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Vol. 45, No. 2
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NUCLEAR REGULATORY COMMISSION ISSUANCES

February 1997

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NUREG-0750
Vol. 45, No. 2
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NUCLEAR REGULATORY COMMISSION ISSUANCES

February 1997

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

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

U.S. NUCLEAR REGULATORY COMMISSION

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

COMMISSIONERS

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

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

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Commission Issuances

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

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

In the Matter of **Docket No. 70-3070-ML**

LOUISIANA ENERGY SERVICES, L.P.
(Claiborne Enrichment Center) **February 13, 1997**

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

ORDER

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

I. SCHEDULING OF BRIEFS

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

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

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

II. REMAINING ISSUES BEFORE THE BOARD

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

IT IS SO ORDERED.

For the Commission[1]

JOHN C. HOYLE
Secretary of the Commission

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

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

Atomic Safety and Licensing Boards Issuances

ATOMIC SAFETY AND LICENSING BOARD PANEL

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LICENSING BOARDS

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

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

In the Matter of | **Docket No. 55-20726-SP**
(ASLBP No. 96-721-01-SP)
(Re: Operator License)

RALPH L. TETRICK
(Denial of Application for Reactor Operator License) | **February 28, 1997**

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

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

INITIAL DECISION

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

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

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

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

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

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

---

[1] This is an informal hearing under 10 C.F.R. Part 2, Subpart L. *See* 10 C.F.R. § 2.1201(a)(2). By letter of November 7, 1996, the NRC Staff ("Staff") submitted the Hearing File pursuant to 10 C.F.R. § 1.1231. On December 30, 1996, Mr. Tetrick filed his written presentation in this proceeding, pursuant to 10 C.F.R. § 2.1233 (Tetrick Presentation). Staff replied, pursuant to this same section of the regulations, on January 23, 1997 (Staff Presentation).

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

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

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

I. QUESTION 63

A. The Question

Examination Question 63 stated as follows:

Plant conditions:

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

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

a. *Verify alarms by checking containment sump level recorder and spent fuel level indication.*

b. *Sound the containment evacuation alarm.*

c. *Initiate containment ventilation isolation.*

d. *Initiate control room ventilation isolation.*

B. Staff Position

Staff contends[3] that the correct answer to this question is "*b. Sound the containment evacuation alarm.*" It relies on Procedure 0-ADM-219, § 3.4.1

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

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

> The plant conditions and indications specified in this question (*i.e.,* the refueling cavity filled and the transfer tube gate valve open with coincident SFP LOW LEVEL and CONTAINMENT SUMP HIGH LEVEL alarms) *are mutually supportive and confirmatory,* and require entry into Off-Normal Operating Procedure 3-ONOP-033.2, "Refueling Cavity Seal Failure" (Hearing File #24[4]). [Emphasis added.]

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

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

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

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

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

C. Mr. Tetrick's Position

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

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

D. Conclusion

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

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

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

II. QUESTION 84

A. The Question

Examination Question 84 stated as follows:

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

a. To ensure that only decay heat and reactor coolant pumps are adding heat to the RCS.

b. To ensure shutdown margin is within Technical Specifications limits for HOT STANDBY.

c. To alert the operator to take further corrective action if the reactor is NOT tripped.

d. To verify that all automatic reactor protective features have functioned as designed.

B. Staff Position

Staff states that the correct answer is "a." Staff argues that the question requests the basis (or reason) for Step 1, Verify Reactor Trip, of FR-S.1, Response to Nuclear Power Generation/ATWS. To determine the basis for Step 1, I first examine Step 1 in the following table:

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

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

C. Mr. Tetrick's Position

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

D. Conclusion

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

III. QUESTION 90

A. The Question

Examination Question 90 stated as follows:

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

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

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

b. A vacuum in the RCS loops will result in level indication being higher than actual levels.

c. A positive pressure in the RCS loops will result in level indication being lower than actual levels.

d. The level instruments automatically compensate for positive or negative pressure.

B. Mr. Tetrick's Position

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

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

C. Staff Position

Staff states that the correct answer is:

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

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

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

The important leg to consider here is the variable leg of the water-level instrument. When there is a vacuum above the water in the pressure vessel, there will be less pressure on the variable leg than if the space above the water were filled by air at atmospheric pressure. The purpose of the "reference leg" of the pressure indicator is to measure the height of water that corresponds to the pressure on the variable leg. Providing that there is no malfunction affecting the reference leg, it does not matter whether the design uses a wet or a dry reference leg. The answer will be the same: an accurate measurement of the height of the water in the variable leg. Staff Affidavit at 16-17, ¶¶37-39.

Staff states that:

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

(Emphasis added.)

D. Conclusion

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

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

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

IV. QUESTION 96

A. The Question

Examination Question 96 stated as follows:

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

a. Nuclear Plant Supervisor

b. Assistant Nuclear Plant Supervisor

c. Senior Nuclear Plant Operator

d. Nuclear Watch Engineer

B. Staff Position

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

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

C. Mr. Tetrick's Position

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

D. Conclusion

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

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

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

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

V. OVERALL CONCLUSION

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

VI. ORDER

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

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

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

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

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

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

Peter B. Bloch, Presiding Officer
ADMINISTRATIVE JUDGE

Rockville, Maryland

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

ENVIROCARE OF UTAH, INC. **February 5, 1997**

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

ATOMIC ENERGY ACT: ENFORCEMENT ACTION (HEARING RIGHT)

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

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING

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

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING

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

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

I. INTRODUCTION

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

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

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

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

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

II. BACKGROUND

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

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

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

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

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

III. DISCUSSION

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

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

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

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

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

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

The NRC's Enforcement Policy describes the various enforcement sanctions available to the Commission once it determines that a violation of its requirements has occurred. In accordance with the guidance in section VI.C.3 of the Enforcement Policy, Revocation Orders may be used: (a) when a licensee is unable or unwilling to comply with NRC requirements; (b) when a licensee refuses to correct a violation; (c) when a licensee does not respond to a Notice of Violation where a response was required; (d) when a licensee refuses to pay an applicable fee under the Commission's regulations; or (e) for any other reason for which revocation is authorized under section 186 of the Atomic Energy Act (e.g., any condition that would warrant refusal of a license on an original application). Pursuant to 10 C.F.R. § 2.202(a)(5), the Commission may issue an immediately effective order to modify, suspend, or revoke a license if the Commission finds that the public health, safety, or interest so requires or that the violation or conduct causing the violation was willful. The Commission's regulations recognize that a licensee should be afforded under usual circumstances a prior opportunity to be heard before the agency suspends a license or takes other enforcement action, but that extraordinary circumstances may warrant summary action prior to hearing. *See Advanced Medical Systems, Inc.* (One Factory Row, Geneva, Ohio 44041), CLI-94-6, 39 NRC 285, 299 (1994).

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

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

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

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

With regard to this request, we have already noted that there is no basis for NRC to take immediate action. In any event, section 2.206 is not a venue for presenting licensing contentions of the sort raised by this aspect of NRDC's petition. Section 2.206 provides for requests for action under that portion of the NRC's regulations governing enforcement actions, namely 10 C.F.R. Part 2, Subpart B. Subpart B is entitled "Procedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties." Since the inception of the section 2.206 process, the Commission has consistently stated that the purpose of section 2.206 is to provide the public with the means for *participating in the enforcement process.*[3] The Commission has determined that the section 2.206 process should be focused on requests for enforcement action rather than evaluations of safety concerns. In accordance with this determination, the Commission's Management Directive 8.11, "Review Process for 10 C.F.R. 2.206 Petitions," