is not correct and that, contrary to his deposition testimony, 90% to 95% of the people in Claiborne Parish do not live in Homer and Haynesville. (I-RB-56, at 104, 105, 107.) Further, we note that the figures "characterized" from census data in the direct testimony of the Staff witnesses on the population and racial makeup of the area around the LeSage site, including the 1-mile site radius (Horn et al. at 11-12 fol. Tr. 904), is markedly different from the 1-mile radius around the site derived from the census data by the Intervenor in I-RB-68. But the Staff witnesses conceded that the numbers actually were much higher. (*Id.*)

criterion, this further Intervenor evidence raises concerns that deserve attention and should be further carefully analyzed as part of the Staff investigation.²¹

III. ENVIRONMENTAL IMPACTS

Although the Staff now must undertake a thorough investigation of whether racial considerations played a part in the CEC site selection process, we nevertheless turn to address the second concern of the Intervenor's environmental justice contention. In the event it is ultimately determined that racial considerations played a role in the site selection process, these findings would become

²¹ In his testimony, Dr. Bullard also claimed that the CEC site selection process was not the orderly, systematic process depicted in the Applicant's ER but rather a process that contained significant irregularities, gaps, and inconsistencies. He asserted that these numerous deficiencies raised an inference of bias in the site selection process. (Bullard at 55-66 fol. Tr. 853.) In light of our conclusion that the Staff must conduct a thorough investigation of the site selection process, we have not attempted to resolve all of the additional evidentiary disputes between the Intervenor and the Applicant over the various aspects of the selection process.

It should be noted, however, that a comparison of the Fluor Daniel Site Recommendation Report (I-RB-63) — the report before the Steering Committee when the Committee selected the LeSage site — with section 7 of the Applicant's ER (App. Exh. 1(h), at 7.1-1 to -11) does not support the Applicant's assertion that the description of the site selection process in the ER is consistent with the Fluor Daniel report. (Dorsey et al. at 46-48.) Even accepting the Applicant's characterization of the correlation between the site selection phases of the Fluor Daniel report and the phases stated in the ER (*id.* at 46), the criteria that the Fluor Daniel report states were applied at several phases of the selection process simply do not match the criteria that the ER states were applied at those corresponding stages. For example, the Applicant states that Phase III of the Fluor Daniel report corresponds to what is called Intermediate Phase I in the ER. (*Id.*) Yet of the 10 criteria applied at Phase III of the Fluor Daniel report (I-RB-63 at 18-19) 5 of those criteria (i.e., square site configuration, topography, no split ownership of land and mineral rights, site access, and wetlands) have no counterpart in the 10 criteria the ER states were applicable at Intermediate Phase I. (App. Exh. 1(h), at 7.1-6.) The Applicant also states that the First Stage of Phase IV of the Fluor Daniel report corresponds to Intermediate Phase II in the ER. (Dorsey et al. at 46 fol. Tr. 840.) Yet of the 15 criteria applied at the First Stage of Phase IV of the Fluor Daniel report (I-RB-63 at 20-23) at least 8 of those criteria (i.e., access control (must), low flood risk (must), low adjacent population, institutions within 5 miles, no airport within 5 miles, single owner, site size, and baseline environmental data) have no counterpart in the 14 criteria the ER states were applicable to Intermediate Phase II. (App. Exh. 1(h), at 7.1-7 to -8.)

Moreover, given the siting criteria that the Fluor Daniel report states were applied, it is not apparent how the LeSage site could survive the early screening criteria much less become the favored site. For example, the Fluor Daniel report states that in Phase II, which the Applicant states corresponds to Intermediate Phase I in the ER, the solicitation to communities seeking the nomination of potential sites indicated that sites should not have operating oil and gas wells or separate mineral rights. (I-RB-63 at 16.) The ER recites the same solicitation criterion and states that Intermediate Phase I sites were screened using a criterion to "[a]void property with operating gas/oil wells." (App. Exh. 1(h), at 7.1-6.) The Executive Summary of the Fluor Daniel report, however, states: "The LeSage site has a number of characteristics which appear to best satisfy the need for a site for CEEP. These can be summarized as follows[:] Environmental. Current land use includes oil and gas wells, timber farming and a county road." (I-RB-63 at ES-4.) Thus, it appears that the Fluor Daniel siting consultants believed throughout the siting process that there was an operating oil and gas well on the LeSage site. This fact seemingly should have disqualified the LeSage site even though it would not have disqualified the Homer community if other nominated sites in Claiborne Parish still met the other criteria. Indeed, nominated sites in other communities such as the Vivian Texaco site (I-RB-65 at 2) were disqualified for having an oil well on the nominated site. Yet the early screening criteria never disqualified the LeSage site. Although the Applicant's SAR indicates that LeSage well #4 is in fact outside the final southern site boundary (App. Exh. 1(a), at 2.1-13 to -14), that fact does not alter the apparent belief of the siting consultant during the siting process that the LeSage site contained oil and gas wells.

Similarly, the Fluor Daniel report indicates that during the First Stage of Phase IV, which the Applicant states corresponds to Intermediate Phase II in the ER, a "must" access control criterion was applied. That criterion stated that the site must be situated and arranged so that access by unauthorized persons could be prevented and

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moot. Should the opposite prove to be the case, however, these issues will have been decided so that any appropriate Staff licensing action can proceed.

The Intervenor's contention J.9, much like the similar component of Executive Order 12898, is concerned with the disparate impacts of the proposed CEC facility on the minority communities of Forest Grove and Center Springs. More particularly, the Intervenor's contention asserts that the Applicant's ER and the Staff's FEIS do not adequately describe and weigh the various environmental, social, and economic impacts of placing the CEC in the midst of Forest Grove and Center Springs. Similarly, as applicable here, the President's Executive Order instructs the agency, to the greatest extent practicable and permitted by law, to make environmental justice part of its mission by identifying and addressing disproportionately high and adverse human health and environmental effects on minority and low income populations as part of its licensing activities.

In the FEIS, the Staff addressed the various impacts of the CEC in chapters 3 and 4. Additionally, in chapter 4, section 4.2.1.7.4, on environmental justice, it states that, in addition to considering environmental justice from the perspective of whether race played a part in the site selection process, the Staff also considered whether minority and economically disadvantaged populations will be disproportionately affected by the CEC. (Staff Exh. 2, at 4-34.) In this regard, the Staff concludes they will not. (*Id.* at 4-35.)

In making this determination, the Staff declares that, to the extent the CEC affects the environment, those living closest to the facility will be most affected, but that all aspects of facility operation will be required to comply with State and Federal environmental regulations. Specifically, the Staff asserts that all effluent releases from the CEC will be below established regulatory limits and doses are expected to be well within regulatory limits. Further, the Staff states that it has not identified any significant offsite adverse impacts that would occur as a result of facility construction and operation. The Staff thus concludes that because the impacts of the CEC will be relatively small and there will not be a disproportionate adverse impact on minority or low-income populations, operating the LES facility will not promote environmental injustice. (*Id.*)

In their prefiled direct testimony, the Staff witnesses, Ms. Horn and Dr. Zeitoun, stated that in evaluating whether there were disproportionately high

indicated a site crossed by a public hiking trail, for example, would be unacceptable. (I-RB-63 at 21.) By applying the reconnaissance level information that was used at this early screening stage, the existence of Parish Road 39 bisecting the LeSage site seemingly should have disqualified the site even though it would not have disqualified the Homer community if there were other nominated sites in the parish that met the criteria. Indeed, nominated sites in other communities such as the Delhi site III, Oak Grove Sheldon site, and Winnsboro Magee site (I-RB-65 at 1) were disqualified for having a road across the site. Yet this early screening criterion never disqualified the LeSage site. A similar situation involving the LeSage site is presented by the proximity to airport criterion applicable to the First Stage of Phase IV in the Fluor Daniel report. (I-RB-63 at 22; I-RB-65 at 1.) Neither of these criteria are included in any of the listings of criteria listed in the ER. Accordingly, these anomalies in the process should be analyzed as part of the Staff investigation.

and adverse impacts to minority and low-income populations from the CEC facility, the Staff considered the term “high and adverse” to mean a significant impact such as one above regulatory limits. The Staff also used the term disproportionate to mean greater. (Horn et al. at 22 fol. Tr. 904.) They further testified that the Staff recognized that to whatever degree the CEC affects the environment, those living closest will be the most impacted. Accordingly, concentrations of uranium in the air or water will be higher close to the facility than in Homer; construction noise will be louder close to the site; and traffic impacts will be greater near the site than in Homer or other parts of the parish. (*Id.* at 21.) The Staff witnesses concluded, however, that, “[a]though Forest Grove and Center Springs residents will receive greater impacts due to CEC operation[,] . . . these impacts are not considered by the Staff to be significant or above regulatory limits, and are therefore not considered to be high and adverse.” (*Id.* at 22.)

In its evidentiary presentation on contention J.9, the Intervenor challenged the adequacy of the Staff’s FEIS treatment of a number of CEC-related effects on the communities of Forest Grove and Center Springs. We must judge the adequacy of the Staff’s treatment of the various impacts in the FEIS by the rule of reason. *See, e.g., Maine Yankee Atomic Power Co.* (Maine Yankee Atomic Power Station), ALAB-161, 6 AEC 1003, 1011-12 (1973). That standard is not one of perfection; rather, it is a question of reasonableness. As the Appeal Board long ago recognized, “absolute perfection in a FES [Final Environmental Statement] being unattainable, it is enough that there is ‘a good faith effort . . . to describe the reasonably foreseeable environmental impact’ of a proposed action.” *Id.* at 1012 (citations omitted).

A. Worst Case Accident Analysis

First, the Intervenor asserts that the FEIS does not adequately consider the worst case accident risk to the neighboring communities of Forest Grove and Center Springs.²² In his prefiled direct testimony, Dr. Bullard asserted that the FEIS identifies the greatest hazard associated with the operation of the CEC as a UF₆ storage area fire. He also conceded that the FEIS sets out the predicted intake of uranium at various distances from the release point in the event of that accident and indicates these accident-related intakes are in excess of the NRC guidance criteria of 10 milligrams (mg). Dr. Bullard further claimed that, other than recognizing it would be released in an accident, the FEIS contains

²² Even though the Intervenor’s contention is aimed at the Applicant’s ER and is understood also to challenge the Staff’s later filed FEIS (*see supra* p. 373), the Intervenor’s evidence is directed exclusively to the adequacy of the FEIS. Accordingly, the focus of our findings is on the Staff’s FEIS, although such findings necessarily encompass the adequacy of the Applicant’s ER because of the Staff’s heavy reliance on the ER in writing the FEIS.

no information about the release of hydrogen fluoride, which combines with atmospheric moisture to form potentially dangerous hydrofluoric acid ("HF"), nor does it discuss the effects of uranium or HF releases on nearby populations, other than to state the bare conclusion that the potential consequences of such an accident are unacceptable. (Bullard at 23-24 fol. Tr. 853.)

Dr. Bullard declared that the asserted Staff failure to address adequately the consequences of a severe accident is based upon the Staff's conclusion that various mitigative measures will keep such an accident from occurring. According to Dr. Bullard, by relying on such mitigative measures the Staff has improperly analyzed the nature of the CEC facility. Instead, the Staff should have recognized that the CEC is a hazardous facility with a certain level of risk that cannot be eliminated by regulation and that licensees, for whatever reason, do not always comply with safety regulations intended to protect the public. He thus claims that there is a foreseeable risk of such an accident and that the minority communities close to the CEC bear that risk to a significantly higher degree than people living further away. Dr. Bullard states that this disproportionate accident risk for Forest Grove and Center Springs should have been analyzed and discussed in the FEIS. (*Id.* at 25-26.)

We agree that the catastrophic failure of a hot cylinder containing liquefied UF₆ presents the greatest offsite hazard associated with the CEC. From the record before us, it appears there are two worst case accident scenarios that can result in such a failure: an autoclave heater malfunction and a UF₆ storage yard fire. In the FEIS, the Staff states that an autoclave heater malfunction is prevented by redundant Class I control systems and, therefore, such an event is neither considered credible nor analyzed. (Staff Exh. 2, at 4-53, 4-62.) The Intervenor did not challenge the Staff's treatment of an autoclave malfunction accident.

The Staff also evaluated a UF₆ storage area fire as part of its accident analysis for the CEC. Specifically, it considered an accident involving a cylinder transporter vehicle collision in which the vehicle fuel tank ruptures and the spilled fuel is ignited engulfing the UF₆ cylinder in flames. Relying on an earlier study of the consequences of this accident scenario that it performed in connection with emergency response requirements for fuel cycle facilities, the Staff set out in the FEIS the quantities of uranyl fluoride and hydrogen fluoride escaping from a ruptured UF₆ cylinder. In a table in the FEIS, the Staff also reproduced from its earlier study the predicted uranium intakes at various distances from the release point under two release scenarios. (*Id.* at 4-62 to -63.) The FEIS then states:

Intakes in excess of the NUREG-1391 guidance criteria (NRC, 1991b) are predicted for considerable distances from the release point. Intakes of uranium below the 10 mg limit and exposure to HF below the 25 mg/m³ limit are not expected to cause adverse health

effects. Substantially higher intakes can cause serious injuries and fatalities. The potential consequences of this type of accident are unacceptable.

(*Id.* at 4-63.)

Because it concludes that the consequences of a storage yard fire are unacceptable, the Staff then states in the FEIS that measures to prevent this accident are being imposed by license condition to limit transporter fuel inventories to less than the quantity of fuel that could sustain a fire causing cylinder rupture. Further, although the FEIS does not expressly state that offsite HF concentrations from a storage yard fire would exceed NRC limits, the Staff witnesses testified that “[i]f a cylinder were to overheat and rupture, uranium and HF concentrations would exceed the criteria at offsite locations and result in some health impacts.” (Horn et al. at 20 fol. Tr. 904.) The Staff witnesses also testified that, because LES will have in place mitigative measures to prevent an accident as well as an NRC-approved emergency plan, “the Staff does not believe that the accident risk to local residents is significant.” (*Id.*)

Contrary to the Intervenor’s assertion, we conclude that the Staff’s treatment in the FEIS of the worst case storage yard fire accident is minimally adequate to inform the reader of the consequences and likelihood of such an accident — the two components of the overall risk. Recognizing that the standard for judging the sufficiency of the discussion of environmental impacts in the FEIS is one of reasonableness, we cannot find that the Staff’s discussion of environmental impacts is so deficient that it requires remediation. As Dr. Bullard conceded, the FEIS sets out, albeit in a table format, the representative predicted uranium intakes from a storage yard fire accident at various distances from the point of release of UF₆. In addition, it is also obvious from the FEIS table that uranium intakes in excess of the NRC limit of 10 mg are predicted in both hypothesized release scenarios at various distances from the point of release. Further, the FEIS states that intakes substantially above the NRC limit can cause serious injuries and death. Thus, contrary to Dr. Bullard’s assertion, the FEIS does more, although not a great deal more, than merely state the conclusion that the consequences of an accident are unacceptable.

There is no question that the information in the FEIS could be stated more clearly and meaningfully. Indeed, one of the purposes of the EIS is to serve as an environmental full disclosure statement to, among others, interested members of the public. *See, e.g., Minnesota PIRG v. Butz*, 541 F.2d 1292, 1299 (8th Cir. 1976), *cert. denied*, 430 U.S. 922 (1977). Nonetheless, the essential information regarding uranium intakes and health consequences of a worst case accident is provided. No doubt, the FEIS would be more informative if it outlined the various levels of uranium intakes that cause serious injury and those that cause death and if it correlated the distances set forth in the table of representative

predicted uranium intakes with the local populations around the CEC. The FEIS is not, however, inadequate for failing to include this information.

Further, as Dr. Bullard asserts, the FEIS does not expressly address the exposure of the surrounding population to HF releases from a storage yard fire. But the FEIS does imply that HF exposures, like uranium intakes, will exceed the agency guidance criterion of 25 mg/m³ and that such exposures can cause serious injuries and fatalities — a fact confirmed by the Staff witnesses at the hearing. Thus, in the circumstances, the FEIS is minimally adequate in this regard as well.

Finally, we do not find meritorious Dr. Bullard's claim that the Staff may not rely on accident prevention measures that lessen the probability of an accident as a basis for concluding the risk to surrounding populations from a worst case storage yard fire is not significant. Here, the Staff relies upon a license condition limiting the fuel quantities carried by cylinder transporters to ensure that a storage yard fire would be deprived of a sufficient fuel source for heating a UF₆ cylinder to the rupture point. (Staff Exh. 2, at 4-63 to -64.) Similarly, the Applicant's ER indicates that a combination of engineered safety features and administrative controls must fail to have a worst case storage yard fire. (App. Exh. 1(h), at 5.1-9.) The Intervenor's disagreement with the Staff's conclusion that the risk to surrounding populations from such an accident is not significant, is supported by nothing more than Dr. Bullard's bare assertion that licensees do not always follow safety regulations. This is hardly sufficient to establish that the Staff's deterministic analysis of the accident risk is flawed.²³ For these reasons, we find that the Staff's treatment of the worst case storage yard fire accident in the FEIS is adequate.

²³ The Intervenor's position that the FEIS is inadequate also is not advanced by Dr. Bullard's reliance on the Commission's finding in the final fuel cycle emergency preparedness rule that releases of uranium hexafluoride in a severe accident occur rapidly with little warning, thereby leaving close neighbors no time to evacuate or even to seek shelter. See 54 Fed. Reg. 14,051, 14,052 (1989). The speed with which UF₆ releases may occur in a worst case storage yard fire does not address the likelihood of the accident occurring when there are a number of preventative measures in place.

Additionally, we note that the rationale for the rule requiring certain fuel cycle facilities like the CEC to have emergency plans rested, in part, on the fact that "[a]ny system of engineered safeguards is considered to have some possibility of failure. No system could ever be perfect." 54 Fed. Reg. at 14,056. On its face, it might appear incongruous for the agency to decide, on the one hand, that the generic risk of failure of engineered safeguards is sufficiently significant to require the emergency preparedness rule but, on the other, that engineered safeguards, along with the LES emergency plan, make the risk of a CEC worst case storage yard fire accident insignificant. Nevertheless, it is important to recognize that the Staff's FEIS conclusion is based upon its deterministic analysis of several specific mitigative measures that reduce the likelihood and hence the risk of a worst case accident to a point where the risk is not considered significant. To be sure, the Staff's assessment of the accident risk is not based upon a quantitative probabilistic risk assessment. The Intervenor, however, has not shown any error in the Staff's assessment.

B. Impacts of Road Closing/Relocation

The Intervenor also asserts that the FEIS is deficient because it fails to address the impacts of closing Parish Road 39, which currently bisects the LeSage site and joins the communities of Forest Grove and Center Springs. (Bullard at 33 fol. Tr. 853.) See generally *supra* p. 370. Dr. Bullard testified that in the FEIS the Staff assumed that Forest Grove Road would be relocated after it is closed. He claimed, however, that it is by no means clear that the road will be relocated because any decision about the road rests not with LES, but with the Claiborne Parish Police Jury that must pay for any road relocation. Dr. Bullard testified that if the road is not relocated it would impose upon the residents of Center Springs and Forest Grove an additional 8- or 9-mile trip by way of Homer to go from one community to the other. (Bullard at 33 fol. Tr. 853.)

Additionally, Dr. Bullard asserted that even if Parish Road 39 is relocated around the site, the Staff incorrectly concluded in the FEIS that the impacts would be very small and not pose unacceptable risks to the local community. According to Dr. Bullard, it is apparent that the Staff did not even consult with any of the residents of Forest Grove and Center Springs before reaching its conclusion for if it had, the Staff would have found that Forest Grove Road is a vital and frequently used link between the two communities, with regular pedestrian traffic. (*Id.* at 33-34.)

For its part, the Staff does indeed state in the FEIS that Parish Road 39 will be relocated to pass to the west of the plant area and that the existing road will not be closed until the relocated road is fully constructed and open. (Staff Exh. 2, at 2-21; see *id.* Fig. 2.8 at 2-22.) Further, the FEIS indicates that the road relocation will add approximately 120 meters (0.075 mile) to the traveling distance between State Roads 2 and 9 and will add an additional 600 meters (0.38 mile) to the 1800 meter (1.1 mile) distance between the Forest Grove Church and the Center Springs Church, which are the approximate centers of the respective minority communities. The Staff also concludes in the FEIS that the impacts associated with the road relocation “are very small and would not impose unacceptable risks to the local community.” (*Id.* at 4-12 to -13.) Finally, in the chapter 4 section on environmental justice, the Staff states that “[t]he minority communities of Forest Grove and Center Springs would be inconvenienced by the Parish Road 39 relocation, increasing the driving time between the communities.” (*Id.* at 4-35.) The Staff then generally concludes that there will not be a disproportionate adverse impact on minority or low-income populations. (*Id.*)

In their prefiled direct testimony, the Staff witnesses added that the relocation of Parish Road 39 is expected to result in the largest disruption to the residents of Forest Grove and Center Springs and that it will certainly affect those living near the road to a greater extent than those living in other locations around the parish. (Horn et al. at 14, 21-22 fol. Tr. 904.) They also testified that LES

had stated in a letter to the agency that the road would not be closed until a new road was built. (*Id.* at 14.) Further, Ms. Horn, the Environmental Project Manager for the LES application, testified the Staff concluded that Parish Road 39 would be relocated because the Applicant's ER so stated and Claiborne Parish had passed a resolution (which she had not seen) indicating the road would be relocated. (Tr. 909-10.) Similarly, Dr. Zeitoun testified that a member of his staff confirmed by telephone with a parish police juror that a resolution had been passed, but admitted no inquiry was made whether funds had been allocated to relocate the road. (Tr. 910-11.) Ms. Horn did acknowledge that the Staff had not considered the impacts on the Forest Grove and Center Springs communities if Forest Grove Road was closed and not relocated. (Tr. 912.)

In their prefiled direct testimony, the Staff witnesses also stated the comments on the draft EIS suggest that much of social interaction between Forest Grove and Center Springs center on the community churches. They asserted that the relocation of Parish Road 39 should not affect those activities and residents who attend church services at either church will still be able to do so, although driving distances will be slightly increased. The Staff witness further indicated that the road relocation may require residents of the communities to adjust carpools. For these reasons, the Staff concluded the road relocation would cause an inconvenience, but it is not expected to have a significant impact. (Horn et al. at 14-15 fol. Tr. 904.)

The Applicant's Licensing Manager, Mr. LeRoy, also stated in his prefiled direct testimony that Parish Road 39 will not be closed. Rather, he stated the segment crossing the LeSage site will be relocated to the western edge of the property and the relocation should not cause hardship to anyone. (LeRoy at 12-13 fol. Tr. 840; App. Exh. 1(h), at 4.1-2). He testified it was not foreseeable that the police jury would not relocate the road because "[t]hey voted unanimously to relocate the road." (Tr. 925.)

Although neither the Applicant nor the Staff offered the parish police jury resolution in evidence, and the Staff witnesses apparently have not even seen it, that resolution is in the record as an attachment to the Intervenor's original contentions.²⁴ As adopted on November 9, 1989, by the Claiborne Parish Police Jury, that resolution hardly can be characterized as the "open and shut case" portrayed by the Applicant and Staff witnesses. It is only a resolution — not an ordinance or other binding legislative enactment with the force of law — and thus merely expresses the prevailing sentiment and opinion of the then police jury. Moreover, the significant "resolved clause" of the resolution uses the disjunctive "or" when it declares the jury agrees to "close or relocate" the road. Therefore, contrary to the apparent belief of the Applicant and Staff witnesses,

²⁴ See Citizens Against Nuclear Trash's Contentions on the Construction Permit/Operating License Application for the Claiborne Enrichment Center (Oct. 3, 1991) following Attach. 13.

the police jury has only expressed a sentiment either to close or to relocate the segment of Parish Road 39 that crosses the LeSage property, but not necessarily to do both. The record before us thus does not support Mr. LeRoy's optimism that the parish will relocate the road. Rather, when all of the record evidence is considered, including that which shows that the minority communities of Forest Grove and Center Springs now are underserved when it comes to receiving even basic parish services (Bullard at 18, 36 fol. Tr. 853; Tr. 870), we have no basis to accept Mr. LeRoy's assurance that the road will be relocated by the parish instead of just closed.

Moreover, the record is clear that the Staff did not analyze the impacts on the communities of Forest Grove and Center Springs of closing Parish Road 39. This substantial shortcoming in the FEIS was remedied at the hearing, however, when LES indicated, for the first time, that it would relocate the road, if necessary. Specifically, Mr. LeRoy, in response to a direct inquiry, testified that LES will relocate the road in the event the police jury fails to do it. (Tr. 925.) We take this as a concession by the Applicant that the impacts of closing the road are sufficiently detrimental to the communities of Forest Grove and Center Springs that those impacts must be addressed by road relocation. Mr. LeRoy's answer thus is a direct commitment that, if the parish does not relocate the road, LES will take all necessary steps, including paying for the road relocation itself, to ensure the segment of Parish Road 39 bisecting the LeSage site is relocated before the current road is closed. Accordingly, we direct that a license condition to that effect must accompany any construction permit and operating license authorization.

The Intervenor also challenged the adequacy of the Staff's treatment in the FEIS of the impact from relocating (as opposed to closing) Parish Road 39 on the communities of Forest Grove and Center Springs and the Staff's conclusion that those impacts were very small. In particular, Dr. Bullard asserted that the Staff did not consider at all that Forest Grove Road was a vital and regularly used pedestrian link between Forest Grove and Center Springs.

The Staff's FEIS treatment of the impacts of relocating Parish Road 39 does not discuss Forest Grove Road's status as a pedestrian link between Forest Grove and Center Springs and the impacts of relocation on those who must walk the distance between the communities on this road. In the FEIS, the Staff calculates how much additional gasoline it will take to drive between the communities when the road is relocated and the added travel time the road relocation will cause for various trips. (Staff Exh. 2, at 4-12.) Similarly, in its hearing testimony, Staff witnesses acknowledged the interaction between the Forest Grove and Center Springs communities but only noted that "[t]he driving distance will be slightly increased." (Horn et al. at 14-15 fol. Tr. 904.)

Dr. Bullard testified, however, that Forest Grove Road is a vital and frequently used link between the communities with regular pedestrian traffic. Neither

the Staff nor the Applicant presented any evidence disputing Dr. Bullard's testimony in this regard. Further, the Bureau of Census statistics introduced by the Intervenor show that the African American population of Claiborne Parish is one of the poorest in the country and that over 31% of black households in the parish have no motor vehicles. (I-RB-67, at 12.) See *supra* p. 371. Again this evidence is undisputed. It thus is obvious that a significant number of the residents of these communities have no motor vehicles and often must walk. Adding 0.38 mile to the distance between the Forest Grove and Center Springs communities may be a mere "inconvenience" to those who drive, as the Staff suggests. Yet, permanently adding that distance to the 1- or 2-mile walk between these communities for those who must regularly make the trip on foot may be more than a "very small" impact, especially if they are old, ill, or otherwise infirm. The Staff in the FEIS has not considered the impacts the relocation of Forest Grove Road will have upon those residents who must walk. Accordingly, we find that the Staff's treatment in the FEIS of the impacts on the communities of Forest Grove and Center Springs from the relocation of Parish Road 39 is inadequate and must be revised.

In doing so, the Staff should identify any impacts of the relocation on local pedestrian traffic and factor those impacts into its weighing of the costs and benefits for the facility and in its environmental justice determination. Further, consideration must be given to whether actions can be taken to mitigate the impacts. In this regard, as we emphasized in LBP-96-25, 44 NRC at 370, it must be remembered that "NEPA is a procedural environmental full disclosure law and it does not dictate any particular substantive outcome as a result of the cost-benefit analysis."

C. Property Value Impacts

In line with that portion of contention J.9 claiming that the CEC will have negative economic impacts on the minority communities of Forest Grove and Center Springs, the Intervenor asserts that property values in the neighboring communities will be adversely affected by the facility and that this economic effect will be borne disproportionately by the minority communities that can least afford it. (Bullard at 22 fol. Tr. 853.) In his prefiled direct testimony, Dr. Bullard acknowledged that the Staff in the FEIS found that some property values may be negatively impacted by the proposed plant, but criticized the Staff for failing to identify the location, extent, or significance of this effect. Instead, Dr. Bullard claims the Staff merely concluded that there will be some unspecified positive and negative changes in property values from the CEC. (*Id.* at 35.)

In support of his assertion that the Staff analysis is inadequate, Dr. Bullard stated that his research shows that negative impacts on property values will occur in the immediate area of the plant and that, because of the housing barriers faced

by African Americans, the residents of Forest Grove and Center Springs will not have the same opportunities to relocate as do whites living in the parish. He asserted that the general beneficial effects on local housing values from the plant cited in the FEIS will have little, if any, effect on the minority communities of Forest Grove and Center Springs. In this regard, Dr. Bullard testified that the general "benefit streams" to counties with large industrial taxpayers do not have significant positive effects on low-income minority communities, which are already receiving a disproportionately low share of the services offered by the county. Further, he stated that the increased demand for property and housing attributable to the facility from migrants coming into the area is unlikely to affect the minority communities of Forest Grove and Center Springs very much, if at all. Dr. Bullard explained that, at the period of peak employment when the proposed facility is expected to have its greatest effect on the local population, which is during the fourth year of construction when some operation already has started, the FEIS states migrants will amount to only 12% of the work force, or 65 workers. He further observed that the FEIS indicates these workers will all be at the very upper end of the skill and pay scale and are expected to be predominantly white. Therefore, according to Dr. Bullard, these workers are extremely unlikely to seek housing in the poor, isolated African American communities of Forest Grove and Center Springs that already receive a relatively low level of services from the parish. (*Id.* at 35-37.)

The Intervenor's expert thus concludes that, although the FEIS acknowledges the proposed facility will depress some property values and increase others, the Staff has failed to address the central fact that in all likelihood the negative impacts of depressed property values will disproportionately affect the minority communities next to the plant. Similarly, he asserts the FEIS fails to address the fact that the minority residents of Forest Grove and Center Springs are among the poorest residents of the parish and are less likely to be able to absorb the diminution in property values than other wealthier, more mobile residents of Claiborne Parish. Dr. Bullard states that the FEIS should have analyzed and discussed these adverse, inequitable impacts. (*Id.* at 37.)

In FEIS section 4.2.1.7 entitled "Socioeconomic and Community Support Services," the Staff "describe[d] the social, economic, and community impacts of CEC operations." (Staff Exh. 2, at 4-31.) It stated that "[t]he towns of Homer and Haynesville have been emphasized due to their proximity to the proposed facility location and their status as providers of community services." (*Id.*) In subsection 4.2.1.7.1, the Staff stated with respect to housing that

For the last 2 years there has been an oversupply of lower quality and older homes on the market. However, there are very few homes, apartments, or mobile homes available for rent. Construction and operation of CEC would be expected to bid up rental prices and, to a lesser extent, home purchase prices; and will probably stimulate new construction. Any

shift of this nature is expected to be minimal since there is an oversupply of homes for sale and people can choose residences over a wide area.

(*Id.* at 4-32.) In subsection 4.5.2 on property values in its cost-benefit analysis, the Staff then stated:

LES is likely to have a significant effect on local housing values and, ultimately, amenities. There is considerable evidence to suggest that property values and amenities are enhanced in counties with large industrial taxpayers (e.g., fossil power plants) (Gamble and Downing, 1982). These benefits are not only via the direct payment to the taxing jurisdiction, but through the increased value of real property as the benefit stream to the property owners is capitalized into property values

The facility is likely to increase both housing and land prices because of increased demand (e.g., from migrants) and because of the benefit-capture effect just described. This is a benefit to all existing property owners, including those acquiring property prior to the actual receipt of the tax revenues. The magnitude of the benefit is difficult to quantify but is not negligible. Real estate prices in the area are likely to be bid up in anticipation of the property tax stream.

(*Id.* at 4-83.) Thereafter, in the summary of the cost-benefit analysis, the Staff notes that there will be “changes in property values (some positive, some negative).” (*Id.* at 4-86.)

In its prefiled direct testimony, the Staff witnesses stated that impacts such as property values “would be distributed throughout the region and are not expected to disproportionately or adversely impact Forest Grove or Center Springs.” (Horn et al. at 20 fol. Tr. 904.) Further, they asserted that “[i]mpacts on individuals cannot be predicted” and that “[a]ll of these types of impacts and benefits will occur throughout the region; however, there is no way to determine if a specific individual or area will benefit or be adversely impacted.” (*Id.*) Ms. Horn and Dr. Zeitoun also stated that the Staff did not consider the racial makeup of the homes surrounding the site when it assessed the impacts of the CEC. (*Id.* at 21.)

For its part, the Applicant stated in its ER that LES anticipates that real estate values of some adjacent properties may be enhanced due to the facility. It indicated that neither the specific adjacent properties nor the precise increase in value can be predicted but that the “[p]roperty value enhancement would be gained primarily through the location of business ventures supporting LES operations (e.g., food service, equipment vendors).” (App. Exh. 1(h), at 8.1-4 to -5.) Further, the Applicant’s Licensing Manager, Mr. LeRoy, testified that, in his experience with Duke Power Company nuclear power plants, property values around the plants dramatically increased after the facilities were constructed. (Tr. 919, 954.) He indicated that he was referring to the Oconee Nuclear Station on Lake Keowee and the Catawba Nuclear Station on Lake Wylie in South Carolina,

and the McGuire Nuclear Station on Lake Norman in North Carolina. (Tr. 956.) Mr. LeRoy then provided one example of residential or vacation property on each of the lakes before and after the nuclear facilities were built showing substantial increases in values from the 1970s and early 1980s through the 1990s. (Tr. 957-59.) He conceded, however, that he did not know whether any of the communities around the three lakes were African American communities. (Tr. 961.)

Additionally, Mr. Dorsey testified that in his 25 to 30 years of experience on a number of significant projects in a wide range of industries, property values have increased in the immediate vicinity of the final site. (Tr. 919.) Likewise, Mr. Schaperkotter added that in his experience the presence of new development quite often creates an increase in property values. (*Id.*)

The Staff's treatment of the economic impacts of the CEC on property values in the FEIS does indeed recognize that the CEC will depress some property values while increasing others, but the Staff fails to identify the location, extent, or significance of impacts. Further, although, the FEIS generally indicates the CEC is likely to increase both housing and land prices because of increased demand and the benefits capture effect, the Staff makes no attempt to allocate the costs or benefits. Dr. Bullard directly challenges the Staff's failure to assess the impacts of the CEC on property values in the communities of Forest Grove and Center Springs asserting that when facilities like the CEC are placed in the midst of poor, minority communities, the facility has negative impacts on property values in the immediate area of the plant. For the reasons specified below, we find his testimony on the negative economic impact of the CEC on property values in these minority communities reasonable and persuasive.

The focus of Intervenor contention J.9 and Dr. Bullard's supporting testimony is that the negative economic impact of the CEC must be assessed as it operates on the minority "communities" of Forest Grove and Center Springs, not just on a particular parcel of property. Dr. Bullard explained that unlike white residents of the parish, the black residents of Forest Grove and Center Springs face substantial "housing barriers" that preclude them from leaving when a large industrial facility is sited in the midst of their residential area. As a consequence, these already economically depressed communities must fully absorb the further adverse impact of having a heavy industrial facility nearby making them even more undesirable. He testified that the beneficial effects on housing values from increased demand by new migrating employees and the benefit capture effect relied upon by the Staff in the FEIS will have no effect on these minority communities that currently receive almost no parish services, are virtually 100% African American, and are inhabited by some of the most economically disadvantaged people in the United States. As Dr. Bullard stated, it is "extremely unlikely" new workers to the area will seek to live in Forest Grove and Center Springs. Dr. Bullard concludes that these factors lead to an

overall negative impact on property values in the minority communities that must host the CEC; yet these communities are made up of people who can least afford the diminution in property values.

The Staff witnesses made no attempt to explain how or why Dr. Bullard might be mistaken. Rather, they testified that the impacts on property values from the CEC would be distributed throughout the region and, therefore, the impacts “are not expected to disproportionately or adversely impact Forest Grove or Center Springs.” (Horn et al. at 20 fol. Tr. 904.) Further, they claimed “there is no way to determine if a specific individual or area will benefit or be adversely impacted.” (*Id.*) We find that the testimony of these Staff witnesses in this regard is neither persuasive nor reasonable in this instance. Indeed, given the Staff’s recognition in the FEIS that there will be some negative impacts on property values from the CEC, it is difficult to envision an economic rationale that would demonstrate those adverse impacts from the CEC are likely to occur to properties well removed from the facility, such as in Homer or Haynesville, as opposed to the Forest Grove and Center Springs areas next to the facility.

We also find the Intervenor’s position persuasive because we find this witness both credible and convincing. Dr. Bullard is a recognized expert on the subject of environmental justice who for years has conducted research, lectured, and written extensively in the areas of housing and community development. He has presented a reasoned, persuasive, and unchallenged explanation why the CEC will negatively impact property values in these minority communities. Additionally, even a cursory look at the references cited by Dr. Bullard in his prefiled direct testimony show there has been substantial research indicating the negative impacts on minority communities in analogous circumstances.

In reaching this conclusion, we recognize that the Staff witnesses stated it was not “expected” the impacts from the CEC on property values would disproportionately or adversely impact Forest Grove or Center Springs. Yet the same witnesses also specifically testified that the Staff did not consider the racial makeup of the homes surrounding the site when they considered the impacts from the CEC. Thus, the Staff apparently has not considered the economic impact on property values of siting the CEC in the midst of these neighboring minority communities, qua minority communities. Indeed, the exploration of this matter would likely be another circumstance that merits scrutiny under Executive Order 12898.

Nor is the Applicant’s evidence about property value increases persuasive here. Applicant’s ER undoubtedly is correct in predicting that a number of adjacent properties will increase in value as sites for food service and equipment vendors supporting the plant. But the number of immediately adjacent properties involved will be relatively few, most likely on State Road 9. The thrust of contention J.9 and Dr. Bullard’s testimony is the impact on the minority communities of Forest Grove and Center Springs as a whole, rather than on two

or three individual parcels of property. The Applicant's ER simply does not address that impact.

By the same token, the opinions of Mr. Dorsey and Mr. Schaperkotter to the effect that industrial facilities often increase property values in the vicinity of a facility are far too general to draw any reasonable conclusions about the impacts on property values in the circumstances presented here. Likewise, Mr. LeRoy's testimony about the positive impact on lakefront vacation home values from the construction of nuclear power plants is neither useful nor reasonable in making a comparison with the economically disadvantaged minority communities of Forest Grove and Center Springs. Certainly, the reality of Forest Grove and Center Springs hardly seems comparable to the description of Lake Wylie in Applicant's Exhibit 19, which states that "[t]he Catawba plant was built on a beautiful lake, dotted with hundreds of expensive homes and homesites." (App. Exh. 19 at 7.) Nor do these communities resemble the description of Lake Keowee in Exhibit 19 as "[o]ne of the most prestigious resort/retirement communities in the United States [which] is less than a mile from Oconee Nuclear Station. At Keowee Key more than 1500 people golf, boat, fish, relax and retire next door to a nuclear plant." (*Id.* at 8.)

On this basis, we find that the Staff's treatment in the FEIS of the impacts from the CEC on property values in the communities of Forest Grove and Center Springs is inadequate. Therefore, the Staff must consider these impacts and factor them into its weighing of the costs and benefits of the facility and in its environmental justice determination.

D. Other Impacts

Finally, the Intervenor also challenges the adequacy of the Staff's treatment in the FEIS of the impacts from the CEC on the communities of Forest Grove and Center Springs concerning a number of other matters, including (1) contamination of surface and groundwater; (2) impacts on groundwater supply; (3) impacts of noise; (4) impacts of traffic, development, and crime; and (5) impacts from the disproportionate distribution of benefits. We have carefully examined all of the evidence regarding each of these claims and find that the FEIS adequately considers the impacts. Further, we find that none of these impacts will cause a disproportionately high and adverse impact on the residents of Forest Grove and Center Springs. In addition to the foregoing findings on contention J.9, we have considered all of the other arguments, claims, and proposed findings of the parties on this contention and find that they either are without merit, immaterial, or unnecessary to this Final Initial Decision.

IV. CONCLUSION

For the reasons detailed in Part II.C, we conclude that a thorough Staff investigation of the CEC site selection process is essential to determine whether racial discrimination played a role in that process, thereby ensuring compliance with the nondiscrimination directive contained in Executive Order 12898. Additionally, for the reasons set forth in Part III.B, we conclude that the Staff's treatment in the FEIS of the impacts of relocating Parish Road 39 on the communities of Forest Grove and Center Springs is inadequate and the Staff must take steps to revise the FEIS consistent with this Decision. Also in connection with the relocation of Parish Road 39, consistent with this Decision a license condition must be included in any ultimate construction permit—operating license authorization that makes the Applicant responsible for ensuring that the current road is relocated before the segment that currently bisects the facility site is closed. Further, we conclude in Part III.C that the Staff's treatment in the FEIS of the economic impact of the CEC on the properties in the communities of Forest Grove and Center Springs is inadequate and that the Staff must take steps to revise the FEIS consistent with this Decision.

In light of the Board's conclusions in the earlier Partial Initial Decisions in LBP-96-25, 44 NRC 331 (1996), and LBP-97-3, 45 NRC 99 (1997), the Staff also must take appropriate steps to address the other identified insufficiencies in the FEIS. Further, the Applicant's requested authorization for a combined construction permit and operating license is hereby *denied*, albeit without prejudice to the Applicant amending its license application in accordance with the Partial Initial Decisions in this proceeding.

Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this Final Initial Decision will constitute the final Decision of the Commission on this contention forty (40) days from the date of its issuance unless a petition for review is filed in accordance with 10 C.F.R. § 2.786, or the Commission directs otherwise. Within fifteen (15) days after service of this Final Initial Decision, any party may file a petition for review with the Commission on the grounds specified in 10 C.F.R. § 2.786(b)(4). The filing of a petition for review is mandatory in order for a party to have exhausted its administrative remedies before seeking judicial review at the appropriate time. Within ten (10) days after service of a petition for review, any party to the proceeding may file an

answer supporting or opposing Commission review. The petition for review and any answers shall conform to the requirements of 10 C.F.R. § 2.786(b)(2)-(3).
It is so ORDERED.

**THE ATOMIC SAFETY AND
LICENSING BOARD**

**Thomas S. Moore, Chairman
ADMINISTRATIVE JUDGE**

**Richard F. Cole
ADMINISTRATIVE JUDGE**

**Frederick J. Shon
ADMINISTRATIVE JUDGE**

**May 1, 1997
Rockville, Maryland**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

G. Paul Bollwerk, III, Presiding Officer
Dr. Charles N. Kelber, Special Assistant

In the Matter of

Docket No. 40-3453-MLA
(ASLBP No. 97-723-02-MLA)

ATLAS CORPORATION
(Moab, Utah Facility)

May 16, 1997

In this 10 C.F.R. Part 2, Subpart L informal proceeding concerning pro se petitioner John Francis Darke's challenge to a request by Atlas Corporation to amend the license for its Moab, Utah uranium milling facility to extend the completion date for placing a final radon barrier on the facility tailings pile, the Presiding Officer rules (1) Petitioner Darke's hearing request is timely and specifies areas of concern that are germane to the subject matter of the proceeding; (2) Petitioner Darke has failed to establish any grounds for using 10 C.F.R. Part 2, Subpart G formal adjudicatory procedures; and (3) despite multiple opportunities to address the issue, Petitioner Darke has failed to meet his burden to establish his standing to intervene in this proceeding.

**RULES OF PRACTICE: INFORMAL HEARINGS (PARTY
ADMISSION REQUIREMENTS)**

To be admitted as a party to an informal adjudication under Subpart L of 10 C.F.R. Part 2 regarding a licensee-initiated materials license amendment, the individual or organization filing a hearing/intervention request must establish three things: (1) the petitioner is a "person whose interest may be affected by the proceeding" within the meaning of section 189a(1)(A) of the Atomic Energy Act of 1954 (AEA), 42 U.S.C. § 2239(a)(1)(A), in that the petitioner has standing

to participate in the proceeding consistent with the standards governing standing in judicial proceedings generally; (2) the petitioner has “areas of concern” regarding the requested licensing action that are germane to the subject matter of the amendment proceeding; and (3) the hearing/intervention petition was timely filed. *See* 10 C.F.R. § 2.1205(e), (h).

RULES OF PRACTICE: INFORMAL HEARINGS (USING OTHER PROCEDURES)

In an informal adjudication under 10 C.F.R. Part 2, Subpart L, the petitioner may request that the proceeding be conducted employing procedures other than those set forth in Subpart L, which could include use of the procedures for formal, trial-type adjudications set forth in Subpart G of Part 2. *See id.* § 2.1209(k).

RULES OF PRACTICE: INFORMAL HEARINGS (SPECIFYING AREAS OF CONCERN)

The “areas of concern” specified in support of a hearing request under Subpart L “need not be extensive, but [they] must be sufficient to establish that the issues the requester wants to raise fall generally within the range of matters that properly are subject to challenge in such a proceeding.” 54 Fed. Reg. 8269, 8272 (1989). Like the requirement that a 10 C.F.R. Part 2, Subpart G formal hearing petition must define the “specific aspect or aspects of the subject matter of the proceeding as to which petitioner wishes to intervene,” 10 C.F.R. § 2.714(a)(2), the Subpart L direction to define “areas of concern” is only intended to ensure that the matters the petitioner wishes to discuss in his or her written presentation are generally within the scope of the proceeding.

RULES OF PRACTICE: INFORMAL HEARINGS (USING OTHER PROCEDURES)

A request to use other procedures in a 10 C.F.R. Part 2, Subpart L proceeding should involve consideration of whether, given the particular circumstances involved in the proceeding, permitting the use of additional, trial-type procedures such as oral cross-examination would add appreciably to the factfinding process. *See Sequoyah Fuels Corp.* (Sequoyah UF₆ to UF₄ Facility), CLI-86-17, 24 NRC 489, 497 (1986).

RULES OF PRACTICE: HEARING REQUIREMENT (MATERIALS LICENSE)

As a request for a revision to a 10 C.F.R. Part 40 source materials license, a licensee's amendment application falls squarely within the designation of a "licensee-initiated amendment" under 10 C.F.R. § 2.1201(a)(1) — as opposed to being a 10 C.F.R. Part 2, Subpart B Staff-imposed amendment that would be subject to the formal hearing procedures in Subpart G — and thus properly is the subject of Subpart L informal procedures.

ATOMIC ENERGY ACT: STANDING TO INTERVENE

RULES OF PRACTICE: STANDING TO INTERVENE

To establish standing to participate as of right in an adjudicatory proceeding regarding an agency licensing action, an individual petitioner must demonstrate that (1) he or she has suffered or will suffer a distinct and palpable "injury in fact" within the "zone of interests" arguably protected by the statutes governing the proceeding (e.g., the AEA, the National Environmental Policy Act of 1969); (2) the injury is fairly traceable to the challenged action; and (3) the injury is likely to be redressed by a favorable decision. *See Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-1, 43 NRC 1, 6 (1996).

**RULES OF PRACTICE: STANDING TO INTERVENE
(CONSTRUCTION OF PETITION)**

Although the petitioner bears the burden of establishing his or her standing, it also is clear under Commission caselaw that in making a standing determination a presiding officer is to "construe the petition in favor of the petitioner." *Georgia Institute of Technology* (Georgia Tech Research Reactor, Atlanta, Georgia), CLI-95-12, 42 NRC 111, 115 (1995).

**ATOMIC ENERGY ACT: STANDING TO INTERVENE
(INJURY IN FACT)**

**RULES OF PRACTICE: STANDING TO INTERVENE
(INJURY IN FACT)**

A licensee's claim that "regulatory limits" are not being exceeded by offsite radiological releases from a facility is not, standing alone, sufficient to show that a petitioner lacks standing. As was noted in the face of a similar assertion, "[r]elative to a threshold standing determination, . . . even minor radiological exposures resulting from a proposed licensee activity can be enough to create

the requisite injury in fact.” *General Public Utilities Nuclear Corp.* (Oyster Creek Nuclear Generating Station), LBP-96-23, 44 NRC 143, 158 (1996).

**ATOMIC ENERGY ACT: STANDING TO INTERVENE
(INJURY IN FACT)**

**RULES OF PRACTICE: STANDING TO INTERVENE
(INJURY IN FACT)**

A showing that there may be some offsite radiological impacts to someone is not enough to establish standing for a particular petitioner. As the Commission has made clear on a number of occasions, in the context of a proceedings other than those for the grant of a reactor construction permit or operating license, a petitioner who wants to establish “injury in fact” for standing purposes must make some specific showing outlining how the particular radiological (or other cognizable) impacts from the nuclear facility or materials involved in the licensing action at issue can reasonably be assumed to accrue to the petitioner. *See Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 246-48 (1996).

**ATOMIC ENERGY ACT: STANDING TO INTERVENE
(INJURY IN FACT)**

**RULES OF PRACTICE: STANDING TO INTERVENE
(INJURY IN FACT)**

In proceedings other than those for the grant of a reactor construction permit or operating license, petitioners generally establish their “injury in fact” by quantifying the distance from the nuclear facility or materials at which they reside or engage in other activities they believe are likely to result in radiological impacts. *See, e.g., Oyster Creek*, LBP-96-23, 44 NRC at 157-59.

**ATOMIC ENERGY ACT: STANDING TO INTERVENE
(INJURY IN FACT)**

**RULES OF PRACTICE: STANDING TO INTERVENE
(INJURY IN FACT)**

A petitioner has not shown any reasonable nexus between himself or herself and any purported radiological impacts when, despite assertions about potential facility-related airborne and waterborne radiological contacts, he or she has not delineated these with enough concreteness to establish some impact on him that is sufficient to provide him or her with standing. By not providing any

information that indicates whether water-related activities are being conducted upstream or downstream from a facility and by describing other activities only using vague terms such as “near,” “close proximity,” or “in the vicinity” of the facility at issue, the petitioner fails to carry his or her burden of establishing the requisite “injury in fact.”

RULES OF PRACTICE: STANDING TO INTERVENE (FACTUAL REPRESENTATIONS)

It generally is the practice for participants making factual claims regarding the circumstances that establish standing to do so in affidavit form that is notarized or includes a declaration that the statements are true and are made under penalty of perjury.

MEMORANDUM AND ORDER (Denying Hearing Request)

Pro se petitioner John Francis Darke has filed a hearing request challenging Atlas Corporation’s (Atlas) December 20, 1996 application to amend its 10 C.F.R. Part 40 license for its uranium milling facility in Moab, Utah. The amendment in question would modify License Condition (LC) 55 A.(3) of the Atlas license (No. SUA-917) to extend by 4 years — until December 31, 2000 — the completion date for placing a final radon barrier on the existing mill tailings pile at the Moab facility. Licensee Atlas opposes Petitioner Darke’s hearing request asserting, among other things, that he lacks standing and has failed to specify any litigable issues.

For the reasons stated below, I find Petitioner Darke has not established his standing to intervene in this proceeding. Accordingly, I deny his hearing request.

I. BACKGROUND

A. Atlas Reclamation Plans for the Moab Facility

Atlas’ Moab uranium milling facility, which is located on the west bank of the Colorado River approximately 3 miles northwest of Moab, Utah, ceased commercial operation in 1984. At present, on site at the facility is a 10.5-million-ton mill tailings pile that needs to be reclaimed (i.e., stabilized) for long-term disposal. This pile, which currently occupies approximately 130 acres of land and rises to a height of some 90 feet, is located within 750 feet of the Colorado River. *See* Office of Nuclear Materials Safety and Safeguards (NMSS), U.S.

Nuclear Regulatory Commission (NRC), NUREG-1531, Draft Environmental Impact Statement [(EIS)] Related to Reclamation of the Uranium Mill Tailings at the Atlas Site, Moab, Utah (Jan. 1996) at 1-4, 2-1.

To comply with agency requirements regarding site stabilization, Atlas initially submitted an onsite reclamation plan in 1981, which the NRC Staff approved the following year. Then, in 1988 Atlas submitted a license amendment application that included a revised onsite reclamation plan. Staff review of that plan resulted in requests for additional information and redesign. Thereafter, in June 1992 Atlas submitted another revised onsite reclamation plan. In July 1993, the Staff issued a notice of its intent to approve this Atlas reclamation plan and made available for public comment an environmental assessment regarding the proposed Atlas plan. *See* NMSS, NRC, NUREG-1532, Draft Technical Evaluation Report [(TER)] for the Revised Reclamation Plan for the Atlas Corporation Moab Mill (Jan. 1996) at 1-4.

Based on public comment, in October 1993 the Staff withdrew the July 1993 notice of intent, and in March 1994 issued another notice declaring its intent to prepare a full-blown EIS. The Staff also began a reevaluation of the entire revised Atlas reclamation plan. *See id.* As part of this reevaluation process, in March 1994 the Staff also issued a notice that included an opportunity for a hearing on the revised Atlas reclamation plan. *See* 67 Fed. Reg. 16,665, 16,665 (1994). No hearing requests apparently were filed in response to this notice, however.

The Staff finally issued a draft EIS and a draft TER on Atlas' proposed onsite reclamation plan in January 1996. A final TER regarding the plan was issued in March 1997, while a final EIS apparently is not expected until the fall of 1997. *See* Licensee's Response (Apr. 7, 1997) at 2 & n.2 [hereinafter Atlas Response].

B. Atlas Request to Extend Radon Barrier Completion Date

Related to the approval of a reclamation plan for the Atlas facility is the item of central interest in this proceeding: the December 31, 1996 target date initially set for the placement of a final earthen cover on the Moab facility tailings to limit radon emissions to a flux of no more than 20 picocuries per meter squared per second (pCi/m²/s). This date came into play by reason of an October 1991 memorandum of understanding between the Environmental Protection Agency and the NRC that set out target dates for final radon barrier emplacement for a number of tailings impoundments, including the Atlas Moab facility. *See* 56 Fed. Reg. 55,434, 55,435 (1991). Subsequently, the December 31, 1996 date for final radon barrier emplacement at the Moab facility was incorporated into the Atlas license as LC 55 A.(3) by Amendment No. 17 issued on November 4, 1992.

Under LC 55 C., which also was adopted under Amendment No. 17, any request to revise the final radon barrier completion date specified in the license “must demonstrate that compliance was not technologically feasible (including inclement weather, litigation which compels delay to reclamation, or other factors beyond the control of the licensee).” *See* Letter from Sherwin E. Turk, NRC Staff Counsel, to Presiding Officer and Special Assistant (Feb. 14, 1997), encl. 1, at 11 (License No. SUA-917, Amendment No. 27) [hereinafter Turk Letter]. Relying on this provision, *see* Atlas Response at 8-9, on December 20, 1996, Atlas asked to amend the Moab facility license to extend by 4 years the December 31, 1996 date specified in LC 55 A.(3) for final radon barrier completion. As the basis for this request, Atlas declared that (1) the December 1996 deadline was footed on the assumption the Moab facility reclamation plan would be approved in 1993, thereby allowing 3 years to perform construction work and still provide an adequate period for consolidation of affected materials placed in the impoundment before placement of the final radon barrier; and (2) because the agency EIS and TER were not completed, Atlas did not have the plan approval needed to begin construction. *See* Turk Letter, encl. 2, at 1-2 (Letter from Richard E. Blubaugh, Atlas Corp., to Joseph J. Holonich, NMSS, NRC (Dec. 20, 1996)).

C. Adjudicatory Proceeding Procedural Posture

On January 14, 1997, the Staff issued a notice stating it had received the December 20 Atlas license amendment application and was offering an opportunity for a 10 C.F.R. Part 2, Subpart L informal hearing on the Licensee’s request. *See* 62 Fed. Reg. 3313, 3313 (1997). In a one-page letter dated January 30, 1997, Petitioner Darke asked for a hearing regarding the Atlas amendment request. *See* Letter from John Francis Darke to Secretary, NRC (Jan. 30, 1997) [hereinafter Darke Hearing Request]. Besides asserting the requested licensing action “is without factual or legal basis,” Petitioner Darke sought to have the matter heard under the rules for formal adjudicatory proceedings set forth in Subpart G of 10 C.F.R. Part 2. *Id.* Further, addressing his standing to become a party to such a proceeding, he stated only that the proposed amendment was “predominately adverse to the health and safety of the requestor and his family, who reside in the vicinity of the subject site.” *Id.*

After being designated as presiding officer for this proceeding, *see* 62 Fed. Reg. 7279 (1997), on February 12, 1997, I issued an initial order. That order established a deadline for the Staff to specify whether it wished to be a party to this proceeding. It also provided Petitioner Darke with an opportunity to supplement his hearing petition to address more fully the issue of his standing and to explain in more detail his areas of concern regarding the Atlas amendment request and his reasons for claiming that a formal adjudication under Subpart G

was appropriate. *See* Presiding Officer Memorandum and Order (Initial Order) (Feb. 12, 1997) at 2-3 [hereinafter Initial Order].

In a February 21, 1997 response to this order, the Staff declared that, in accordance with 10 C.F.R. § 2.1213, it would not participate as a party in this proceeding. *See* Letter from Sherwin E. Turk, NRC Staff Counsel, to Presiding Officer and Special Assistant (Feb. 21, 1997). Petitioner Darke responded to the initial order with two substantive filings.¹ In the first, submitted on February 24, 1997, he addressed the question of why this proceeding should be conducted under Subpart G formal procedures. *See* [First Response to Presiding Officer's Memorandum and Order Dated February 13, 1997] (Feb. 24, 1997) [hereinafter Darke February 24 Response]. In his second filing, dated March 3, 1997, Petitioner Darke discussed his areas of concern regarding the proposed amendment and the basis for his standing to intervene in this proceeding. *See* [Second Response to Presiding Officer's Memorandum and Order Dated February 13, 1997] (Mar. 3, 1997) [hereinafter Darke March 3 Response].

On March 5, 1997, the Staff submitted a letter declaring that, in accordance with 10 C.F.R. § 2.1205(m), the previous day it had issued the license amendment sought by Atlas, thereby revising LC 55 A.(3) to change the date for final radon barrier placement at the Moab facility to December 31, 2000. *See* Letter from Sherwin E. Turk, NRC Staff Counsel, to Presiding Officer and Special Assistant (Mar. 5, 1997). Although a petitioner may contest a Staff determination to issue a license amendment during the pendency of a hearing, *see* 10 C.F.R. § 2.1263, Petitioner Darke did not initiate such a challenge.

Thereafter, in a March 11, 1997 memorandum and order, I afforded Petitioner Darke an opportunity to make an additional submission addressing the issue of standing. *See* Presiding Officer Memorandum and Order (Permitting Additional Filing) (Mar. 11, 1997) at 2-3 [hereinafter Additional Filing Order]. He filed that pleading on March 24, 1997. *See* [Response to Presiding Officer's March 11, 1997 Memorandum and Order] (Mar. 24, 1997) [hereinafter Darke March 24 Response]. Atlas then submitted its response to all of Petitioner Darke's prior filings, asserting he lacked standing and had failed to specify areas of concern germane to the proceeding or to establish an adequate basis for his request that formal adjudicatory procedures be used. *See* Atlas Response at 4-11. In lieu of a prehearing conference/oral argument on these issues, I permitted Petitioner Darke to file a reply to this Atlas response. *See* Presiding Officer Order (Permitting Reply Filing) (Apr. 11, 1997) at 2 [hereinafter Reply Filing Order]. Petitioner Darke did so on April 21, 1997. *See* [Response to Presiding

¹ In addition, Petitioner Darke filed a third pleading in which he provided corrections to the first two pleadings. *See* [Third Response to Presiding Officer's Memorandum and Order Dated February 13, 1997] (Mar. 13, 1997).

Officer's April 11, 1997 Memorandum and Order] (Apr. 21, 1997) [hereinafter Darke Reply].

II. ANALYSIS

Section 2.1205 of title 10 of the *Code of Federal Regulations* makes it clear that to be admitted as a party in an informal adjudication under Subpart L of Part 2 regarding a licensee-initiated materials license amendment, the individual or organization filing a hearing/intervention request must establish three things: (1) the petitioner is a "person whose interest may be affected by the proceeding" within the meaning of section 189a(1)(A) of the Atomic Energy Act of 1954 (AEA), 42 U.S.C. § 2239(a)(1)(A), in that the petitioner has standing to participate in the proceeding consistent with the standards governing standing in judicial proceedings generally; (2) the petitioner has "areas of concern" regarding the requested licensing action that are germane to the subject matter of the amendment proceeding; and (3) the hearing/intervention petition was timely filed. *See* 10 C.F.R. § 2.1205(e), (h). In addition, as Petitioner Darke's hearing request illustrates, the petitioner may request that any proceeding be conducted employing procedures other than those set forth in 10 C.F.R. Part 2, Subpart L, governing informal adjudications, which could include use of the procedures for formal, trial-type adjudications set forth in Subpart G of Part 2. *See id.* § 2.1209(k).

A. Timeliness, Areas of Concern, and Additional Adjudicatory Procedures

As he seeks to address these threshold matters, Petitioner Darke's various filings present a decidedly mixed bag. For instance, as he points out in his March 3 response, because he filed (i.e., mailed) his hearing request within 8 days of *Federal Register* publication of the Staff's notice of opportunity for hearing, Petitioner Darke's hearing request clearly is timely. *See* Darke March 3 Response at 5.

So too, his hearing request, as supplemented by his filings of March 3 and March 24, sets forth "areas of concern" that are sufficient to support the grant of his hearing request. As the Commission has indicated, the "areas of concern" specified in support of a hearing request under Subpart L "need not be extensive, but [they] must be sufficient to establish that the issues the requester wants to raise fall generally within the range of matters that properly are subject to challenge in such a proceeding." 54 Fed. Reg. 8269, 8272 (1989). Like the requirement that a Subpart G formal hearing petition must define the "specific aspect or aspects of the subject matter of the proceeding as to which petitioner

wishes to intervene,” 10 C.F.R. § 2.714(a)(2), the Subpart L direction to define “areas of concern” is only intended to ensure that the matters the petitioner wishes to discuss in his or her written presentation are generally within the scope of the proceeding. In this instance, Petitioner Darke has made it apparent that, among other things, he wishes to address the validity of the reasons cited by Licensee Atlas for requesting the amendment (i.e., whether completion under the prior schedule “was not technologically feasible” in accordance with LC 55 C. and 10 C.F.R. Part 40, App. A, Criterion 6A(1)) and the efficacy of the extended completion date, both of which are appropriate subjects for consideration relative to the license amendment in question. *See Darke March 3 Response at 5-8.*

On the other hand, Petitioner Darke’s request that Subpart G formal adjudicatory procedures be used for this proceeding is well off the mark. The Commission has indicated that such a request should involve consideration of whether, given the particular circumstances involved in the proceeding, permitting the use of additional, trial-type procedures such as oral cross-examination would add appreciably to the factfinding process. *See Sequoyah Fuels Corp. (Sequoyah UF₆ to UF₄ Facility), CLI-86-17, 24 NRC 489, 497 (1986).* Petitioner Darke has taken a different tack, asserting this proceeding should be held using Subpart G formal procedures because it does not involve the type of “licensee-initiated amendment” of a nuclear materials license to which Subpart L is applicable under 10 C.F.R. § 2.1201(a)(1). *See Darke February 24 Response at unnumbered 2-3.* There is not the slightest doubt, however, that as a request for a revision to its 10 C.F.R. Part 40 source materials license, the Atlas amendment application falls squarely within that designation — as opposed to being a 10 C.F.R. Part 2, Subpart B Staff-imposed amendment that would be subject to the formal hearing procedures in Subpart G — and thus properly is the subject of Subpart L informal procedures. Because Petitioner Darke has made no other showing in support of his request for the use of Subpart G formal procedures, I have no basis for recommending to the Commission that such procedures be used.

B. Standing to Intervene

My decision on Petitioner Darke’s request to convene a hearing thus comes down to the question whether he has made a showing sufficient to establish he has standing to intervene in this proceeding. To establish standing to participate as of right in an adjudicatory proceeding regarding an agency licensing action, an individual petitioner must demonstrate that (1) he or she has suffered or will suffer a distinct and palpable “injury in fact” within the “zone of interests” arguably protected by the statutes governing the proceeding (e.g., the AEA, the National Environmental Policy Act of 1969); (2) the injury is fairly traceable to the challenged action; and (3) the injury is likely to be redressed by a favorable decision. *See Yankee Atomic Electric Co. (Yankee Nuclear Power Station),*

CLI-96-1, 43 NRC 1, 6 (1996). Further, while the petitioner bears the burden of establishing his or her standing, it also is clear under Commission caselaw that in making a standing determination a presiding officer is to "construe the petition in favor of the petitioner." *Georgia Institute of Technology* (Georgia Tech Research Reactor, Atlanta, Georgia), CLI-95-12; 42 NRC 111, 115 (1995).

As was noted previously, in his initial hearing request Petitioner Darke's only statement regarding his standing to intervene was that the Atlas amendment request was "predominately adverse" to his health and safety and that of his family, "who reside in the vicinity of the subject site." Darke Hearing Request at 1. In an effort to learn more about his standing claim, in my February 12 initial order I gave Petitioner Darke an opportunity to supplement his hearing petition to address "in detail" the basis for his standing. Initial Order at 2-3. Petitioner Darke did discuss his standing further in his March 3 response, declaring in toto:

That interest (the health and safety of the requestor and his family, who reside in the vicinity of the Moab facility) would be challenged by the granting of the amendment proposed by the Application as offered by the Applicant/Licensee submittal of December 20, 1996.

The undersigned and his family would suffer direct harm, radiological and other wise by such granting.

Darke March 3 Response at 8-9.

After reviewing that pleading, I issued an additional order that described the parameters of the agency caselaw on standing, including the need for an individual petitioner to make a specific showing of the "distance (in miles)" from the facility at which the petitioner either resides or engages in recreational or other activities, and permitted Petitioner Darke to make a further filing on the subject. Additional Filing Order at 2-3. He made that submission on March 24, 1997, the substance of which is discussed below. Thereafter, although Licensee Atlas in its April 7 response challenged Petitioner Darke's asserted bases for standing, *see* Atlas Response at 5-8, and Petitioner Darke had an opportunity to respond to any of the arguments in that response, *see* Reply Filing Order at 2, he made no further assertions concerning the grounds for his standing to intervene in this proceeding. *See* Darke Reply at 4.

Consequently, on the question of Petitioner Darke's standing to intervene in this proceeding, the pertinent pleading is his March 24, 1997 response in which he provided essentially all the information now before me regarding the basis for his standing. In that filing, Petitioner Darke declared that while he does not live within or on the boundary of the Moab facility, he and his family do undertake certain activities that establish his interests are affected by the facility such that he has standing to intervene in this proceeding. These include (1) obtaining potable water for drinking and cooking from "a source that is within

a short walk” of the Moab facility; (2) using fire fuel driftwood taken from the Colorado River, which flows by the Moab facility; (3) bathing with or in the waters of the Colorado River; (4) using a public telephone that is a “short walk” from the Moab facility; (5) undertaking various other activities, including recreational and educational activities, on public and private lands in “close proximity” to the Moab facility; and (6) using local transportation corridors in “close proximity” to the Moab facility. Darke March 24 Response at 2-3. Petitioner Darke also declared that certain structures, systems, or components found within or “nearby” the facility impede his use of the Colorado River in violation of 33 U.S.C. §§ 401-413 and that the facility precludes him from using certain “necessary” amenities provided by the Colorado River that are “proximate (a short walk)” from the facility. *Id.* at 4. Petitioner Darke then concluded that as a result of these various activities, he and his family “most probably intercept numerous overloaded exposure pathways (some radiological) which originate” within the Moab facility, thereby resulting in “direct harm” to him and to them. *Id.*

In its April 7, 1997 response to Petitioner Darke’s filings, Licensee Atlas argued that he had failed to make any allegation of “injury in fact” sufficient to support a finding that he has standing to be admitted as a party to this proceeding. According to Atlas, the tailings pile at the Moab facility has an interim cover that virtually eliminates windblown particulate emissions so that Atlas complies with the applicable agency dose limits in 10 C.F.R. §§ 20.1301-.1302. Licensee Atlas further declared that Petitioner Darke’s assertions regarding use of water from the Colorado River for drinking, cooking, and bathing are not sufficient because he has not indicated whether the source of this water is surface water or ground water and whether it is upstream or downstream from the Moab facility. Licensee Atlas also maintained Petitioner Darke’s concern about exposure pathways is “nonsense” that bears no relationship to the license amendment at issue. Atlas Response at 5-7.

To be sure, Licensee Atlas’ claim that “regulatory limits” are not being exceeded by offsite releases from the Moab facility is not, standing alone, sufficient to show that Petitioner Darke lacks standing. As was noted recently in the face of a similar assertion, “[r]elative to a threshold standing determination, . . . even minor radiological exposures resulting from a proposed licensee activity can be enough to create the requisite injury in fact.” *General Public Utilities Nuclear Corp.* (Oyster Creek Nuclear Generating Station), LBP-96-23, 44 NRC 143, 158 (1996). As Licensee Atlas’ own annual dose calculations indicate, currently the facility does provide at least some radiological exposures to offsite individuals, albeit small. *See* Atlas Response, exh. C. Further, on this record there is nothing to suggest there is a reasonable expectation that such exposures will not occur during the additional period that is the subject of the

license amendment. As such, the potential for offsite radiological impacts from the facility, and thus for injury in fact to offsite individuals, exists.

By the same token, a showing that there may be some offsite radiological impacts to someone is not enough to establish standing for Petitioner Darke. As the Commission has made clear on a number of occasions, in the context of a proceedings other than those for the grant of a reactor construction permit or operating license, a petitioner who wants to establish "injury in fact" for standing purposes must make some specific showing outlining how the particular radiological (or other cognizable) impacts from the nuclear facility or materials involved in the licensing action at issue can reasonably be assumed to accrue to the petitioner. *See, e.g., Yankee Atomic Electric Co. (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 247-48 (1996); 55 Fed. Reg. 36,801, 36,804 (1990); 54 id. at 8272.* As I noted in my March 11, 1997 memorandum and order, *see Additional Filing Order at 2*, petitioners generally do this by quantifying the distance from the nuclear facility or materials at which they reside or engage in other activities they believe are likely to result in radiological impacts. *See, e.g., Oyster Creek, LBP-96-23, 44 NRC at 157-59.*

Petitioner Darke's problem in this instance is that he has failed to carry his burden to provide the specific information needed to establish his injury in fact.² Simply put, he has not shown any reasonable nexus between himself and any purported radiological impacts. Petitioner Darke certainly has made assertions about potential facility-related airborne and waterborne radiological contacts. He has not, however, delineated these with enough concreteness to establish some impact on him that is sufficient to provide him with standing.³

For instance, Petitioner Darke claims he may suffer radiological impacts as a result of drinking, bathing, and cooking with water from the Colorado River that flows next to the Moab facility. Yet, he has not provided any information that indicates whether these water-related activities are being conducted upstream or downstream from the facility, a fact critical to establishing whether these activities will provide the requisite injury in fact. So too, his description of his other activities near the facility are all quantified with vague terms such as "near," "close proximity," or "in the vicinity." Notwithstanding the Commission's

² Petitioner Darke also refers to impacts on his family in seeking to establish his standing to be a party to this proceeding. His ability to gain standing for himself based on injury in fact to the interests of his spouse or children (especially if those children are not minors) is problematic. *See Detroit Edison Co. (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-470, 7 NRC 473, 474 n.1 (1978)* (mother cannot represent interests of nonminor son attending medical school in vicinity of proposed nuclear facility). Nonetheless, because Petitioner Darke has not sought to establish his interests are based on circumstances different from those of the members of his family, I need not reach this issue.

³ Petitioner Darke does refer to "numerous overloaded exposure pathways (some radiological)" emanating from the Moab facility that will harm him and his family, *see Darke March 24 Response at 4*, apparently suggesting there also is a nonradiological component to his injury in fact. He has not, however, provided any detail about the nature of any purported nonradiological impacts so as to give me a basis for considering them in making a standing determination.

general guidance to afford a liberal construction to petitioner hearing requests, I am unable to find these cryptic references adequate to establish the required nexus with any facility radiological impacts, particularly in light of the repeated guidance given Petitioner Darke about the need to make a specific showing in this regard.⁴

I thus conclude Petitioner Darke has not met his burden of showing that Atlas' requested license amendment will result in injury in fact to him or his family.⁵ Because he has failed to establish this element that is vital to demonstrating his standing to intervene in this proceeding, his hearing request must be dismissed.

