NUCLEAR REGULATORY COMMISSION ISSUANCES

June 1997

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NUCLEAR REGULATORY COMMISSION ISSUANCES

June 1997

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

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

U.S. NUCLEAR REGULATORY COMMISSION

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

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COMMISSION

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).

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. Tetric, 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

**Directors'
Decisions
Under
10 CFR 2.206**

DIRECTORS' DECISIONS

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.

⁴ The 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

¹ Section 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.

² The 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
72-5CONSUMERS POWER COMPANY
(Palisades Nuclear Plant)Docket Nos. 50-255
72-7ENTERGY 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 conservatisms 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.