III. CONCLUSION

In accordance with 10 C.F.R. § 2.1205(e), (h), Petitioner Darke has established that his hearing request challenging applicant Atlas' December 20, 1996 license amendment application is timely and specifies areas of concern that are germane to the subject matter of the proceeding. Nonetheless, despite multiple opportunities to address the issue, for the reasons outlined above Petitioner Darke has failed to meet his burden to establish his standing to intervene in this proceeding. Accordingly, I deny Petitioner Darke's hearing request and terminate this proceeding.⁶

For the foregoing reasons, it is, this sixteenth day of May 1997, ORDERED that:

1. The January 30, 1997 hearing request of John Francis Darke is *denied* and this proceeding is *dismissed*.
2. In accordance with the provisions of 10 C.F.R. § 2.1205(o), as it rules upon a hearing request, this Memorandum and Order may be appealed to

⁴ In my initial order, I also advised Petitioner Darke that it generally is the practice for participants making factual claims regarding the circumstances that establish standing to do so in affidavit form that is notarized or includes a declaration that the statements are true and are made under penalty of perjury. See Initial Order at 3. As Licensee Atlas notes, Petitioner Darke apparently has made no effort to comply with this guidance. See Atlas Response at 5. Providing this assurance of the accuracy of factual representations about standing is important; nonetheless, because Petitioner Darke appears pro se and generally is making representations about himself (rather than about other individuals), I am not dismissing this case because of his failure to comply with this instruction.

⁵ As was noted above, see *supra* p. 425, Petitioner Darke also has made assertions about facility-related impacts impairing his use of navigable waters in violation of 33 U.S.C. §§ 401-413. Besides suffering from the vagueness problem already identified, it is not apparent how this claim meets the standing requirement that any purported injury in fact come within the "zone of interests" that is being protected by the statutes governing this proceeding.

⁶ In his pleadings, Petitioner Darke repeatedly champions the need to establish a local public document room in the vicinity of the Moab facility. See, e.g., Darke Hearing Request at 1. Because I am denying his hearing request and terminating this proceeding, there is no cause for me to consider that entreaty further. Petitioner Darke does, of course, have toll-free access to information regarding the Moab facility through reference assistance and a public users' on-line data base provided in conjunction with the agency's Washington, D.C. public document room or he can seek facility-related documents through requests under the Freedom of Information Act, 5 U.S.C. § 552.

the Commission by filing an appeal statement that succinctly sets out, with supporting arguments, the errors alleged. To be timely, an appeal statement must be filed within 10 days after this Memorandum and Order is served (i.e., on or before *Monday, June 2, 1997*).

G. Paul Bollwerk, III
ADMINISTRATIVE JUDGE

Rockville, Maryland
May 16, 1997

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

Peter B. Bloch, Presiding Officer
Charles N. Kelber, Special Assistant

In the Matter of

Docket No. 40-8681-MLA
(ASLBP No. 97-726-03-MLA)
(License Amendment)
(Re: Alternate Feed Material)

ENERGY FUELS NUCLEAR, INC.
(White Mesa Uranium Mill)

May 27, 1997

The Presiding Officer in this proceeding under 10 C.F.R. Part 2, Subpart L, explained what was required for a party to show standing, including affidavits of residence, a statement of authorization to represent particular members of the organizations, and a plausible allegation of injury in fact resulting from the amendment that is the subject of the licensing proceeding. Petitioner were permitted to file supplemental filings to fulfill these requirements. In addition, various procedural requirements for Subpart L filings were explained.

RULES OF PRACTICE: STANDING

To attain standing, petitioners should show a plausible way in which activities licensed by the challenged amendment would injure them. The injury must be due to the amendment and not to the license itself, which was granted previously. The injury must occur to individuals whose residence is demonstrated in the filing and whom the organizations are authorized to represent.

MEMORANDUM AND ORDER

(Additional Filings Required)

This proceeding involves a challenge to a license amendment that was issued by the Staff of the Nuclear Regulatory Commission (Staff) on April 2, 1997.¹ The amendment permits the receipt and processing of alternate feed material (i.e., material other than natural ore) at Licensee's White Mesa Uranium Mill located near Blanding, Utah. See 10 C.F.R. Part 40, Appendix A, which sets forth several design criteria and requires that licensing decisions "take into account the risk to the public health and safety and the environment with due consideration to the economic costs involved . . ."; 40 C.F.R. Part 192, Subparts D & E. See also the following nonbinding Staff guidance: "Final Position and Guidance on the Use of Uranium Mill Feed Material Other Than Natural Ores," 60 Fed. Reg. 49,296 (Sept. 22, 1995).

The following requests for a hearing have been filed:

1. Native American People's Historic Foundation, April 16, 1997, Winston M. Mason, Head of Council.
2. Mr. Norman Begay, April 30, 1997. Mr. Begay writes on behalf of himself and his community.
3. Westwater Navajo Community, May 5, 1997, Lula J. Katso, Community Spokesperson.
4. U.S. Department of Energy, May 5, 1997, G. Leah Dever, Assistant Manager for Environmental Management.

The Staff filed its response to these filings on May 21, 1997 (Staff Response). Although the Staff Response is admittedly untimely, based on "some confusion,"² I have decided to permit its filing out-of-time. The Staff Response is very helpful because it reviews in detail the Commission's requirements for standing. In particular, the Staff draws attention to the need to specify "the particular manner in which those persons or entities may be affected by the instant license amendment."

My review of the filings persuades me that there is a need for greater particularity concerning standing. Among petitioners, Mr. Begay comes closest to alleging a ground for standing. He states:

Our Community and our water wells lie adjacent to, as well as downstream and downwind from the EFN Mill. The radionuclides which make up the Cotter Concentrate originally came from Belgium Congo Ore containing approximately 60% Uranium, and now still contain 10%

¹ Letter from Joseph J. Holonich, Chief, Uranium Recovery Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards, April 2, 1997, Attachment 4 to the Letter of the Native American Peoples Historical Foundation, April 25, 1997.

² Staff Response at 2 n.1.

Uranium. Not only does this hazardous waste contain extremely high radioactivity and radon gas properties, but each time it is processed it adds further harmful constituents, which are perhaps more immediately dangerous to human health than the radionuclides. According to reports, your agency, and the Department of Energy have stated that DOE is unable to stabilize the Cotter Concentrate. Therefore, on the basis of concerns for the health and safety of myself, my family, and my community, I ask for standing to argue against bringing these contaminants to the White Mesa Mill.³

Because the license to operate the White Mesa Uranium Mill is not at issue in this proceeding, a petitioner's standing must not be based on harm resulting from the license to operate. The only issues that may be raised must relate to the specific actions proposed to be taken under the license amendment. To show standing, an individual or an organization must show how it may be harmed ("injury in fact") by the amendment.⁴ It is typical in our proceedings that an individual would submit an affidavit concerning where they live and how far that is from the proposed activity. An organization typically would file an affidavit showing that its interests as an organization will be injured or that a particular person or group of people, whom it is authorized to represent, live in particular addresses, stating how far they live from the proposed activity.

In addition to proximity, petitioner should show a plausible way in which activities licensed by the challenged amendment would injure them. For example, Mr. Begay is concerned about the contamination of water wells, and he states that the Cotter Concentrate is "unstable." This, in itself, does not show a plausible mechanism for injury. The license permits these materials to be stored according to prescribed procedures and methods of monitoring. If a petitioner alleges a way in which it fears that this particular material would fail to be properly confined and would escape into the groundwater, then a requirement for standing would appear to be met.⁵ Alternatively, if intervenor can show that there is a law preventing this particular material from being stored pursuant to the amendment, then there may also be a presumption of injury sufficient to establish standing. One way or another, a petitioner must show the specific injury that is feared and how that injury might occur.

At this stage of the proceeding, I will interpret the petition favorably to the petitioner and will not require the same kind of proof of injury that would be required to render a decision in its favor. But a plausible mechanism for

³ Norman Begay's Letter of April 30, 1997, at 1.

⁴ The requirement of "injury in fact" must not be taken literally. It is fulfilled by demonstrating that there is reason to believe an accident may occur. *Curators of the University of Missouri*, LBP-90-18, 31 NRC 559, 566. (1990). Note that this Subpart L case interprets "injury in fact" in light of the extent to which facts may be available to a petitioner.

⁵ A petitioner may not allege an injury to anyone other than itself. For example, a member of the general public may not allege an injury to a worker at the plant. *Florida Power and Light Co.* (St. Lucie Nuclear Power Plant, Units 1 and 2), CLI-89-21, 30 NRC 325, 329 (1989).

injury must be described. I recommend that Petitioners become familiar with an excellent discussion of standing found in *Consumers Power Co.* (Palisades Nuclear Plant), LBP-79-20, 10 NRC 108 (1979).

I note that it is the policy of the United States Nuclear Regulatory Commission to encourage settlement in cases pending before it. Pursuant to that policy, I have encouraged the parties to negotiate and have offered my services in on-the-record mediation. At this time, there is no interest in those efforts and I have abandoned them. Parties are still encouraged to negotiate. Even if they do not negotiate a settlement, parties may find negotiations fruitful in facilitating the exchange of information and devising efficient ways of proceeding with this case. There is no rule prohibiting contact among parties. The Presiding Officer continues to offer, on request, either his own mediation services, which must be on the record, or the mediation services of a Settlement Judge, who could be appointed on request and could assist in private discussions.

Procedural Requirements

In accordance with my authority under 10 C.F.R. § 2.1209, I set forth the following directives regarding the further conduct of this proceeding:

I. SCHEDULE FOR ADDITIONAL FILINGS REGARDING PETITIONERS' HEARING REQUEST

A. Supplements to Petitioners' Hearing Requests

On or before *Monday, June 9, 1997*, Petitioners may file supplements to their hearing requests. In the supplements, a petitioner should address in detail the following items:

1. An interest in the proceeding and how that interest may be affected by the results of the proceeding, including the reasons why the judicial standards for standing are met, so as to be permitted a hearing, with particular reference to the factors set forth in 10 C.F.R. § 2.1205(h); and
2. Amended areas of concern about the license amendment.

Any factual information provided in support of the petitioner's supplement (such as statements providing details regarding the petitioner's proximity to the facility) should be set forth in an accompanying affidavit that (a) is notarized, or (b) states that all statements in the affidavit are true to the best of the affiant's knowledge and belief and are made under penalty of perjury.

B. Answer to Petitioner's Hearing Request and Supplement

This order is being served by express mail. Any Applicant answer to a petitioner's hearing request and any supplement thereto shall be filed so that it is received by all recipients on or before *Monday, June 23, 1997*. A Staff answer likewise shall be filed so that it is received by all recipients on or before *Monday, June 23, 1997*.

II. NOTICE OF APPEARANCE

If they have not already done so, within *15 days* of the date of this Memorandum and Order, each attorney or representative for each participant shall file a notice of appearance complying with the requirements of 10 C.F.R. § 2.713(b). In each notice of appearance, in addition to providing a business address and telephone number, if an attorney or representative has a facsimile number and/or an Internet e-mail address, the attorney or representative should provide that information as well.

III. SERVICE ON THE PRESIDING OFFICER AND THE SPECIAL ASSISTANT

For each pleading or other submission filed before the Presiding Officer or the Commission in this proceeding, in addition to submitting an original and two conforming copies to the Office of the Secretary as required by 10 C.F.R. § 2.1203(c) and serving a copy on every other participant in accordance with sections 2.701(b) and 2.1203(e), a participant should serve conforming copies on the Presiding Officer and on the Special Assistant by one of the following methods:

1. Regular Mail. To complete service via United States Postal Service first-class mail, a participant should send conforming copies to the Presiding Officer and the Special Assistant at the following address:
Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

For regular mail service, the Staff may use the NRC internal mail system (Mail Stop T-3F23) in lieu of first-class mail.

2. Overnight or Hand Delivery. To complete service via overnight (e.g., express mail) or hand delivery, a participant should send conforming copies to the Presiding Officer and the Special Assistant at the following address:

Atomic Safety and Licensing Board Panel
Third Floor, Two White Flint North
11545 Rockville Pike
Rockville, MD 20852

3. Facsimile Transmission.⁶ To complete service by facsimile transmission, a participant should (1) send one copy by facsimile transmission to the attention of the Presiding Officer and the Special Assistant at (301) 415-5599 (verification (301) 415-7405); and (2) that same date, send conforming copies to the Presiding Officer and the Special Assistant by regular mail at an address given in paragraph 1, above.
4. Timely Service. To be timely, any pleading or other submission served on the Presiding Officer and the Special Assistant by hand delivery, facsimile transmission, or e-mail must be received by the Presiding Officer, the Special Assistant, and each of the other parties no later than 4:30 p.m. Eastern Time on the date due. The Secretary of the Commission also should receive a copy, which may be mailed regular mail at the same time the other service is effected.
5. Parties may send, for my convenience, a computer-readable copy of any filing, either on a floppy disk or as an attachment to e-mail. Any format readable by Wordperfect 6.1 would be useful.

IV. MOTIONS FOR EXTENSION OF TIME

For any motion for extension of time filed with the Presiding Officer in this proceeding, except upon a showing of good cause, the participant requesting the extension shall:

1. Ascertain whether and when any other participant intends to oppose or otherwise respond to the motion and apprise the Presiding Officer of that information in the motion; and
2. Serve the motion on the Presiding Officer and the parties so that, if possible, it is in their hands at least three business days before the due date for the pleading or other submission for which an extension is sought.

V. EXHIBITS/ATTACHMENTS TO FILINGS

If a participant files a pleading or other submission with the Presiding Officer that has additional documents appended to it as exhibits or attachments, a

⁶ E-mail filing also will be accepted providing paper copies also are served. The Presiding Officer will respond to questions about e-mail service.

separate alpha or numeric designation (e.g., Exhibit 1, Attachment A) should be given to each appended document, either on the first page of the appended document or on a cover/divider sheet in front of the appended document. Each attachment also should have a tab so that it may be easily accessed without thumbing through all the pages.

It is so ORDERED.

Peter B. Bloch, Presiding Officer
ADMINISTRATIVE JUDGE

Rockville, Maryland
May 27, 1997

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of

Docket No. 70-3070-ML

LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center)

June 30, 1997

The Commission grants petitions filed by the Staff and Louisiana Energy Services for Commission review of the Atomic Safety and Licensing Board's May 1, 1997 Final Initial Decision, LBP-97-8, 45 NRC 367 (1997), and sets a briefing schedule pursuant to 10 C.F.R. § 2.786(d). The Commission also denies Nuclear Energy Institute's (NEI's) motion for leave to file an *amicus curiae* brief in support of the petition for review.

RULES OF PRACTICE: *AMICUS CURIAE*

Our rules contemplate *amicus curiae* briefs only after the Commission grants a petition for review, and do not provide for *amicus* briefs supporting or opposing petitions for review. See 10 C.F.R. § 2.715(d).

ORDER

The Nuclear Regulatory Commission Staff and Louisiana Energy Services (LES) have filed petitions for Commission review of the Atomic Safety and Licensing Board's May 1, 1997 Final Initial Decision, LBP-97-8, 45 NRC 367 (1997), concerning contention J.9 (raising "environmental justice" claims).

This proceeding involves LES's application for a license to construct and operate the Claiborne Enrichment Center (CEC) near Homer, Louisiana. The Intervenor, Citizens Against Nuclear Trash (CANT), opposes the petitions for Commission review. In accordance with the considerations set forth in 10 C.F.R. § 2.786(b)(4), the Commission has decided to grant the petitions and will review the issues raised in the Staff's and LES's petitions.¹

I. SCHEDULING OF BRIEFS

Pursuant to 10 C.F.R. § 2.786(d), the Commission sets the following briefing schedule:²

1. The Staff and LES shall file their briefs on or before August 8, 1997. Each brief shall be no longer than 30 pages.
2. CANT shall file a single responsive brief on or before September 18, 1997. Its response shall not exceed 40 pages. We allow 40 pages for CANT's brief so that CANT will have adequate space to respond to separate approaches that may be taken in the opening briefs of the Staff and LES. It is also possible that CANT will face an *amicus curiae* brief filed by NEI. See discussion below.
3. The Staff and LES may file reply briefs on or before September 30, 1997. Their replies shall not exceed 10 pages each.

Briefs in excess of 10 pages must contain a table of contents, with page references, and a table of cases (alphabetically arranged), statutes, regulations, and other authorities cited, with references to the pages of the brief where they are cited. Page limitations on briefs are exclusive of pages containing a table of contents, table of cases, and of any addendum containing statutes, rules, regulations, etc.

II. MOTION TO FILE *AMICUS CURIAE* BRIEF IN SUPPORT OF PETITIONS

The Nuclear Energy Institute (NEI) has sought leave to file an *amicus curiae* brief in support of the petitions for review. We deny the motion. Our rules

¹ The Commission also has before it three petitions for review, two by CANT and one by LES, raising various challenges to the Board's handling of waste disposal issues, including its decision in LBP-97-3, 45 NRC 99 (1997). In addition, the Commission is considering the briefs filed by the parties after the Commission granted earlier petitions for review raising NEPA and financial qualifications issues. See CLI-97-3, 45 NRC 49 (1997). The Commission will act on those matters in due course.

² In a letter dated June 5, 1997, CANT's lawyers asked the Commission, in setting a briefing schedule, to take into consideration their "previously scheduled family obligations out of town during the entire month of August." LES opposes any delay in the proceeding. The Commission has taken into account both concerns in establishing the briefing schedule in this case.

contemplate *amicus curiae* briefs only after the Commission grants a petition for review, and do not provide for *amicus* briefs supporting or opposing petitions for review. See 10 C.F.R. § 2.715(d); cf. *Sequoyah Fuels Corp. and General Atomics* (Gore, Oklahoma Site), CLI-96-3, 43 NRC 16, 17 (1996). No special circumstances here warrant an exception to our rules.

Without further motion, however, we will permit NEI to file an *amicus* brief on the merits, not to exceed 20 pages, should it choose to do so. See *Sequoyah Fuels Corp.*, 43 NRC at 17. NEI must file its *amicus* brief no later than the filing date of the briefs for the parties whose position NEI supports. See 10 C.F.R. § 2.715(d).

IT IS SO ORDERED.

For the Commission³

JOHN C. HOYLE
Secretary of the Commission

Dated at Rockville, Maryland,
this 30th day of June 1997.

³ Commissioner Dicus was not available for the affirmation of this Order. If she had been present, she would have approved the Order.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

Peter B. Bloch, Presiding Officer
Dr. Peter S. Lam, Special Assistant

In the Matter of

Docket No. 55-20726-SP
(ASLBP No. 97-727-01-SP-R)
(Re: Senior Reactor
Operator License)

RALPH L. TETRICK
(Denial of Application for Reactor
Operator License)

June 25, 1997

The Presiding Officer in this Subpart L proceeding, having requested further information in this remand proceeding, affirmed his earlier determination that Mr. Tetrick had incorrectly answered the remanded question on his Senior Reactor Operator's examination. Plant procedures involved in this question were interpreted to require an understanding of the root cause of the incident described in the question.

**RULES OF PRACTICE: MOTION FOR RECONSIDERATION;
REMAND**

The Presiding Officer expressed confidence that in deciding this case the Commission will be aware that motions for reconsideration are frequently filed before presiding officers, both at the end of cases and after interim orders. *Puerto Rico Electric Power Authority* (North Coast Nuclear Plant, Unit 1), ALAB-648, 14 NRC 34, 37-38 (1981).

MEMORANDUM AND ORDER

(Determination of Remand Question)

Memorandum

The purpose of this Memorandum is to determine the question remanded to me by the Commission, in light of the additional evidence provided to the Commission on appeal and then to me in response to questions asked of the parties.

I. PROCEDURAL HISTORY

On May 20, 1997, the Commission issued CLI-97-5, 45 NRC 355 (1997), concerning an appeal of my initial decision, LBP-97-2, 45 NRC 51, 53 (1997). In that decision, the Commission charged me with redetermining the correctness of Mr. Tetrick's answer to Question 63 on his examination, in light of a letter of May 1, 1997, from R.J. Hovey, Vice President of the Turkey Point Plant (Hovey letter).¹ The Hovey letter was submitted by the NRC Staff to the Commission as an attachment to a Staff brief filed on May 2, 1997.

On May 27, 1997, I issued an unpublished Memorandum and Order in which I asked the parties a series of questions designed to elicit information helpful in determining this remand. In response, the parties filed: (1) Memoranda from Ralph L. Tetrick, with attachments (including plant procedures, a letter from R.J. Hovey of May 1, 1997, and a Memorandum from Brian J. Stamp, undated) dated June 6, 1997 (Tetrick Answers); and (2) "NRC Staff's Response to the Presiding Officer's Memorandum and Order (Questions Relevant to Remand), June 13, 1997 (Staff Answers) and "Supplemental Affidavit of Brian Hughes and Thomas A. Peebles, June 13, 1997 (Staff Supplemental Affidavit).

II. QUESTION 63

Examination Question 63, which is the subject of this remand, stated as follows:

¹ Unless there is a showing of "compelling cause," matters raised for the first time on appeal generally will not be considered, especially when they involve factual matters that could have been raised before the presiding officer. *Puerto Rico Electric Power Authority* (North Coast Nuclear Plant, Unit 1), ALAB-648, 14 NRC 34, 37-38 (1981). In accordance with the Commission's directions in this remanded case, the parties' filings before the Commission are considered to be a part of the decisional record.

Plant conditions:

- *Preparations are being made for refueling operations.*
- *The refueling cavity is filled with the transfer tube gate valve open.*
- *Alarm annunciators H-1/1, SFP LO LEVEL and G=9/5, CNTMT SUMP HI LEVEL are in alarm.*

Which ONE of the following is the required IMMEDIATE ACTION in response to these conditions?

- a. *Verify alarms by checking containment sump level recorder and spent fuel level indication.*
- b. *Sound the containment evacuation alarm.*
- c. *Initiate containment ventilation isolation.*
- d. *Initiate control room ventilation isolation.*

III. THE INITIAL DECISION

In my initial decision, LBP-97-2, I decided, based on the record then before me, that:

The Staff has persuaded me that when two concurrent annunciators sound, indicating that there is an off-normal event that could cause harmful radiation within the containment, that the operator should take the required IMMEDIATE ACTION. Given the important safety problem that is being indicated by two different annunciators, there is not the time to verify that each of the annunciators is working properly. That they sound *together* is enough corroboration to act immediately to prevent injury to the health of plant employees.

45 NRC at 55. Thus, I concluded that the correct response to this question was "b" rather than "a," which was Mr. Tetrick's answer.

IV. ADDITIONAL INFORMATION

A. Applicable Plant Procedures

Mr. Tetrick has demonstrated, in his memorandum of June 6, 1997, that 3-ONOP²-033.2 — Refueling Cavity Seal Failure is not the only plant procedure that requires an immediate action. The phrase "immediate action" also occurs in 3-ARP³-097.CR — Control Room Annunciator Response and in 3-ONOP-033.1 Spent Fuel Pool (SFP) Cooling System Malfunction.

² ONOP stands for "off normal operating procedure."

³ ARP stands for "annunciator response procedure" and also is referred to as "annunciator response guidelines."

B. Important “Note” Contained in Procedure

In the attachments filed with me by Mr. Tetrick, on page 7 of 3-ARP-097.CR, there is a box that sets forth a general principle that the indicated actions are “a guide for operators in responding to *single annunciators*.” Note that they are “a guide.” Note also that they apply to single annunciators and not to multiple annunciators, where understanding the pattern or the root cause becomes more important and where “applicable off-normal and emergency procedures” come into play. The relevant section of 3-ARP-097.CR, called *NOTES*, states:

1. The annunciator panel attachments indicate appropriate operator action for Control Room panel annunciators. The actions listed are intended to be a guide for operators in responding to *single* annunciators and not intended to be a substitute for good judgment based on thorough understanding of plant conditions and equipment.
2. Many off-normal plant conditions will result in several annunciators lighting almost simultaneously. In such a case, operators are expected to respond to the root cause of the problem and maintain the unit in a safe condition IAW [in accordance with] applicable off-normal and emergency procedures. This action may not necessarily correspond to that of the attachments.

C. Staff Argument

The Staff has discussed extensively the root cause of the signals postulated to be present in Question 63. It bases its answer to the question on this understanding of root cause. It states (Staff Supplemental Affidavit at 9-11):

We have carefully considered Mr. Tetrick’s answer to this question. In our view, it reflects a fundamental misunderstanding of the importance and significance of an ONOP, in contrast to a nuclear facility’s many other plant procedures. Further, Mr. Tetrick’s answer ignores the significance of the specific plant conditions described in the stem of Question 63, which must be considered in an SRO applicant’s selection of the proper answer to this question. Question 63 explicitly posited the following specific plant conditions:

Plant conditions:

- Preparations are being made for refueling operations.
- The refueling cavity is filled with the transfer tube gate valve open.
- Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5, CNTMT SUMP HI LEVEL are in alarm.

Under these plant conditions, where these two mutually supportive and confirmatory annunciators (spent fuel pool low level and containment sump high level) are sounding together, a competent applicant for a senior reactor operator license should have recognized, unequivocally, that the operator is *required* to sound the containment evacuation alarm, in accordance with 3-ONOP-033.2. We note that although Mr. Tetrick’s July 1996 submittal did not discuss this ONOP, in his filings before the Presiding Officer in September and December 1996 he

agreed the two annunciators specified in Question 63 are "mutually supportive and sufficient to enter 3-ONOP-033.2 "REFUELING CAVITY SEAL FAILURE."

. . . Question 63 does not constitute an abstract question of only theoretical interest. Rather, the question seeks to test applicants on their fundamental competence to respond to actual plant conditions, specified therein. Question 63 describes a potential refueling cavity seal failure, during refueling operations. The initial plant conditions provided in the stem of the question state that "the refueling cavity is filled with the transfer tube gate valve open." This condition means that the Spent Fuel Pool is connected (through the transfer tube) to the refueling cavity in the Containment Building. Another initial condition states "Alarm annunciators H-1/1, SFP LO LEVEL and G-9/5, CNTMT SUMP HI LEVEL are in alarm." The concurrent sounding of these two alarms would indicate that the water level has decreased in the Spent Fuel Pool and has increased in the Containment Building sump. Because the Spent Fuel Pool is connected to the Refueling Cavity (inside the Containment Building) through the transfer canal, the actuation of these two alarms at the same time would confirm leakage from the Refueling Cavity to the Containment Building sump. This leakage would most probably be due to the refueling cavity seal leaking or failing. Under the conditions described in Question 63, prompt notification to plant personnel of the nature of the emergency by sounding the containment evacuation alarm is the only appropriate IMMEDIATE ACTION.

. . . Question 63 is based upon a real-life incident that occurred at the Haddam Neck plant, where a refueling cavity seal failure resulted in a substantial drainage of the water in the refueling cavity within a matter of minutes — an event which could have potentially resulted in lethal radiation doses to plant personnel. This event led to the issuance of IE Bulletin 84-03 on August 24, 1984. At the time of the event, the refueling cavity was filled in preparation for refueling and, fortuitously, the transfer tube gate valve (which connects the spent fuel pool to the refueling cavity) was closed. The Staff evaluated this event as Generic Issue 82, and determined that it has significant safety implications for all water-cooled nuclear power plants in the United States, — and each such facility, including Turkey Point, was required to address this problem. See NUREG/CR-4525, "Closeout of IE Bulletin 84-03: Refueling Cavity Water Seal" (June 1990) (portions of which are provided as Attachment 1 hereto).

It should be further noted that Question 63 posits a situation in which "the refueling cavity is filled with the transfer tube gate valve *open*" — unlike the event at Haddam Neck, where the gate was closed. While significant radiation doses may have been avoided at Haddam Neck due to the transfer tube gate being closed, a different result might have occurred at Turkey Point, under the conditions stated in Question 63, if the plant operators decided, like Mr. Tetrick, to verify alarms before taking the required "IMMEDIATE ACTION" of sounding the containment evacuation alarm.

V. ANALYSIS AND CONCLUSIONS

I am persuaded by the Staff that I should uphold my initial determination. An operator must act on an understanding of the root cause of an event, trusting the plant's instruments to deduce what is happening. Turkey Point does have procedures for "responding to *single annunciators*." Note from 3-ARP-097.CR, discussed above at p. 444. As also discussed above, at p. 444, these procedures

specifically state that they are “not intended to be a substitute for good judgment based on thorough understanding of plant conditions and equipment.”⁴

I asked several questions in my order of May 27. Among those questions were the following:

What precisely would he [Mr. Tetrick] do during these 20 seconds [that he says he would use to verify the validity of instrument readings]? What evidence might he find that would persuade him not to take the required IMMEDIATE ACTION after he took steps to verify the alarm?

The answers to these questions were very important because they would show whether there was any legitimate reason to hesitate in taking the immediate action required by the ONOP. For example, is there some instrument reading that could be easily taken and that would give an operator confidence that the instruments were wrong? If so, then the decision to check further could be based on an understanding of what was happening in the reactor and not based solely on a mechanical reading of a tangential provision that relates to single annunciators. However, Mr. Tetrick did not respond directly to my question. In particular, he gave no indication of any instrument reading or set of readings that would persuade him not to take the required immediate action in the 3-ONOP-033.2. Tetrick Answers, bottom of p. 1 (responding to Question #2).

I conclude that Mr. Tetrick should have acted from an understanding of the root cause of the event portrayed in Question 63. Had he done so, then only answer “b.” would be correct. His failure to understand that failed to mitigate the risks described by Staff and quoted at p. 445, above.

I am unpersuaded by Mr. Tetrick’s attempt to rely on the Turkey Point training program and “management expectations.” See Tetrick Answers at 1, second paragraph from the bottom. He is responsible for knowing the correct, safe action to take in response to plant conditions. The NRC cannot be expected to certify an operator based on his reliance on an incorrect response allegedly taught to him. NRC licenses only those operators who demonstrate that they will respond correctly and safely to plant conditions.

I am not convinced by the letter from R.J. Hovey of Florida Power and Light to Mr. Stuart A. Richards of the NRC. (Tetrick Reply, unnumbered Attachment.) Mr. Hovey states, in one key sentence, “If the question is interpreted to be asking for an immediate action for the receipt of an annunciator, response (a) is correct.” I do not interpret the question as Mr. Hovey suggests. There is not one annunciator, but two. What is called for by the question is an understanding of plant conditions and how to respond to two consistent,

⁴ Procedure 3-ONOP-033.1 requires an “immediate action” consisting of: “verify annunciated alarm is valid.” However, with the simultaneous indications postulated in Question 63, the two alarms verify the validity of one another. Thus, there is no further need to verify these alarms.

simultaneous annunciators. Moreover, the Annunciator Response Procedure (ARP) contains a note that makes it clear that it cannot be mechanically applied under these circumstances. (See Note 5 of 3-ARP-097, CR, above.)

Similarly, I am not persuaded by the memorandum of Brian J. Stamp, Acting Operations Supervisor, because I consider his understanding of Question 63 to be the same as that of Mr. Tetrick and thus incorrect. (Tetrick Reply, unnumbered Attachment).

I conclude, after considering all the information before me, that Mr. Tetrick answered Question 63 incorrectly.

VI. PROCEDURAL IMPLICATIONS

In this remand, I have addressed information filed by Mr. Tetrick that was not filed in a timely manner prior to my Initial Decision. I would note that the Staff's appeal also seems to be based on new information. I am confident that in deciding this case the Commission will be aware that motions for reconsideration are frequently filed before presiding officers, both at the end of cases and after interim orders. It is important for the efficiency of licensing procedures that there be a clear principle that requires parties to file information prior to the decisions of judges rather than waiting for an opinion before adding new information to the record.

Order

For all the foregoing reasons and upon consideration of the entire record in this matter, it is, this 25th day of June 1997, ORDERED that:

In response to CLI-97-5, 45 NRC 355 (1997), the Presiding Officer reaffirms his determination that the response of Ralph L. Tetrick to Question 63 of his Examination to be a Senior Reactor Operator (SRO) was incorrect.

Peter B. Bloch, Presiding Officer
ADMINISTRATIVE JUDGE

Rockville, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Carl J. Paperiello, Director

In the Matter of

Docket No. 040-8948
(License No. SMB-1507)SHIELDALLOY METALLURGICAL
CORPORATION
(Newfield, New Jersey)

June 6, 1997

By a letter dated July 22, 1996, Mr. Sherwood Bauman (Petitioner) requested that the following actions be taken with regard to NRC Licensee Shieldalloy Metallurgical Corporation (SMC): (1) that the previous site Licensee have its license reinstated such that it and SMC become co-responsible for the remediation and decommissioning of the SMC site; (2) that all NRC or State of Ohio parties involved in wrongdoing related to this issue be dismissed from employment and criminally charged where appropriate; (3) that the NRC terminate its development of an environmental impact statement (EIS) for the SMC site; (4) in place of the EIS, the NRC order SMC and its predecessor to submit a decommissioning plan limited to remediation of licensed material; and (5) that the Ohio Environmental Protection Agency and Department of Health should evaluate all unlicensed slag found at the SMC site. The request was considered as a petition submitted pursuant to 10 C.F.R. § 2.206.

In a Director's Decision issued on June 6, 1997, the Director of Nuclear Material Safety and Safeguards denied the relief sought by Petitioner. The Director concluded that it would be inappropriate to reinstate the previous Licensee's license for the SMC site, as SMC was the current Licensee and therefore responsible for decommissioning the site. For similar reasons, the Director denied Petitioner's request to order SMC and the previous Licensee to submit a decommissioning plan. With regard to Petitioner's allegations of wrongdoing, with respect to any such activity by NRC employees the allegation was referred to the NRC Office of the Inspector General. The Director also

concluded that the current EIS was properly evaluating all slag at the SMC site, contrary to Petitioner's claim that the scope of the EIS exceeded NRC authority. Finally, the Director concluded that Petitioner's request for action by State of Ohio agencies was properly addressed by those agencies and not the NRC.

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

I. INTRODUCTION

By letter dated July 22, 1996, addressed to the U.S. Nuclear Regulatory Commission (NRC) and Ohio Department of Health (ODH), Sherwood Bauman, Chairperson of the organization "Save Wills Creek Water Resources Committee" (Petitioner), requested certain actions concerning NRC Licensee Shieldalloy Metallurgical Corporation (Shieldalloy) and former NRC Licensee Foote Mineral (now Cyprus Foote Mineral Company (CFM)). NRC is treating the request as a petition under 10 C.F.R. § 2.206 of the Commission's regulations. The Petitioner requested that the following actions be taken:

- (1) NRC should reinstate Foote Mineral's original license so that Shieldalloy and CFM become co-responsible licensees concerning the proper remediation and decommissioning of the Shieldalloy site;
- (2) Any and all parties involved in any wrongdoing, as alleged in the Petitioner's letter, should be terminated from employment, and, where appropriate, criminal charges pursued;
- (3) NRC should terminate the development of the environmental impact statement (EIS) for the Shieldalloy site;
- (4) In place of the EIS, Shieldalloy and CFM should be jointly ordered to submit a decommissioning plan, for licensed material, that includes only a plan to remediate licensed material, including grading and evaluation of all various assorted options. One option considered should be offsite disposal at a licensed disposal facility; and
- (5) The Ohio Environmental Protection Agency (OEPA) and Ohio Department of Health (ODH) should evaluate all unlicensed slag found at the Shieldalloy site.

NRC acknowledged receipt of the petition in a letter to the Petitioner dated October 11, 1996. The petition was also noticed in the *Federal Register* on April 10, 1997 (62 Fed. Reg. 17,650).¹

¹ Normally, the Petitioner (by letter) and the public (through a *Federal Register* notice) are notified at approximately the same time. In this case, because of an administrative omission, the *Federal Register* notice was not published until April 1997.

The Petitioner also sent an undated letter to President Clinton at approximately the same time as his July 22, 1996 letter to NRC. The White House referred that letter to NRC for response. All of the substantive issues raised in that letter are addressed in this Director's Decision.

II. BACKGROUND

Plant History

Shieldalloy owns and operates a plant that produces ferroalloys, near the city of Cambridge, Ohio. Cambridge is in eastern Ohio, approximately 130 km (80 miles) east of Columbus, Ohio, on Interstate 70. The facility is between Cambridge and Byesville, Ohio, and within the valley of Wills Creek, the major stream in the area.

Ferroalloys are mixtures (alloys) of iron and one or more other elements (e.g., vanadium, titanium, and niobium) that are typically used in steel production or other alloy manufacturing processes. The principal alloy produced today at the Shieldalloy plant is a 60% vanadium/40% iron alloy. Shieldalloy sells its product to steel manufacturing companies, which then add it to batches of steel to produce vanadium alloy steels with a fraction of 1% concentration of vanadium. Vanadium imparts increased strength and hardness to steel.

Facility operations began in the early 1950s under the ownership of Vanadium Corporation of America (VCA). Foote Mineral Company merged with VCA in 1967. In 1987, Shieldalloy purchased the facility from Foote Mineral Company and has continued alloy production at the site since then. The plant has produced a variety of alloys for the steel industry over the years.

The production of metal alloys has resulted in waste byproducts, the principal one being slag, a hard, rock-like residue. During alloy production, almost all radionuclides contained in the incoming ores were incorporated into the waste slag. Since the inception of operations until the late 1980s, the facility disposed of most of its waste slags and other wastes on site. At the present time, the facility's waste is largely contained in the East and West slag piles (named for their onsite locations). Together they contain approximately 250,000 cubic meters (seven million cubic feet) of slag and cover approximately 5.7 hectares (14 acres) of land. The slag itself contains both radioactive materials, such as uranium and thorium isotopes (including their daughter products² such as radium and radon), and nonradioactive metals such as vanadium, chromium, arsenic, copper, and zinc. The slag produced today is largely recycled in steel

² *Daughter products* are atomic species (or nuclides) formed by the radioactive decay of another nuclide, which is called the "parent." For example, when U^{238} decays, Th^{234} is produced. This thorium isotope also decays and produces additional "daughters." Radium and radon are daughter products in the uranium decay chain.

manufacturing as a flux, i.e., a material that removes impurities. Shieldalloy halted onsite slag disposal in the late 1980s.

Several types of radioactive slag are contained in the East and West piles. In the early years of plant operation, ferrocolumbium (now known as ferroniobium) ores were used for alloy production. These ores contained licensable quantities of source material (i.e., uranium (U) and/or thorium (Th) in concentrations greater than 0.05%). See 10 C.F.R. §§ 40.4, 40.13(a). The slag from processing these ores contains elevated concentrations of U^{238} and Th^{232} and their daughter products, and emits gamma radiation that is easily detected.

Two other types of slag at the site, ferrovanadium and Grainal[®], are also radioactive, but neither was produced under the original license that expired in 1975. Radioactive ferrovanadium slag is believed to have resulted from the plant using vanadium concentrates as feed material for alloy production. These concentrates probably resulted from ores processed in another facility to remove the uranium for use in weapons and/or nuclear fuel production. The radioactive daughter products of the uranium, such as Th^{230} , and valuable elements, such as vanadium, remained in the byproduct material. Only small amounts of the parent radionuclides, U^{238} and U^{235} , and much less than would be expected in material that had not been processed to remove these radionuclides, are present.

Unlike the ferroniobium slag produced under the original license, the radioactivity of the ferrovanadium and Grainal[®] slags is difficult to detect. Some radionuclides in the ferroniobium slag are strong emitters of gamma radiation, which can easily be detected with hand-held instruments. The ferrovanadium and Grainal[®] slag radiation is principally emitted as alpha particles from Th^{230} , which are much more difficult to detect with field instruments. The significant radioactivity in these slags was not discovered until after they were produced. The license issued to Shieldalloy in 1987 is for "uranium and thorium . . . as a contaminant in slag from previous alloy furnace operations."

Both the radioactive materials and metals contained in the onsite East and West slag piles could have potentially adverse effects on human health and the environment. In fact, some metals have leached into streams and sediments next to the slag piles. Little or no migration of radioactive materials has taken place to date. Because of the potential effects of the slag on the environment and human health, both the State of Ohio and NRC plan to oversee remediation and cleanup of contamination at the site.

Ferrovanadium slag containing small amounts of radioactive contamination and possibly other slag with radioactive elements have been used in some residential and commercial properties in the Cambridge, Ohio area. The slag was sold or given away by the company for use as construction and driveway fill material before 1987. The short-term hazard from this slag is negligible. The long-term hazard is small and principally derives from unlikely scenarios such as a family growing crops adjacent to their driveway for their consumption as

food. Most calculated doses are a fraction of background radiation. CFM has a separate program under way to identify these properties, evaluate any long-term hazards, and perform any necessary remediation. Several properties have been identified for remediation, and CFM is taking steps to remove the material and safely store it elsewhere. Although the homeowners possess the slag, CFM is carrying out measures to ensure that the offsite slag is addressed, although CFM is no longer a licensee. NRC and the State of Ohio are overseeing CFM's evaluation and remediation of these offsite properties, and have met with the public in the area to discuss the issue.

NRC Regulatory Program Related to Decommissioning the Shieldalloy Facility

VCA and its successor, Foote Mineral Company, held a license to possess source material from 1953 to 1975. At that time, Foote Mineral allowed the license to expire and did not request its renewal, although it continued to possess source material. In 1987, Shieldalloy obtained an NRC license (SMB-1507) for the possession of the source material at the facility. When a licensee is no longer performing the principal activities for which the license was issued (in this case, metal alloy production from radioactive ores), NRC regulations require that the site be decommissioned and the license terminated. *See* 10 C.F.R. § 40.42. Thus, NRC's regulatory program for the Shieldalloy site is directed toward these goals.

In 1987 and 1990, Shieldalloy submitted decommissioning plans to NRC proposing in-situ disposal of the slag piles. Subsequent to the development of these plans, however, NRC determined that an environmental impact statement (EIS) would need to be prepared, in accordance with NRC regulations contained in 10 C.F.R. Part 51, which implements the National Environmental Policy Act of 1969 (NEPA). In order to evaluate Shieldalloy's proposal for onsite disposal of the slag piles, it is necessary to assess impacts on the environment, through preparation of an EIS. The EIS examines onsite disposal alternatives, as well as other alternatives including offsite disposal of the slag.

Under the Atomic Energy Act of 1954 (AEA), NRC is responsible for regulating the safe use of certain radioactive materials (source, byproduct, and special nuclear radioactive materials) to ensure that public health and safety are protected from the effects of radiation. Under NEPA, NRC is obligated to take a range of environmental impacts into account in its decisionmaking process on decommissioning alternatives. The environmental costs of an action are to be weighed against its benefits. As described above, NRC considers the regulatory decision on decommissioning of the Shieldalloy facility to be a major federal action that may significantly affect the quality of the human environment. For that reason, and pursuant to NRC regulations in Part 51 implementing NEPA, NRC is preparing an EIS. The scope of the EIS includes both radiological and

nonradiological impacts of the proposed action and alternatives to it, including impacts on land use, air quality, noise, and transportation, in addition to the radiological impacts to the public that NRC regulates under the AEA.

When the EIS is completed (expected to be in late 1997), Shieldalloy will be required, under NRC regulations in section 40.42, to submit a revised decommissioning plan consistent with the findings of the EIS. Thus, Shieldalloy's previous submittals of decommissioning plans will be superseded by the newest one.

Ohio's Regulatory Program for Remediation of the Shieldalloy Site

The Ohio Environmental Protection Agency also has a program to oversee remediation of the Shieldalloy facility, consistent with its implementation of the Comprehensive Environmental Response, Compensation, and Liabilities Act (CERCLA). Ohio's effort covers contamination at the facility and on property next to the site (mostly wetlands and stream sediments). Vanadium compounds and other waste have migrated into soils and sediments, both on site and off site, and into a stream that runs through the property. The State has entered into a proposed Preliminary Injunction Consent Order (Consent Order) to require Shieldalloy and CFM to carry out a remediation plan described in Ohio's Decision Document.³ The proposed Consent Order would also require Shieldalloy and CFM to pay civil penalties to Ohio. Public comments were received by the State of Ohio on the proposed Consent Order, and it is expected to be made final in the near future.

Relationship Between State of Ohio and NRC Programs for Shieldalloy

Remediation of the Shieldalloy site involves various potential pollutants regulated under overlapping laws. Ohio is responsible for overseeing the remediation of contamination under the CERCLA process. Pursuant to NEPA, NRC is responsible for considering the impacts on the environment from all contamination at the facility in evaluating various alternatives for site remediation. Under the AEA, and once the EIS is completed and a decommissioning program approved, NRC is also responsible for ensuring that the site is properly decommissioned, in this case meaning that radiological contamination is reduced to safe levels, and if onsite disposal is approved, that appropriate institutional controls for long-term land use and monitoring are established and implemented. NRC and the State of Ohio have been coordinating their individual efforts to ensure that a

³ Ohio Environmental Protection Agency Decision Document for the Shieldalloy Metallurgical Corporation Site, Cambridge, Ohio EPA, April 1, 1997.

coordinated approach to site remediation is required of Shieldalloy and CFM by the Federal and State governments.

III. DISCUSSION

NRC Staff has examined the Petitioner's requests in his Petition of July 22, 1996, as follows:

- (1) The NRC should reinstate FM's original license so that Shieldalloy and CFM become co-responsible licensees with regard to the proper remediation and decommissioning of the Shieldalloy site.

The Petitioner argues that Foote Mineral should now be made a co-responsible licensee along with Shieldalloy because Foote Mineral allowed the license to expire and it was not appropriately retired by NRC. The Petitioner states that NRC did not investigate the Licensee's claims that no materials of licensable concern were remaining on site when the license expired.

In a September 9, 1975 letter, the NRC notified Foote Mineral Company that its Source Material License (SMB-850) had expired on August 31, 1975. FMC submitted a "Certificate of Disposition of Materials, AEC-314" to the NRC on September 15, 1975, and the NRC retired the license on October 14, 1975. A site visit was not conducted by NRC Staff to verify disposal of the licensed material. As NRC stated to the Petitioner in a January 19, 1995 letter from NRC's Region III office, .

Although the license record is unclear, it appears to NRC staff that Foote Mineral may have mistakenly assumed that thorium and uranium in the slag were no longer considered source material because their concentrations were generally less than 0.05% by weight. The NRC retired the license based on the completed AEC-314 form, which indicated that "No materials have been procured by the licensee."⁴

Retirement of expired licenses without conducting an onsite inspection was accepted NRC practice in 1975, although the policy has since changed to require onsite inspection to verify that sites of this type have been properly decontaminated. There is no evidence that Foote Mineral Company personnel committed any wrongdoing in this matter.

⁴The January 19, 1995 letter from NRC's Region III office to the Petitioner stated that AEC Form 314 indicated that "all remaining source material (e.g., ores) had been transferred and no longer existed on the Cambridge site. Since the license was retired and licensed operations ceased, the NRC did not inspect during the period of October 1975 until early 1987." AEC Form 314 states that "No materials have been procured by the licensee," as noted above. In any case, there appears to have been confusion by the Licensee over what constituted source material, and the Licensee appears to have mistakenly assumed that the slag was not covered by the existing license. NRC's retiring of the license was based on the information in AEC-314 that no licensable material was on site.

With respect to issuing a license to CFM, NRC's licensing authority is contained in the AEA, and specific licensing provisions have been incorporated into NRC's regulations in 10 C.F.R. Part 40. See 10 C.F.R. § 40.1(a). The regulations generally require that, where applicable, a possessor of radioactive materials obtain an NRC license. Shieldalloy is the owner and possessor of the slag piles, and controls them in accordance with NRC license SMB-1507. Thus, NRC regulates the radioactive materials on the Shieldalloy site through its Licensee, Shieldalloy Metallurgical Corporation.

The State of Ohio, however, has entered into a proposed Consent Order with Shieldalloy and CFM that would require those companies to implement remediation activities at the site. Thus, the Petitioner's request that CFM be made co-responsible for remediation of the site is satisfied in part by that Consent Order. CFM's responsibility, however, is defined and required by the proposed Consent Order with the State of Ohio and not by NRC license as the Petitioner had requested. NRC is satisfied that this approach is adequately protecting public health and safety.

For the above reasons, the Petitioner's request that CFM be made a co-responsible Licensee for remediation of the site is denied.

- (2) Any and all parties involved in any wrongdoing, as alleged in the Petitioner's letter, should be terminated from employment, and where appropriate, criminal charges pursued.

As a general matter, NRC takes enforcement action against individuals who engage in deliberate misconduct involving NRC-regulated activities. However, the Petitioner has not provided any specific information to support a charge of deliberate misconduct by any individual. As noted earlier in this response, Foote Mineral did provide incorrect information more than 20 years ago to support NRC's retiring of the license. It appears that they mistakenly assumed that the uranium and thorium in the slag were no longer considered source material and thus did not require a license. There is no evidence of deliberate misconduct by Foote Mineral Company in this matter.

As a separate matter, Petitioner's assertions of wrongdoing by NRC employees (i.e., collusion with Ohio agencies regarding jurisdiction of offsite slag so as to avoid "legal problems"), have been referred to the NRC Office of the Inspector General (OIG).

- (3) The NRC should terminate the development of the environmental impact statement (EIS) for the Shieldalloy site.

The Petitioner requests that the current EIS being developed for this facility be terminated, as federal law, according to the Petitioner, does not allow NRC to evaluate waste streams that fall outside of its jurisdictional control. According to

the Petitioner, the EIS is evaluating both "licensed" and "unlicensed" slag, which exceeds NRC's authority. The Petitioner also argues that NRC consideration of "unlicensed" materials will result in inadequate protection of the public from "licensed" materials.

The Petitioner is correct that NRC is evaluating all of the onsite slag as part of the EIS, including nonradioactive slag containing metals such as vanadium. The Petitioner is in error, however, in stating that federal law does not allow NRC to evaluate these wastes. The requirements to assess environmental impacts of major federal actions affecting the environment under NEPA are quite broad and extend beyond NRC's usual licensing authority under the AEA. Environmental impacts that are to be assessed under NEPA include impacts on local schools, traffic, and noise that result from different alternatives for remediating the site. The environmental impacts that are required to be evaluated also include those resulting from onsite chemicals (including vanadium and other metals contained in the slag and their possible migration into groundwater), in addition to radioactive materials. NRC's draft EIS issued for public comment (NUREG-1543, July 1996) contains a comprehensive discussion of all environmental impacts, not just those from radioactive materials. Thus, contrary to the Petitioner's assertion, federal law in this case requires NRC to consider a broad range of environmental impacts and, therefore, all of the slag at the facility. Whether the slag is "licensed" or "unlicensed" is not a factor in determining the scope of the EIS.

The Petitioner also states, as a reason for this request, that the radiation doses to members of the public would be well above 600 millirem/year (mrem/yr) from licensed materials, and higher than those calculated for "licensed" and "unlicensed" waste when included together. The Petitioner is incorrect. In the draft EIS (NUREG-1543, July 1996), NRC has modeled the slag piles as they currently exist, and used conservative modeling assumptions to help ensure that actual releases, if any, will be bounded by the EIS calculations. These calculations of radiation doses to members of the public are based on the actual slag piles, and are not affected by any arbitrary divisions of the material into, for example, "licensed" and "unlicensed" slag. Each pile has certain concentrations of radionuclides and chemicals, and each is modeled in the EIS. Releases from both piles are used to evaluate potential impacts on human health. In the draft EIS analysis, NRC has calculated a maximum dose of 6 mrem/yr for an offsite individual. The annual cancer mortality risk for this dose is approximately 3×10^{-6} . NRC has also calculated a radiation dose of 42 mrem/yr to an onsite residential farmer, when both piles are capped with clay. The annual cancer mortality risk for this dose is approximately 2×10^{-5} . (The residential farmer scenario assumes failure of institutional controls, such as fences and deed restrictions. Then, the hypothetical farmer that establishes a residence and farm on site is assumed to drink water obtained from a well that

is drilled adjacent to the piles, and eat crops grown on site that are irrigated with groundwater from the well.)

In summary, as explained above, NRC is appropriately evaluating the environmental impacts of all slag at the Shieldalloy site. Therefore, this request is denied.

- (4) In place of the EIS, Shieldalloy and CFM should be jointly ordered to submit a decommissioning plan for licensed material that includes only a plan to remediate licensed material, including grading and evaluation of all various assorted options. One option considered should be offsite disposal at a licensed disposal facility.

As noted above, Shieldalloy, as the NRC Licensee, is responsible for radiological decommissioning of the site. Therefore, this request is denied for the same reasons as the request to require that CFM obtain an NRC license. Furthermore, as noted above, the option of offsite disposal of the slag is being considered, albeit pursuant to the EIS and not the Petitioner's suggested joint decommissioning plan. Finally, the Staff has previously noted, in response to the first request, that the State of Ohio has made CFM responsible for certain aspects of remediation by means of a proposed Consent Order.

- (5) The Ohio Environmental Protection Agency and Ohio Department of Health should evaluate all unlicensed slag found at the Shieldalloy site.

This request can only be implemented by the State of Ohio and is, therefore, not properly addressed here. The Petitioner did contact the Ohio Department of Health regarding his request. As noted earlier, however, the State of Ohio has entered into a proposed Consent Order with CFM and Shieldalloy, and has been conducting its own review of all of the materials at the site in accordance with CERCLA.

IV. CONCLUSION

For the reasons discussed above, the Petitioner's requests for action pursuant to section 2.206 are denied. A copy of this Decision will be placed in the Commission's Public Document Room at 2120 L Street, NW, Washington, DC, 20555, and at the local public document room for the Shieldalloy facility in the Guernsey County Public Library. The Director's Decision will also be made available on the NRC Electronic Bulletin Board at 1-800-952-9676. A copy of this Decision will be filed with the Secretary for the Commission's review, in accordance with section 2.206.

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 6th day of June 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. 030-16055

ADVANCED MEDICAL SYSTEMS, INC.
(Cleveland, Ohio)

June 13, 1997

The Director of the Office of Nuclear Material Safety and Safeguards (NMSS) denies a petition filed with the Nuclear Regulatory Commission (NRC or Commission) by letter dated March 3, 1993, by William B. Schatz, Esq., on behalf of the Northeast Ohio Regional Sewer District (District or Petitioner), requesting that actions be taken regarding Advanced Medical Systems, Inc. (the Licensee). The petition was partially granted, as explained in the Decision. The Director denies the remaining requests of the petition on the basis of analysis of the technical issues and the Commission's authority to grant the requested relief, set forth in the Decision, which analysis showed that the Commission did not have such authority and that no technical basis warranted granting the petition.

JURISDICTION

No statute or regulation grants the Commission authority to require a licensee to pay, in effect, compensatory damages to private individuals. *Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 269 (1996). A court of competent jurisdiction, and not the NRC, is the proper forum for such an individual to seek compensatory damages from a licensee.

TECHNICAL ISSUE DISCUSSED

The following technical issue is discussed: Contamination of sewer line.

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

I. INTRODUCTION

By letter dated March 3, 1993, addressed to Mr. James Taylor, *former* Executive Director for Operations, U.S. Nuclear Regulatory Commission (NRC), William B. Schatz, Esq., on behalf of the Northeast Ohio Regional Sewer District (District), requested that NRC take action with respect to Advanced Medical Systems, Inc. (AMS), of Cleveland, Ohio, an NRC Licensee. The District requested, pursuant to 10 C.F.R. § 2.206, that NRC: (1) modify AMS License No. 34-19089-01 to require that AMS assume all costs resulting from the offsite release of cobalt-60 that has been deposited at the District's Southerly Wastewater Treatment Center (SWTC); and (2) order AMS to decontaminate the sewer connecting its London Road facility with the public sewer at London Road, and continue downstream with such decontamination to the extent that sampling indicates is necessary.

The District alleges the following bases for its request: (1) cobalt-60 has been discovered in the ash piles resulting from the incineration of sewage sludge at the District's SWTC; (2) AMS is the only Licensee in the District's service area authorized to process cobalt-60 in a loose metallic form consistent with the form present in the ash; (3) AMS is the only entity (except for the former owner of the London Road facility) that has reported discharging cobalt-60 to the sanitary sewer system leading to the SWTC; (4) NRC documents present ample evidence of cobalt-60 contamination at the London Road facility, including numerous drains inside the building; (5) there are excessive exposure rates in the sewer connecting the building to the public sewer system; (6) this sewer line has been classified as a restricted area, which effectively denies the District access to the manhole for sampling industrial discharges; and (7) the AMS London Road facility is the source of the cobalt-60 at the SWTC.

By letter dated April 2, 1993, the Director, Office of Nuclear Material Safety and Safeguards, NRC, formally acknowledged receipt of the petition and informed the District that its request was being treated pursuant to section 2.206 of the Commission's regulations. A notice of the receipt of the petition was published in the *Federal Register* on Tuesday, April 13, 1993 (58 Fed. Reg. 19,282). Staff sent a copy of the letter dated April 2, 1993, with a copy of the petition, to AMS.

By letters dated September 13, 1994, October 13, 1994, and April 29, 1996, the District filed supplements to its March 3, 1993 petition. The District's September 1994 supplement requested that NRC commence enforcement actions against AMS for violations of 10 C.F.R. §§ 20.401(c)(3) and 20.303(a), based on assertions that the disposal records maintained by AMS are grossly inaccurate, in

violation of section 20.401(c)(3), and that AMS discharged material to the sewer that was not readily soluble in or dispersible in water, in violation of section 20.303(a). In addition, the September 1994 supplement requested that the March 3, 1993 petition be granted immediately insofar as it requested that AMS be held responsible for all costs arising from contamination of the District's treatment plant and that AMS be required to decontaminate the sewer downstream from the London Road facility. In its October 1994 supplement, the District requested that NRC commence an enforcement action against AMS for violation of 10 C.F.R. § 20.2003, based on the assertion that AMS had recently discharged cobalt-60 to the sewer that was not soluble or readily dispersible biological material, in violation of that provision. In its April 1996 supplement, the District requested NRC action on a license requiring AMS to safely and reasonably decontaminate the London Road interceptor (the sewer), or, if NRC's position is that such action has already been ordered, NRC action requiring AMS to actually complete the decontamination.

Since receipt of the March 3, 1993 petition, NRC has amended AMS' license such that one of the District's requests has already been partially granted, as set forth below. I have completed my evaluation of the remaining matters raised by the District and have determined that, for the reasons stated below, the other requests in the petition and its supplements should be denied.

II. BACKGROUND

NRC issued License No. 34-19089-01 to AMS on November 2, 1979. Picker Corporation had previously owned and operated the licensed operation, facilities, and equipment since 1959. From 1979 to mid-1991, the AMS license authorized the possession of 150,000 curies (5550 terabecquerels) of cobalt-60 in solid form for the purpose of manufacturing sealed sources for distribution to authorized recipients for use in teletherapy units (used at medical facilities for treatment of medical conditions). The AMS license currently limits possession to 150,000 curies (5550 terabecquerels) as solid metal and 135,000 curies (4995 terabecquerels) in sealed sources, for use in installing and servicing teletherapy units, and training; the current license does not authorize manufacture of sealed sources for distribution. The license also authorizes possession of 40,000 curies (1480 terabecquerels) of cesium-137 in sealed sources, and 4040 kilograms of plated depleted uranium shielding, incident to teletherapy and industrial radiography installation, maintenance, and service. The facility that houses the licensed material is located on London Road in Cleveland, Ohio.

The District is responsible for operating three wastewater treatment facilities in and around the Cleveland, Ohio metropolitan area. The District's SWTC has been operating since 1927 to remove grit and debris from wastewater that the

District services. This process involves incineration of sludge, transport of the residual ash in a slurry to settlement and evaporation ponds, and eventual transfer of the dried ash to landfills. The SWTC also incinerates sludge generated at other facilities, including the District's Easterly Plant, which services the area where AMS is located.

In April 1991, NRC identified cobalt-60 at the SWTC in ash piles coincidental to an aerial radiation survey of an unrelated site. In September 1991 and March 1992, at the request of NRC, Oak Ridge Institute for Science and Education (ORISE) performed surveys at the SWTC to determine the extent of the cobalt-60 contamination at the facility. The results of the ORISE surveys are reported in "Radiological Characterization Survey for Selected Outdoor Areas, Northeast Ohio Regional Sewer District, Southerly Wastewater Plant, Cleveland, Ohio," Final Report, August 1992 (hereafter referred to as "ORISE report"). The ORISE report indicated that there were elevated direct radiation readings that were caused by cobalt-60 contamination, with elevated concentrations in soil and sediment samples. Based on this ORISE report and information collected and examined by NRC Staff, NRC estimated that a total activity of 414 millicuries (15.3 gigabecquerels) of cobalt-60 existed at the SWTC in 1992.

Since the District needs to transfer the dried ash from the evaporation ponds to continue operations, NRC approved the site remediation strategy for ash removal, and had ORISE perform an independent survey to evaluate the radiological status of the remediated area. The District performed a radiological characterization of the facility to better determine the amount of cobalt-60 that is actually present on the SWTC site; the District's consultant estimated the quantity of cobalt-60 in the North Fill Area, as of 1993, to be about 443 millicuries (16.4 gigabecquerels).

As discussed below, NRC has evaluated the District's concerns and bases for its requests for NRC action. Although NRC has amended AMS' license to require remediation of the interceptor sewer line operated by the District in the vicinity of the connecting line from the AMS facility, which partially grants one of the District's requests, the District's remaining requests are denied for the reasons discussed below.

III. DISCUSSION

A. Timing and Source of Contamination Identified at the SWTC

In 1991, cobalt-60 was discovered in the North Fill Area.* The Staff's review of the history of the SWTC revealed that, after renovation of the incinerators

*Significant levels of cobalt-60 requiring remediation were discovered in the North Fill Area, in the existing In-Place Ash section of the South Fill Area, and in the northern section of the South Fill Area. Only the North Fill Area contamination can be dated with any degree of certainty, although AMS records indicate that 1989 was the last year AMS discharged cobalt-60 directly into the sanitary sewer system.

between 1975 and 1978, the incinerators came back on line in November 1978, and the current ponds were put into use for the first time. The ponds were then cleaned for the first time from December 1982 to March 1983. The District removed the ash from the evaporation ponds and placed it in the North Fill Area, which was then landscaped. This was the only time the North Fill Area was used for ash disposal. Accordingly, the cobalt-60 entered the District's system and was deposited at SWTC between late 1978 (when the ponds were first used) and December 1982 (when the ponds were first cleaned and the ash placed in the North Fill Area). See Memorandum for Carl J. Paperiello, former Deputy Regional Administrator, NRC Region III, from Loren J. Hueter, Radiation Specialist, Division of Nuclear Material Safety, NRC Region III, on the subject of "Report on Trip to General Chemical Corporation (Non-licensee), 5000 Warner Road, Cleveland, Ohio, and to Northeast Ohio Regional Sewer District, 6000 Canal Road, Cleveland, Ohio" (Docket No. 030-18276; License No. 34-17726-02) dated June 13, 1991. The Staff's conclusion as to when cobalt-60 contamination entered the sanitary sewer system is supported by the District's letter, dated September 13, 1994, which stated that the earliest possible date that the cobalt-60 could have been discharged into the sanitary sewer was not more than a week or two before the opening of lagoons in October 1978.

In an attempt to determine all possible contributors of cobalt-60 contamination to the SWTC, NRC conducted a file review of all licenses issued since 1975, active and terminated, for activities at facilities in the zip code areas serviced by the District. NRC contacted existing and previous licensees for additional information. The U.S. Department of Energy was also contacted to determine if any of its operations in the Cleveland metropolitan area could have contributed to the cobalt-60 contamination at the SWTC. Although other cobalt-60 users were found in the NRC's file search, it was concluded that no facility, other than AMS' facility at 1020 London Road, Cleveland, Ohio, was authorized to possess the quantities of unsealed cobalt-60 that could have contributed to the levels of cobalt-60 contamination found at the SWTC. Memorandum from Roy Caniano, Chief, Materials Inspection Branch, Division of Radiological Safety & Safeguards (DRSS), Region III, to William L. Axelson, Director, DRSS, dated November 7, 1994 (hereafter "Caniano Memo").

Given the information as to the timing of the disposals into the sewer system that caused the cobalt-60 contamination at the SWTC, the Staff included Picker, which previously used the facility under NRC license, in its review and inspection, although the District did not seek action against Picker. Current and former Picker employees, as noted in Inspection Report No. 030-16055/93003 (Section 3.C), issued November 7, 1994, stated that liquid radioactive waste was routinely discharged from the London Road facility. They stated, however, that the 1-curie (37-gigabecquerels) per year annual gross quantity disposal limit (10 C.F.R. § 20.303) was never exceeded during their respective tenures.

Based on the information gathered during the inspection, it is highly likely that Picker Corporation discharged cobalt-60 into the sanitary sewerage system every year that it operated the London Road facility, including the 1978 and 1979 time period of interest. As for AMS, its records indicate that a total of 209 millicuries (7.73 gigabecquerels) of unsealed cobalt-60 was disposed of into the sanitary sewerage system during the period 1980 to 1989. Caniano Memo at 3. AMS records indicate that 1989 was the last year that cobalt-60 was discharged directly into the sanitary sewerage system. NRC Inspection Report No. 030-16055/93003 (DRSS) at 7, issued November 7, 1994. AMS records also specifically list releases during the 1980-1982 time frame. Inspection Report No. 030-16055/93002 at 17, issued August 2, 1993. The information gathered by the Staff indicates, therefore, that cobalt-60 was likely released from the London Road facility during the 1979-1982 period of interest by both Picker and AMS.

AMS has recorded discharging cobalt-60 to the sanitary sewer system that eventually leads to SWTC, as described above. AMS records indicate, however, that it had been discharging cobalt-60 in accordance with the quantities and concentrations authorized by the then-applicable regulations and license. NRC's inspection and review of records have not revealed any documentation at AMS or other evidence that would indicate discharges in excess of authorized limits.

B. Request for NRC Action to Require AMS to Assume the Cost Resulting from Offsite Release of Cobalt-60

The Staff has carefully considered the action the District has requested and the bases stated by the District for its request. In addition, the Staff has evaluated the results of its inspections and all available information related to the District's requests. None of the available information, individually or taken together, demonstrates that AMS violated NRC regulatory limits or other requirements related to the discharge of cobalt-60 into the sanitary sewer system.

In a proceeding involving the decommissioning of the Yankee Nuclear Power Station near Rowe, Massachusetts, the Commission stated that it had no authority to grant an intervenor's request for compensation similar to the District's. *Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235 (1996). In the *Yankee* proceeding, the licensee had initiated substantial decommissioning of its facility through a "Component Removal Project" (CRP) under a new Commission policy interpreting the decommissioning rule (10 C.F.R. § 50.82) and had removed and disposed of many radioactive components through the CRP. The intervenors succeeded in challenging the Commission policy, which had allowed the licensee to initiate the CRP without an opportunity for a hearing. *CAN v. NRC*, 59 F.3d 294 (1st Cir. 1995). As relief for the failure to offer an opportunity for a hearing, and based on their assertion that

the CRP had caused workers and the public to receive doses far above those as low as reasonably achievable, the intervenors requested the Commission to require the licensee to establish a fund for the treatment of cancers caused by the doses resulting from the CRP. *Yankee*, CLI-96-7, 43 NRC at 268. In rejecting the intervenors' arguments, the Commission held that "no statute or regulation grants the Commission authority to require the Licensee to pay (in effect) compensatory damages to private individuals." *Id.* at 269.

The District's request for compensation from AMS for costs resulting from offsite releases of cobalt-60 from the London Road facility is not materially different from the *Yankee* intervenors' request for compensation. No statute authorizes the NRC to require any licensee to pay such compensatory damages, especially in a case in which the releases that resulted in the third party's damages were within applicable NRC limits.

The District, in addition to filing its petition with NRC, instituted a court action against AMS and other defendants for tort remedies, including property damage and remediation costs, resulting from the discharge of cobalt-60 into the District's system. The action, which was pending before the United States District Court for the Northern District of Ohio, Eastern Division (Case No. 1:94 CV 2555), has been settled. Letter dated January 2, 1997, from L.K. English, Esq., Northeast Ohio Regional Sewer District, to J. Madera, Division of Nuclear Material Safeguards, NRC. A court of competent jurisdiction, and not NRC, is the proper forum for the District to seek compensatory damages from AMS. Accordingly, the District's request for NRC action to require AMS to assume the costs resulting from the release of cobalt-60 is denied.

C. Request to Require AMS to Decontaminate the Sewer Connecting Its London Road Facility with the Public Sewer at London Road and Continue Downstream to the Extent AMS/NRC Sampling Indicates Is Necessary

By letter dated April 29, 1996, the District supplemented its original petition with a request that AMS be required to "safely and reasonably" decontaminate the London Road interceptor. In addition, the District requested that NRC take action to have AMS complete the decontamination of the interceptor if NRC believed that it had already ordered AMS to take action to decontaminate the interceptor. The indicated sewer connection that was identified as having excessive exposure rates is on AMS property. NRC did issue a Notice of Violation (NOV) for AMS' violation of 10 C.F.R. § 20.105, in that the exposure rates in the accessible sewer line on the AMS facility were excessive for an unrestricted area. NOV issued to AMS, License No. 34-19089-01, dated May 5, 1988, resulting from a special safety inspection conducted on April 13, 1988 (NRC Inspection Report No. 16055/88001 (DRSS)). However, the

manhole controlling access to the sewer connection was designated a restricted area; the sewer cover on the AMS property was secured with a lock and bar; and the sewer connection area was partially decontaminated, reducing the contamination and exposure-rate levels. Letter from T.J. Hebert, Chairman, Radioisotope Committee, AMS, to R.E. Burgin, Senior Radiation Specialist, NRC Region III, dated May 23, 1988. These facts were confirmed by Oak Ridge Associated Universities, contracted by NRC to perform a radiological survey to determine the then-current conditions at the AMS facility. See Oak Ridge Associated Universities Report, "Radiation Survey of the Advanced Medical Systems, Inc., London Road Facility, Cleveland, Ohio," Final Report, at 20 (April 1989). The exposure rates are no longer considered excessive as a result of the decontamination performed by AMS and the designation of the manhole as a restricted area. Moreover, in 1995, AMS permanently sealed the lateral from the old manhole to the sewer line. AMS also removed most of the original foundation underdrain system and replaced it with a new, clean system. AMS is currently required to test the groundwater pumped from the new foundation underdrain system, to ensure compliance with section 20.2003.

The NRC has taken action by issuing Amendment No. 32 to AMS' license, dated March 17, 1995, in which the NRC, through Condition 19.F, required AMS to remediate the London Road interceptor in the vicinity of the abandoned lateral, as described in an AMS letter proposing action to remediate contaminated piping. See "Action Plan for the London Road Facility" at 2 (Jan. 27, 1996). License Condition 19 required that remediation of the interceptor be completed within 90 days (i.e., by June 15, 1995). In Amendment No. 35 to AMS' license, dated June 16, 1995, NRC required AMS to initiate remediation activities no later than July 8, 1995, and to notify NRC no later than July 14, 1995, to confirm initiation of the remediation of the interceptor. Amendment No. 35, however, deleted the June 15, 1995 date for completion of remediation of the interceptor imposed by Amendment No. 32.

By a letter dated July 12, 1995, AMS informed NRC that it would not start the remediation of the interceptor until July 29, 1995, and did not provide an estimated completion date for the remediation, as AMS further informed NRC that it needed the District's approval to access the interceptor. Letter from R. Meschter, Radiation Safety Officer (RSO), AMS, to J. Caldwell, NRC, dated July 12, 1995. By a letter dated July 19, 1995, AMS informed NRC that, for the same reasons given in the July 12, 1995 letter, it would not initiate remediation until August 11, 1995. Letter from R. Meschter, RSO, AMS, to J. Caldwell, NRC, dated July 19, 1995. At that time, AMS and the District still had not agreed on arrangements for entry and evaluation of the interceptor.

In a letter dated January 2, 1997, from L.K. English, Esq., Northeast Ohio Regional Sewer District, to J. Madera, NRC, the District forwarded a copy of a settlement agreement between the District and AMS regarding their court

litigation. The settlement indicates that AMS agreed, *inter alia*, to pay the District a fixed sum, and the District agreed to allow reconnection of the AMS facility to the London Road interceptor after AMS' taking certain actions pertaining to conditions of the facility, and to design and construction of the connection. The part of the agreement concerning reconnection provides an alternative to use the present manhole located in London Road, provided that the plans include decontamination of the interceptor, at AMS' cost, before such use. The agreement specifies conditions and procedures under which AMS may plan to use the present manhole in the interceptor. In a meeting with NRC and AMS on February 10, 1997, AMS indicated that it was its intention to reconnect. Official Transcript of Proceedings: "Public Meeting with Advanced Medical Systems, Inc.," at 50-51 (Feb. 10, 1997). AMS stated that it will probably take from 9 months to a year and a half for reconnection to actually happen. *Id.* at 51. In summary, insofar as Amendments No. 32 and 35 require AMS to remediate the sewer connecting its London Road facility with the public sewer, this request of the District has been partially granted. Although access to the interceptor is now controlled, License Condition 19.F requires AMS to remediate the interceptor. The Staff intends to pursue this matter in the near future. It is the Staff's intent that the access concerns be resolved promptly, so that remediation may begin and be completed as soon as practical.

D. Other Issues Raised in Supplements to Petition

By letters dated September 13, 1994, and October 13, 1994, the District supplemented its original petition with a request that NRC commence an appropriate enforcement action against AMS for the maintenance of grossly inaccurate records of disposal of radioactive material from 1978 to 1993, in violation of section 20.401(b)(3) (in effect through December 31, 1993). The District also asserted that AMS had disposed of cobalt-60 that was not "readily soluble or dispersible in water," in violation of section 20.303 (in effect through December 31, 1993), and had more recently discharged cobalt-60 which was not "readily soluble or dispersible biological material," in violation of section 20.2003 (in effect on January 1, 1994, and thereafter).

The Staff has conducted numerous recent inspections at the London Road facility to address the District's concerns over cobalt-60 discharges into the sanitary sewerage system. On March 15, 1995, NRC issued a Notice of Violation to AMS for failures to: (1) evaluate the quantity of cobalt-60 released to the sewer system resulting from facility floods and certain decontamination activities; and (2) remove nonsuspendible solids by the use of a cloth filter, as required by AMS' license conditions. The background relating to unmonitored releases resulting from facility floods and certain decontamination activities is set forth below.

The information as to when the unmonitored releases occurred came from current and former Picker and AMS employees and identified several occasions in the late 1960s and the mid- to late 1980s when the basement was flooded, resulting in backflow into the sewer system. The available information indicated that not all of these occurrences were evaluated to identify the amount of radioactivity that may have been released. Inspection Report No. 030-16055/93003, at 16-19. Based on the extensive information provided by the interviewees, the Staff concluded that it was unlikely that the cumulative total quantity of cobalt-60 released during these unmonitored releases exceeded a few hundred millicuries. *Id.*

As to the filtering of the wastewater pumped from holding tanks in the Waste Hold-Up Tank room, the information gathered from the interviewees strongly indicated that the filter was not always in place from the mid-1970s through the mid-1980s, thus raising the potential for cobalt-60 pellets to have been discharged through this route into the sewer system. *Id.* at 14.

The NRC has already taken enforcement action for the failures to: (1) evaluate and report certain releases into the sewer system as a result of facility floods or decontamination activities that likely included cobalt-60; and (2) ensure that wastewater in the holdup tanks was passed through filters that should have captured any cobalt-60 pellets before the release of the water to the sewer system. The Staff does not believe that further enforcement action for the matters identified in the September 1994 supplement is warranted.

Regarding the October 1994 supplement's request for enforcement action for violation of section 20.2003, the Staff has not found evidence, based on NRC interviews and review of records, that AMS intentionally disposed of cobalt-60 into the facility's drains leading to the District's sanitary sewerage system since May 1989. The AMS records contain no discharge log entries after this date. Furthermore, AMS has not generated liquid radioactive waste from manufacturing operations in several years, and has no plans to do so in the future, because of termination of source manufacturing operations. See Inspection Report No. 030-16055/93002. However, both the District and the Staff performed sampling (post-January 1, 1994, the effective date of revision of 10 C.F.R. Part 20) that identified cobalt-60 at the point of discharge of the sanitary sewerage piping from the London Road facility into the District's sewer line. See the District's supplement to its petition, dated October 13, 1994, and Inspection Report No. 030-16055/94003, issued on December 6, 1994. The presence of the cobalt-60 appears to be a result of plate-out of cobalt-60 onto the walls of the piping leading from the London Road facility. The Staff had characterized the results of its sampling as indicating an apparent violation of section 20.2003. *Id.*

The sampling performed by the District and subsequent sampling performed by the Staff in early 1995 indicated that some or all the cobalt-60 detected might

be "soluble," as that term is defined in NRC Information Notice No. 94-07, dated January 28, 1994. The uncertainty as to the solubility of the cobalt-60 prompted the Staff to begin preparations for a solubility analysis of the sample taken on August 17, 1994. In accordance with Region III policy, those samples had been transferred back to the District, on whose property the samples had been taken. Because of further analyses the District had performed on the samples, the samples no longer existed in their original form; therefore, further solubility analyses could not be performed. Further representative samples of the water at this point in the waste stream could not be taken because of the District's plugging of the pipe. In view of the inability of the Staff to determine that the cobalt-60 in the sampled water was, in fact, insoluble, there was an insufficient basis to cite AMS for a violation of section 20.2003. Furthermore, there is not now a significant potential for discharge of cobalt-60 from the London Road facility to the District's system because: (1) old piping connecting the facility to the District's lines has been plugged; (2) the District has not permitted AMS to connect new clean piping installed by AMS to the District's lines; and (3) AMS collects and treats all water used on the site and holds it in tanks before it is determined not to contain insoluble cobalt-60.

The Staff believes that the vast majority of cobalt-60 inventory and activity discharged into the District's sanitary sewerage system was dispersible. It can be expected that a small amount of readily dispersible material would plate-out onto the sewer system pipes over the long history of cobalt-60 discharges by Picker and AMS. Staff concludes that the fact that a small amount of cobalt-60 built up over time in sewer pipes leading from the AMS facility, by itself, does not support the District's assertion that a discharge in violation of section 20.303 or 20.2003 occurred.

IV. CONCLUSION

For the reasons discussed above, no basis exists for taking any action, in addition to the action described above, in response to the requests in the petition and its supplements. Accordingly, no further action pursuant to section 2.206 is being taken in this matter.

As provided by 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. The Decision will become the final action of the Commission twenty-five (25) days after

issuance unless the Commission on its own motion institutes 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 13th day of June 1997.**

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)

June 16, 1997

By a petition dated April 1, 1997, Berkeley Township Environmental Commission (Petitioners) requested that the NRC direct Oyster Creek Nuclear Generating Station (OCNGS or Licensee) to shut down its operations during a planned transfer of fuel from wet to dry storage. The request was considered as a petition submitted pursuant to 10 C.F.R. § 2.206.

In a Director's Decision issued on June 16, 1997, the Director of Nuclear Reactor Regulation dismissed Petitioners' request as premature. The Director concluded that because OCNGS would first have to submit a request for a license amendment to perform the action in question, which it had not yet done and on which the Petitioners would have an opportunity to comment, there was no basis for the Commission to take the requested action at this time.

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

I. INTRODUCTION

By a petition submitted pursuant to 10 C.F.R. § 2.206 and dated April 1, 1997 (petition), Berkeley Township Environmental Commission (Petitioners) requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to Oyster Creek Nuclear Generating Station (OCNGS) operated by GPU Nuclear Corporation (GPU or Licensee). The Petitioners requested that the NRC

direct the Licensee to shut down OCNGS during an upcoming planned transfer of fuel from wet to dry storage.

The Petitioners based their request on the following assertions: (1) the load transfer path for the 100-ton fuel transfer casks passes over the reactor's containment mechanism and other safety-related equipment; (2) NRC Bulletin 96-02, dated April 11, 1996, states that a dropped cask could damage both isolation condensers and the torus, creating the possibility of an unisolable leak, which in industry jargon describes a situation perilously close to a nuclear meltdown; (3) the operating record of GPU demonstrates it is capable of human error, including dropping heavy loads; (4) Berkeley Township could not be successfully evacuated in the event of a serious nuclear accident at OCNGS; and (5) the safer, simpler alternative of turning off the reactor while lifting 100-ton loads over the containment can be easily implemented.

For the reasons stated below, I have dismissed the Petitioners' request as premature.

II. DISCUSSION

The Petitioners have requested that the NRC take action against the Licensee on a matter involving the potential transfer of spent fuel during plant operation. However, this is an activity for which the Licensee has not yet requested authorization from the Commission. At a public meeting on February 29, 1996, the NRC informed GPU that it would have to obtain a license amendment to move fuel from wet to dry storage, using the facility's existing crane, while the reactor is operating at power. The Staff had reviewed the Licensee's safety evaluation of its crane, including the crane upgrades, and concluded that all safety concerns had been addressed and resolved and that the planned movement of spent fuel to the dry storage facility during plant operation would be safe and in accordance with all license requirements. However, the NRC also determined that because the possibility of an unreviewed safety question existed before GPU made modifications to upgrade its reactor building crane, GPU would have to submit a request for a license amendment for the proposed cask movement. If GPU submits such an amendment request to the NRC, pursuant to 10 C.F.R. § 50.91,¹ it will be published in the *Federal Register* for public comment, and an opportunity for a public hearing will be provided. The Petitioners and other interested members of the public then would have the opportunity to express their concerns about the amendment. As noted above, the Licensee cannot

¹Section 50.91 specifies the Commission procedures to be followed when it receives an application requesting an amendment to an operating license, including procedures for consulting the state in which the facility is located and procedures for notifying the public of the license amendment and the opportunity for a hearing.

transfer the fuel while operating with its current crane configuration without being issued a license amendment.²

III. CONCLUSION

The NRC Staff has reviewed the Petitioners' request that GPU shut down its reactor during its transfer of fuel from wet to dry storage. The Licensee does not now have a request before the Commission to amend its license to allow such a transfer. As a result, before any Commission action could even be contemplated, the Licensee would have to make such a request pursuant to NRC regulations, with the aforementioned opportunities for public participation in the resolution of any such request. For this reason, the petition is dismissed as premature.

A copy of this Director's Decision will be filed with the Secretary of the Commission for the Commission to review as stated in 10 C.F.R. § 2.206(c). This Decision will become the final action of the Commission 25 days after issuance, unless the Commission, on its own motion, institutes 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 16th day of June 1997.

²The Licensee is currently considering various options for moving the spent fuel from wet to dry storage, such as requesting a license amendment based on already completed upgrades to the reactor building crane, transferring the spent fuel when the reactor is shut down, and further upgrading the reactor building crane to meet the criteria for a single-failure-proof crane in which case an amendment to transfer fuel from wet to dry storage may not be required. The Commission has not required license amendments for facilities handling heavy loads that employ a crane meeting the specifications and design criteria in NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." However, NRC technical staff will evaluate any option selected to ensure that all safety concerns are adequately addressed and documented.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Malcolm R. Knapp, Acting Director

In the Matter of

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
SIERRA NUCLEAR CORPORATION	Docket No. 72-1007 June 18, 1997

By a petition filed on October 18, 1996, the organizations Don't Waste Michigan and Lake Michigan Federation requested, pursuant to 10 C.F.R. § 2.206, that the NRC prohibit the loading of Ventilated Storage Casks until an independent, third-party review of the design has been performed to address their concerns and the certificate of compliance, safety analysis report, and safety evaluation report for the casks have been amended to contain operating controls and limits to prevent hazardous conditions. The Director of the Office of Nuclear Material Safety and Safeguards, in the following Decision, denies the Petitioners' request.

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

I. INTRODUCTION

On October 18, 1996, Don't Waste Michigan and the Lake Michigan Federation (Petitioners) filed a petition pursuant to section 2.206 of Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206) requesting that the U.S. Nuclear Regulatory Commission take the following action:

Prohibit loading of Ventilated Storage Casks (VSC-24s) until the certificate of compliance (COC), the safety analysis report (SAR), and the safety evaluation report (SER) are amended following an independent, third-party review of the VSC-24 design, to address concerns raised by the Petitioners' engineering consultant, Dr. Rudolf Hausler.

The petition has been referred to me pursuant to section 2.206. By letter dated December 10, 1996, to Dr. Mary Sinclair and Ms. Eleanor Roemer, on behalf of the Petitioners, NRC 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 January 13, 1997 (62 Fed. Reg. 1783).

On the basis of the NRC Staff's evaluation of the issues and for the reasons given below, I have determined that the Petitioners' request should be denied.

II. BACKGROUND

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 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. On June 3, 1996, the NRC issued confirmatory action letters (CALs) to those licensees using or planning to use VSC-24s for dry storage of spent nuclear fuel, i.e., Licensees for Point Beach Nuclear Plant, Palisades Nuclear Generating Plant, and Arkansas Nuclear One (ANO). The CAL issued to the Licensee for ANO was supplemented on June 21, 1996, and the CALs issued to the Licensees for Point Beach and Palisades were supplemented on June 27, 1996. The CALs, as supplemented, documented the Licensees' commitments not to load or unload a VSC-24 without resolution of material compatibility issues identified in a forthcoming generic communication and subsequent NRC confirmation of corrective actions taken by the Licensees. The generic communication was issued on July 5, 1996, in the form of NRC Bulletin 96-04, "Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks." NRC

Bulletin 96-04 notified addressees about the potential for adverse chemical, galvanic, or other reactions among the materials of a spent fuel storage or transportation cask, its contents, and the environments the cask may encounter during use. The actions requested in Bulletin 96-04 included reviewing the cask materials for potential adverse reactions, evaluating the short-term and long-term effects of any identified reactions, and determining the adequacy of cask operating procedures to minimize the consequences of any identified reactions. The NRC Staff has acknowledged that the event demonstrated that the cask vendor's (Sierra Nuclear Corporation) SAR for the VSC-24 and related NRC review, as documented in the NRC Staff's SER, did not adequately address the use of a zinc-based coating and its reaction with the acidic water in spent fuel pools.

In response to Bulletin 96-04 and to subsequent NRC Staff inquiries, the Licensees for ANO, Point Beach, and Palisades submitted to the NRC evaluations of possible material interactions and the effects of such interactions on cask performance and operation. The Licensees also submitted information on the operating controls and limits that were implemented to prevent hazardous conditions that may result from adverse material interactions. The operating controls and limits included 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 evaluated the responses submitted by the Licensee for ANO. As documented in the Staff's safety evaluation dated December 3, 1996, the Staff determined that the Licensee's submittals provided the necessary level of confidence that the VSC-24 can be used to safely store spent fuel over the 20-year period of the certificate. The Staff also determined that the operating controls and limits proposed by the Licensee 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.

The NRC Staff also evaluated the responses submitted by the Licensees for Point Beach and Palisades. As documented in the Staff's safety evaluations dated, respectively, April 8, 1997, and June 12, 1997, the Staff determined that the Licensees' evaluations and proposed operating controls and limits are acceptable and satisfy regulatory requirements. However, the CALs placed on Point Beach and Palisades still remain in place until an NRC inspection is performed to verify that the Licensees' corrective actions are properly implemented.

III. DISCUSSION

The petition requests an NRC order to users of VSC-24s not to load additional casks until: (1) the COC, SAR, and SER are amended to contain operating controls and limits to prevent hazardous conditions; (2) an independent third-party review team has examined the safety issues raised by the Petitioners; (3) the potential impacts of all material aspects of the casks have been fully assessed; (4) there is experimental verification of temperature calculations and heat transfer assessments and other design assumptions; and (5) the safety of the material coatings on components and structures has been justified.

Item 1: Prohibit Loading of VSC-24s Pending Amendment of Documents

As noted in the NRC letter to the Petitioners on December 10, 1996, the Petitioners' request to amend the COC, SAR, and SER is similar to a request made by the Citizen's Utility Board (CUB) in a petition dated September 30, 1996. The NRC Staff denied the CUB petition on April 17, 1997, for the reasons that are identical to the reasons stated here in denying the first part of the Petitioners' request.

The circumstances set forth above made clear that, following the event at Point Beach, the NRC Staff recognized that additional evaluation of potential material interactions was warranted for all spent fuel transportation and storage casks. In regard to the VSC-24, 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-24s pending completion of the NRC Staff's review of the responses to 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-24s.

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 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 NRC Staff has documented the process, information, and results of its review of the Licensees' responses to Bulletin 96-04 for use of the VSC-24 at ANO, Point Beach, and Palisades in safety evaluations available for public review.

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, the NRC Staff agrees with the Petitioners that it would be beneficial if the SAR and other licensing-basis documents accurately describe the identified chemical reaction and the associated operating controls and limits. The NRC Staff is currently reviewing a proposed amendment to the SAR and COC for the VSC-24 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 documents. In addition, the NRC Staff is processing a petition for rulemaking, PRM-72-3, that may lead to additional updating of independent spent fuel storage installation 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 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 Petitioners as a precondition for resuming cask operations at the facilities using VSC-24s. Therefore, there would be no regulatory basis for granting the first part of the petition to require amendment of the COC, SAR, or SER before further loading of VSC-24s.

Item 2: Prohibit Loading of VSC-24s Pending Independent, Third-Party Review

Petitioners request the NRC to prohibit loading of VSC-24s until the COC, SAR, and SER are amended following an independent, third-party review to address concerns raised by the Petitioners. The NRC Staff performed a review of the VSC-24 design prior to certification in 1993. As a result of the review, the Staff determined that the design and operation of the cask system is in compliance with 10 C.F.R. Part 72. The Staff also concluded, with a high degree of assurance, that the VSC-24 will safely store spent fuel over the 20-year period of the certificate. Notwithstanding the Staff's review and determination in 1993, the Petitioners are claiming that a new, independent review is needed before further VSC-24s are loaded.

While the event at Point Beach revealed the need for additional evaluation by licensees and NRC of potential material interactions in the VSC-24 (and other transportation and storage casks), the actions already taken, in the Staff's judgment, provide an adequate response. In particular, Bulletin 96-04 was issued to request additional information from licensees using the VSC-24 on material interactions and compatibility in the VSC-24 and on the corrective actions implemented. The NRC Staff then received and reviewed the responses

submitted by the Licensees for ANO, Point Beach, and Palisades. The Staff's reviews (as well as the Licensees') have been exhaustive and were performed by an interdisciplinary team of engineers knowledgeable in materials, corrosion, metallurgy, chemistry, structural engineering, heat transfer, nuclear engineering, and other technical fields needed to perform the review. The results of the Staff's reviews, including the necessary corrective actions, are documented and justified in the Staff's December 3, 1996, April 8, 1997, and June 12, 1997 safety evaluations. These corrective actions include: cleanliness checks before placing the cask in the spent fuel pool, venting and monitoring of the air space beneath the VSC-24 shield lid during welding or cutting activities, discontinuing welding or cutting should the hydrogen concentration exceed 0.4% by volume (10% of the minimum amount necessary for a combustible concentration), and sampling the boron concentration in the spent fuel pool and multiassembly sealed basket (MSB) water. While the Staff agreed that the corrective actions were necessary to prevent hazardous conditions during the loading and unloading of VSC-24s, the information submitted by the Petitioners does not raise any new issues or provide any reason for the Staff to question its conclusion that the VSC-24 will safely store spent fuel over the 20-year period of the certificate.

In reaching this conclusion, the NRC Staff evaluated the specific concerns raised by the Petitioners related to the design of the VSC-24. The Staff believes that these concerns have already been addressed by the recent evaluations submitted in response to Bulletin 96-04, by information submitted to NRC to support the certification of the VSC-24 design in 1993, or by other information submitted in support of NRC review and inspection activities. Each of the Petitioners' specific concerns is addressed below.

(i) The Petitioners claim that the cask design allows for fuel elements to be in contact with the zinc primer, creating a galvanic couple that will accelerate the corrosion of the zinc. The NRC Staff considered galvanic effects between the Zircaloy fuel rods and the Carbo Zinc 11 coating. The Staff agrees that a galvanic effect would increase the corrosion rate of the zinc, with a corresponding increase in the hydrogen gas generation rate, as the zinc in the Carbo Zinc 11 coating is polarized to a more active potential. However, in the VSC-24 design, several factors reduce the amount of zinc polarization such that there would not be a significant increase in hydrogen generation. One factor is the contact resistances between the stainless steel fuel assembly end-fittings and the Zircaloy fuel rods and between the end-fittings and the Carbo Zinc 11 paint. Another factor is the geometry of the VSC-24 and the fuel assemblies. The fuel assemblies are placed in fuel storage sleeves with a clearance of approximately 0.1 inch to 0.5 inch between the sides of the fuel assembly and the sleeves. This clearance and the physical design of the fuel assemblies create shielding between the fuel rod surfaces and the Carbo Zinc 11 coating. This shielding effectively reduces the galvanic action between the Zircaloy fuel rods and the Carbo Zinc

11 coating. The Zircaloy fuel rods could contact the Carbo Zinc coated sleeves if the fuel assembly is not centered in the storage sleeves or if the fuel rods are bowed. However, the shielding effect and small Carbo Zinc/Zircaloy contact area would still prevent significant galvanic action. Hydrogen concentration measurements made at Point Beach and the hydrogen monitoring performed at ANO during loading of a VSC-24 in December 1996 (NRC Inspection Report Nos. 50-313/96-25 and 72-13/96-02) support the conclusion that significant galvanic action between the Zircaloy and zinc coating, and hence, increased hydrogen generation, is not occurring in the VSC-24. In addition, even if there was an increase in hydrogen generation because of the galvanic action, the Staff has determined that the controls implemented by the Licensees for ANO and Point Beach would prevent accumulation of a combustible concentration of hydrogen and its ignition. The Staff will also review and verify the adequacy of the controls implemented by the Licensee for Palisades.

(ii) The Petitioners claim that there were numerous discrepancies in the responses to Bulletin 96-04. As noted, the NRC Staff completed its review of responses for ANO, Point Beach, and Palisades. The Staff found these responses to be acceptable and found no discrepancies of concern. There were minor differences in the operating controls implemented at the three facilities. However, the Staff reviewed these controls and concluded that all three sets of controls are adequate to preclude hazardous conditions during cask operation.

(iii) The Petitioners claim that the epoxy coating applied to the exterior of the multiassembly sealed basket (MSB) could not withstand the temperatures developed during long-term storage. Technical data on the type of epoxy coating used on the MSB were provided by the Licensees in their responses to Bulletin 96-04. The data show that the epoxy is temperature-resistant up to 350°F. The SAR for the VSC-24 (which the Staff reviewed and accepted prior to certification in 1993) shows that under normal or off-normal storage conditions, the temperature of the MSB exterior will not exceed 300°F, for the maximum allowable heat load of 24 kW and, therefore, will not degrade the epoxy.

(iv) The Petitioners claim that the low-temperature specification in the COC for moving the VSC-24 MSB was not properly translated to the MSB shell material compositions. Low-temperature embrittlement of the MSB shell material was evaluated by the NRC Staff during its safety review before certification of the VSC-24. The composition of the MSB shell material (SA516, Grade 70 carbon steel) is specified in the American Society for Mechanical Engineers, Boiler & Pressure Vessel Code, Section II, SA-516, "Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service." The impact testing requirements for the MSB material are found in American Society for Testing and Materials Specification A370 (ASTM A370), "Methods and Definitions for Mechanical Testing of Steel Products." As specified in the COC, SER, and SAR, each MSB shell material must be

shown, during fabrication, by Charpy test per ASTM A370, to have 15 ft-lbs of absorbed energy at -50°F . Further, movement of the MSB must occur only at ambient temperatures of 0°F or above to avoid potential brittle fracture of the MSB material.¹ The NRC Staff considers the 50°F temperature difference to provide sufficient margin because it places the MSB material at a temperature that is significantly above the temperature where brittle fracture could occur. It should also be noted that the temperature of the MSB shell itself would actually be substantially higher than the ambient temperature (e.g., 20°F for 25-year-old fuel), thus providing an even higher margin. In addition, it is highly unlikely that any MSB movement activity would take place at temperatures below 0°F .

(v) The Petitioners claim that zinc-steel interaction at 800°F to 1000°F and possible steel embrittlement over a 20-year period were not considered. Zinc-steel interaction at the 800°F to 1000°F temperature range was not considered and is not a concern because, as documented in the VSC-24 SAR, temperatures in the MSB will not reach 800°F during storage. Maximum temperatures would be 688°F under normal conditions and 708°F under off-normal conditions, for the maximum allowable heat load of 24 kW. Furthermore, over the storage period, the temperatures within the MSB will continue to decrease as the heat load decreases due to the decay of the spent fuel.

(vi) The Petitioners claim that the effect of molten zinc on Zircaloy has not been verified experimentally. The NRC Staff evaluated the durability and behavior of the zinc coating under the range of storage temperatures. The presence of molten zinc is not expected under the storage temperatures and conditions; thus the effect of molten zinc on Zircaloy is not a concern. However, as documented in the Staff's safety evaluations for ANO (dated December 3, 1996), Point Beach (dated April 8, 1997), and Palisades (dated June 12, 1997), the Staff did evaluate the potential interaction between zinc vapor and Zircaloy and the effect of this interaction. Based on the information provided in the responses to Bulletin 96-04, the Staff concluded that the potential interaction between zinc vapor and Zircaloy presented no immediate or long-term safety concern for the spent fuel stored in the VSC-24.

(vii) The Petitioners claim that the vacuum-drying process does not seem to have been experimentally verified. Vacuum drying is a well-established, widely used method for removing moisture from spent fuel storage and transportation casks. The process used for the VSC-24 is a common process, which the NRC Staff evaluated and determined to be acceptable during the safety review before

¹ At Palisades, the Licensee has administratively set a minimum ambient temperature of 10°F for moving the first four MSBs (CMSB-01 through 04) to be loaded because the shell material for these MSBs does not have 15 ft-lbs of absorbed energy at -50°F . Rather, these MSBs have 15 ft-lbs of absorbed energy at -40°F . Thus, to retain the 50°F temperature margin, the Licensee has restricted movement of these four MSBs to an ambient temperature of 10°F or above. The NRC Staff has reviewed and approved the Licensee's administrative limit, as documented in NRC safety evaluation dated September 26, 1995.

certification in 1993. In the Staff's judgment, experimental testing to verify a well-established process is unnecessary.

(viii) The Petitioners claim that the thermal analyses for the VSC-24 have not been experimentally verified. The thermal analyses for the VSC-24 contained conservative key assumptions, including a total heat generation of 1 kW per assembly (a total of 24 kW per cask). This assumption is conservative because it is highly unlikely that each assembly loaded in the cask will generate 1 kW of heat. In addition, the assembly and total cask heat loads will continually decrease over time as the spent fuel decays. In light of the conservatisms in the thermal analyses, the Staff does not see the need for requiring experimental verification of the VSC-24 thermal analyses. Nevertheless, the COC requires that a thermal test be performed on the first VSC-24 to be loaded. The purpose of the test is to measure the heat removal performance of the VSC-24 system. The Licensee for Palisades performed such a test and summarized its results in a letter to NRC dated June 10, 1993. The temperatures measured during the test were lower than the predicted temperatures. The results thus indicate that the VSC-24 performs its intended heat removal function. The thermal test at Palisades was performed with a 12-kW heat load. To date, no VSC-24s have been loaded with greater than 12-kW heat load. As required by the COC, the thermal test must be performed for the first cask to use any higher heat loads, up to 24 kW.

The NRC Staff believes, based on the foregoing, that an independent, third-party review is not warranted by the Petitioners' specific concerns. However, NRC review activities relating to the VSC-24 will nonetheless continue. In particular, NRC inspection activities at the facilities operated by the Licensees, the VSC-24 vendor, and the VSC-24 fabricators may lead to additional reviews of the VSC-24. In addition, the Staff is currently reviewing a proposed amendment, submitted by the VSC-24 vendor, to the SAR and COC for the VSC-24 design. This review will be performed in accordance with the Staff's "Standard Review Plan for Dry Cask Storage Systems" (NUREG-1536) to ensure the thoroughness, quality, and consistency of the review. Where relevant, recent operational, technical, and safety issues related to the VSC-24 design will be considered by the Staff in this review.²

In addition, it is my judgment that the NRC Staff is fully capable of fulfilling the responsibility for reviewing, approving, and certifying dry cask storage systems to be used under 10 C.F.R. Part 72 which, by law, belongs to the NRC. In conducting its review, the NRC Staff must have reasonable assurance that the cask system will safely store spent fuel over the period of the certificate.

² Recent concerns relating to the MSB closure welds, as documented in NRC Inspection Report No. 72-1007/97-204, dated April 15, 1997, may result in further evaluations of the VSC-24 design and if necessary, appropriate regulatory action to ensure continued safe use of the VSC-24.

Further, the Staff will assign the necessary resources and expertise to perform such reviews. When the NRC Staff lacks either the resources or expertise to perform all or portions of the review in-house, the NRC may, and does, supplement its own ranks by using outside specialists.

Item 3: Prohibit Loading of VSC-24s Pending Assessment of Cask Materials

Petitioners request the NRC to prohibit loading of VSC-24s until the potential impacts of all material aspects of the casks have been fully assessed. As previously stated, Bulletin 96-04 was issued to request information on material interactions and compatibility in spent fuel storage and transportation casks. In response to this request, the Licensees for ANO, Point Beach, and Palisades submitted evaluations on possible material interactions in the VSC-24 and the effects of such interactions on cask performance and operation. The only significant material interaction identified was between the zinc-based coating and the borated spent fuel pool water. As previously discussed, the operating controls and limits put in place by the Licensees provide an adequate level of confidence to prevent the adverse effects of this interaction (generation and possible ignition of hydrogen gas and possible depletion of boron in the water). The Staff reviewed these evaluations and, based on the information provided, concluded that none of the identified material interactions would adversely affect the VSC-24's ability to safely store spent fuel over the 20-year period of the certificate. The results of the Staff's reviews are documented in the Staff's December 3, 1996, April 8, 1997, and June 12, 1997 safety evaluations for ANO, Point Beach, and Palisades, respectively.

Item 4: Prohibit Loading of VSC-24s Pending Experimental Verification of Thermal and Other Design Assumptions

Petitioners request the NRC to prohibit loading of VSC-24s until there is experimental verification of temperature calculations and heat transfer assessments and other design assumptions. The thermal and other engineering and design analyses for the VSC-24 contained conservative key assumptions which are discussed in the SAR and SER. In addition, the acceptance criteria for these analyses have margins of safety that the Staff considers to be sufficient. In light of the conservatism and safety margins in the thermal and other analyses, the Staff does not see the need for requiring experimental verification of the thermal and other design assumptions used in evaluating the VSC-24.

Item 5: Prohibit Loading of VSC-24s Pending Assessment of Material Coatings

Petitioners request the NRC to prohibit loading of VSC-24s until the safety of the material coatings on components and structures has been justified. As discussed above, material interactions within the VSC-24 and their effect on cask operations and performance were evaluated by the Licensees in response to Bulletin 96-04 and reviewed by the Staff. Specifically, the Licensees evaluated, and the Staff reviewed, the use of the zinc-based coating, its reaction with borated water and other cask environments, and the effect of the reaction or reaction products on cask operations and on the performance of the various cask components and structures. The Staff concluded that use of existing VSC-24s with the zinc-based coating is acceptable in light of the operating controls and limits for preventing hazardous conditions that must be properly implemented by Licensees during cask loading and unloading. Based on the information provided, the Staff also concluded that neither the coating itself, nor its reaction with borated water or other cask environments, would have an adverse effect on the performance of the cask components or structures during the period of spent fuel storage.

IV. CONCLUSION

The Petitioners requested that the NRC prohibit loading of VSC-24s until the COC, SAR, and SER are amended to contain operating controls and limits to prevent hazardous conditions. After reviewing each of the Petitioners' claims, I conclude that, for the reasons discussed above, no adequate basis exists for granting the Petitioners' request to prohibit Licensees' use of the VSC-24 for dry cask storage of spent nuclear fuel at Palisades, Point Beach, or ANO pending: (1) revision of the SAR, SER, and COC for the VSC-24 to contain operating controls and limits to prevent hazardous conditions; (2) an independent third-party review to examine the safety issues raised by the Petitioners; and (3) experimental verification of temperature calculations and heat transfer assessments and other design assumptions. Furthermore, I conclude that the Petitioners' other two requests, an assessment of potential impacts of VSC-24 material aspects and a safety justification of material coatings on components and structures, have already been fulfilled through the Staff's review of the Licensees' responses to Bulletin 96-04.

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

Malcolm R. Knapp, Acting Director
Office of Nuclear Material Safety
and Safeguards

Dated at Rockville, Maryland,
this 18th day of June 1997.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Samuel J. Collins, Director

In the Matter of

Docket No. 50-160

GEORGIA INSTITUTE OF
TECHNOLOGY
(Georgia Tech Research Reactor,
Atlanta, Georgia)

June 27, 1997

The Director of the Office of Nuclear Reactor Regulation (NRR) denies a petition filed with the Nuclear Regulatory Commission (NRC or Commission) by letter dated October 23, 1994, by Ms. Pamela Blockey-O'Brian (Petitioner), requesting that actions be taken regarding the Georgia Tech Research Reactor (GTRR) operated by the Georgia Institute of Technology (the Licensee). The petition was deferred pending a decision by the Atomic Safety and Licensing Board (ASLB) on Georgia Tech's license renewal application, in which issues substantially similar to the Petitioner's were raised. The petition is denied based on the Director's analysis of the technical issues, set forth in the Decision, which analysis showed no technical basis warranting granting the petition.

**DISPOSITION OF ISSUES RAISED VIA PETITION PURSUANT TO
10 C.F.R. § 2.206 THAT ARE ALSO UNDER CONSIDERATION IN A
PROCEEDING BEFORE A PRESIDING OFFICER**

The Commission ordinarily expects the Staff to deny a petition filed pursuant to 10 C.F.R. § 2.206 that raises the same issues that are being considered in a pending adjudication on the basis of the pendency of the identical matters in a proceeding involving the same licensee or facility. *Georgia Power Co.* (Hatch Nuclear Plant, Units 1 and 2; Vogtle Electric Generating Plant, Units 1 and 2), CLI-93-15, 38 NRC 1, 2-3 (1993); *see General Public Utilities Nuclear Corp.* (Three Mile Island Nuclear Station, Units 1 and 2; Oyster Creek Nuclear

Generating Station), CLI-85-4, 21 NRC 561, 563-65 (1985); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-81-6, 13 NRC 443, 446 (1981). (This general rule is not intended to bar a petitioner from seeking immediate enforcement action from the Staff in circumstances in which the presiding officer is not empowered to grant such relief. *Vogtle*, 38 NRC at 3.) The same result can be achieved by the Staff deferring consideration of issues raised in a petition filed pursuant to 10 C.F.R. § 2.206 that are being considered in a pending proceeding involving the same licensee and facility.

TECHNICAL ISSUES DISCUSSED

The following technical issues are discussed: Management of the GTRR; Security.

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

I. INTRODUCTION

On October 23, 1994, Ms. Pamela Blockey-O'Brien (the Petitioner) filed a petition with the U.S. Nuclear Regulatory Commission (NRC) Staff pursuant to 10 C.F.R. § 2.206. This petition requested that the NRC Staff revoke the license for the Georgia Tech Research Reactor (GTRR), shut down this research reactor and its support facilities, and remove all radioactive material and contamination off site to a government-created "National Sacrifice [A]rea" such as the Savannah River or Oak Ridge facilities. In addition, the Petitioner requested that the NRC Staff withdraw all license authority nationwide involving the discharging or dumping of any quantity of radioactive material into all the sewers or waters in the United States or oceans of the world, and withdraw all licenses to all nuclear facilities, including nuclear power plants (NPPs), that operate under "as low as reasonably achievable" (ALARA) principles. Finally, the Petitioner requested that the NRC Staff prohibit the transportation of radioactive material by mail and modify every license issued to transporters of radioactive materials and builders of NPPs to require these parties to put, in 2-foot-high letters, on everything they transport or build, the words "DANGER-RADIOACTIVE" and, in smaller letters, "there is no safe level of radiation, any exposure can effect health."

As bases for the request to shut down and decontaminate Georgia Tech Research Reactor, the Petitioner asserted that (1) a water flume comes out of the ground "destabilizing the reactor and the ground in some way"; (2) "[r]adiation levels in soil and vegetation climb markedly in GA EPD [Georgia

Environmental Protection Division] documents” around the reactor; (3) there is no record of air monitoring ever having been done; (4) heavy rainfall causes water to back up in the sewer and drainage lines causing flooding of the reactor parking lot and campus, as well as causing sinkholes, “puff-ups” on campus ground, and welded-shut manhole covers to be blown off; (5) radioactive contaminants have been routinely discharged into the sanitary sewer from the reactor’s wastewater holding tank and contamination spread by backup of the sewage system; (6) should the reactor be further destabilized, the reactor and the tank holding cobalt-60 could “break apart,” causing radioactive contaminants to “drain into groundwater/down sewers/into the runoff ditch”; (7) the reactor is in an earthquake zone; (8) there is absolutely no reason to keep the reactor operating; (9) security at the reactor is extremely lax; and (10) in case of an accident or terrorist attack, evacuation of the campus and downtown Atlanta would be impossible, especially during the 1996 Olympics.

In a Partial Director’s Decision Under 10 C.F.R. § 2.206, dated July 31, 1995 (DD-95-15), the Acting Director, Office of Nuclear Reactor Regulation (NRR), for the reasons stated in that decision, denied the Petitioner’s requests except for the request that the NRC Staff revoke the license of the GTRR, shut down this research reactor and its support facilities, and remove all radioactive material and contamination off site to a government-created “National Sacrifice [A]rea” such as the Savannah River or Oak Ridge facilities, insofar as that request rested on bases numbers (8) and (9), and that portion of basis (10) that deals with potential terrorist attacks, as set forth above. *See* DD-95-15, 42 NRC 20, 40 n.37 (1995). (The portion of basis (10) that relates to evacuation and emergency planning also is discussed in DD-95-15, 42 NRC at 40-43.)

Basis (8) includes concerns that substantial management deficiencies persist. Basis (9) involves concerns about security. Basis (10) includes concerns about evacuation in case of a terrorist attack. Since these concerns were related to issues in an ongoing license renewal proceeding before an Atomic Safety and Licensing Board (ASLB), they were not addressed in DD-95-15. The Commission ordinarily expects the Staff to deny a petition filed pursuant to section 2.206 that raises the same issues that are being considered in a pending adjudication on the basis of the pendency of the identical matters in a proceeding involving the same licensee or facility. *Georgia Power Co.* (Hatch Nuclear Plant, Units 1 and 2; Vogtle Electric Generating Plant, Units 1 and 2), CLI-93-15, 38 NRC 1, 2-3 (1993); *see General Public Utilities Nuclear Corp.* (Three Mile Island Nuclear Station, Units 1 and 2; Oyster Creek Nuclear Generating Station), CLI-85-4, 21 NRC 561, 563-65 (1985); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-81-6, 13 NRC 443, 446 (1981). (This general rule is not intended to bar a petitioner from seeking immediate enforcement action from the Staff in circumstances in which the presiding officer is not empowered to grant such relief. *Vogtle*, 38 NRC at 3.)

The same result can be achieved by the Staff deferring consideration of issues raised in a petition filed pursuant to section 2.206 that are being considered in a pending proceeding involving the same licensee and facility, as was done with regard to Petitioner's concern regarding the management of the GTRR. The NRC Staff received additional letters dated November 12 and December 4, 1994, and February 21, February 23, March 6, March 28, April 19, May 18, June 27, and July 18, 1995, from the Petitioner and also considered these letters in DD-95-15.

This Final Director's Decision addresses the management concerns in issue (8) above and security concerns in issues (9) and (10) above for the request to shut down and decontaminate the GTRR in the section 2.206 petition of October 23, 1994. The NRC Staff received additional letters from the Petitioner dated August 18, August 21, August 28, August 31, September 17, and October 27, 1995; and January 10, January 27, March 14, and May 24, 1996. All letters related to this petition were considered in this Final Director's Decision and have been placed in the Public Document Room and docketed under the GTRR Docket Number (50-160). For the reasons set forth below, the Petitioner's remaining request is denied.

II. DISCUSSION

A. Management of the GTRR

Petitioner stated that "[t]here is no reason to keep the [GTRR] operating," and asserted that substantial management deficiencies persist. As stated above, DD-95-15 did not address the management issue since it had been admitted in a proceeding on the renewal of the license for the GTRR.

The history of the license renewal proceeding is set forth in the ASLB's Initial Decision in that proceeding. LBP-97-7, 45 NRC 265, 268-70 (1997). A copy of that decision was sent to the Petitioner. In the Initial Decision, the ASLB concluded, in part, that:

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."

2. . . . 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."

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.

Id. at 312-13 (citations omitted).

The ASLB's Initial Decision considered all the evidence submitted on the record during the proceeding. The Petitioner did not submit any information to the NRC in support of its petition that was significantly different from the evidence considered by the ASLB in the license renewal proceeding on the management issue.

Since the ASLB proceeding record closed in June 1996, four additional NRC inspections of the GTRR facility have been conducted (NRC Inspection Reports No. 50-160/96-02, 50-160/96-03, 50-160/96-04, and 50-160/96-05 which were sent to the Petitioner). Three of the inspections found no violations; the violations that were found and documented in NRC Inspection Report No. 50-160/96-02, do not provide a basis for changing the NRC Staff's conclusion with regard to Georgia Tech's management of the facility.

The NRC Staff's inspection findings subsequent to the close of the ASLB record do not provide a basis for concluding that substantial management deficiencies have arisen with regard to the GTRR since the record in the license renewal proceeding closed. The Petitioner does not otherwise provide any information that would be a basis for the NRC Staff to conclude at this time that the management and organization of the Georgia Tech Research Reactor fails to comply with the Atomic Energy Act and NRC regulations. Although the Petitioner in very broad terms opposes operation of the facility, the application makes clear that its intended purpose is in keeping with lawful uses authorized in the Atomic Energy Act of 1954, as amended. The proposed operation has been found to acceptably comply with all applicable NRC regulatory requirements. Based on the foregoing, the NRC Staff concludes that no information has been provided on this issue to warrant the action requested by the Petitioner.

B. Security Issues

Petitioner raised two issues regarding security, asserting that (1) security at the GTRR is extremely lax and (2) in case of accident or terrorist attack, evacuation of the campus and downtown Atlanta would be impossible, especially during the 1996 Olympics. These two issues are discussed below.

Georgia Tech has implemented a security plan for the research reactor that is consistent with the applicable requirements of 10 C.F.R. Part 73, "Physical

Protection of Plants and Materials.” This has been confirmed through the relatively recent NRC safeguards and security-related inspection activities in NRC Inspection Reports No. 50-160/95-02, 50-160/95-04, 50-160/95-05, 50-160/96-01, 50-160/96-03, and 50-160/96-04. (Inspection Reports No. 50-160/95-02, 50-160/95-04, and 50-160/96-01 were admitted into evidence in the license renewal proceeding.)

Inspection Report No. 50-160/95-02 identified a violation for a failure to submit material status reports in a timely manner. Otherwise the inspection found that the safeguards and security activities were acceptable.

On October 26, 1995, a television news media crew entered the Neely Nuclear Research Center, which houses the GTRR, and explored and filmed portions of the center. In response, the NRC conducted an inspection of the GTRR from October 3 to November 3, 1995, as documented in NRC Inspection Report No. 50-160/95-04, which states:

This Special announced safeguards inspection was conducted to review the circumstances surrounding an uninvited tour of portions of the Neely Nuclear Research Center by a television news media crew which occurred, apparently, on the morning of October 26, 1995. . . . Neither the licensee nor the inspector could find any evidence of a security breach of the protected area. One licensee employee was identified who had seen parts of the video made by the television crew supposedly on October 26, 1995; according to that employee, the video shows two security doors being challenged by the television crew which remained locked. This employee stated that the video shows the crew touring interior and exterior areas of the Center which are open to the public or students and staff. On November 10, the inspector viewed the television showing of the video taken during this event and could find no indication that the television crew had unauthorized access to the protected/radiation controlled area. . . . No violations or deviations were identified.

In view of these inspection findings, the television media crew’s tour is not a basis for granting the Petitioner’s request.

The ASLB discussed these events in the context of the contention regarding management deficiencies, and made findings of fact consistent with this conclusion. LBP-97-7, 45 NRC at 296-98. It stated:

Upon review of the evidence of this event, we agree with the [s]taff . . . that the Fox Television film crew’s intrusion into the reactor complex does not reflect inadequate management by the [a]pplicant. To the contrary, the security plan appears to have worked as intended, in compliance with applicable regulatory requirements. Further, as observed by the [s]taff, the [a]pplicant’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.

Id. at 298 (citation omitted).

Inspection Report No. 50-160/95-05 refers to the inspection conducted December 5-7, 1995:

The special inspection addressed the facility's reactor status, physical inventory determinations, and other activities associated with maintaining a material control and accounting program within regulatory requirements, the licensed possession limit, and authorized uses of special nuclear material. . . . Within the scope of the inspection, no non-compliance issues were identified. The inspector determined that the licensee had implemented adequate controls for special nuclear material (SNM), and that accurate SNM accounting records were being maintained.

Inspection Report No. 50-160/96-01 refers to the inspection conducted on January 17 and 18, 24 and 25, 29 and 30, and February 5-7, 9, 15-18, and March 15, 1996. This inspection examined security provisions for fuel processing and shipment offsite. As an additional precaution in regards to security during the Olympic Games, the Licensee had determined to remove all GTRR fuel from the facility prior to the Games and not to replace it until after the Games. The inspection found that in addition to meeting regulatory requirements the Licensee provided additional measures (e.g., a guard was assigned to various observed activities).

Inspection Report No. 50-160/96-03 refers to the inspection conducted on June 17, 18, and 27, and July 3, 5, and 11, 1996. This inspection included onsite and offsite review of security preparations for the Olympic Games. The inspection concluded: "The controls implemented by the licensee and the precautions taken are adequate to protect licensee personnel and the public."

The inspection documented in Inspection Report No. 50-160/96-04 was conducted on July 17 and 29, 1996. This inspection reviewed the preparation for the Summer Olympic Games and found that:

[T]he university had taken additional safeguards measures to control access to the Campus and to the Research Control Area. The licensee had taken additional safeguards measures to control access to the Neely Nuclear Research Center (NNRC). The additional security measures taken as a result of the 1996 Olympic Games were reviewed and/or observed by the inspectors. . . . On July 17 and 29, 1996, the inspectors visited the Neely Nuclear Research Center, met with the Director of the Center, toured the facility and verified continued compliance with the Physical Security Plan (PSP). The inspectors were granted unfettered access to the Research Control Area as well as to the Center and emergency access during the Olympics was assured because the inspectors and selected management of Region II had been provided with special picture badges to facilitate NRC response. The presence of military police, Campus police and additional State and Federal law enforcement officers in the immediate vicinity of the Center was observed by the inspectors. The access controls, barriers, assessment capabilities, communication capabilities and detection equipment required by the NRC were in place. Additional exterior lights had been installed by the licensee to assist patrolling officers. Additional fencing around the Center was also noted by the inspectors. . . . The inspector concluded that the licensee was meeting NRC requirements and had effectively imposed proactive security measures.

With regard to the contention on the physical security of the site during the 1996 Summer Olympic Games held in Atlanta, Georgia, the ASLB decision observed that

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 . . . and we issued a Partial Initial Decision dismissing the contention as moot.

LBP-97-7, 45 NRC at 270. *See* LBP-95-19, 42 NRC 191 (1995).

In summary, the physical security plan was verified to provide acceptable procedures for event response and access control, and the security preparations for the Olympics were acceptable. Observations of the facility and activities confirmed the use of security-related equipment and controls as required by the physical security plan and consistent with the special nuclear material that is present at the facility. The Petitioner asserted that security at the research reactor was lax; however, access is controlled and monitored as required. Further, this evaluation confirmed the continued acceptability of the security provisions to deal with potential terrorists attacks. The findings do not provide a basis for changing the conclusion reached in DD-95-15 on the adequacy of emergency plans for the facility. DD-95-15, 42 NRC at 40-43. The NRC Staff has found no reason to conclude that the security at the reactor is not acceptable. The Petitioner provided no facts to conclude otherwise.

III. CONCLUSION

With regard to the requests made by the Petitioner discussed herein, the NRC Staff finds no basis for taking such actions. Accordingly, the Petitioner's requests for action, pursuant to section 2.206 on the Georgia Tech Research Reactor, are denied.

A copy of this Decision will be filed with the Secretary for the Commission as provided by 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 review of the Decision in that time.

**FOR THE NUCLEAR
REGULATORY COMMISSION**

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

**Dated at Rockville, Maryland,
this 27th day of June 1997.**

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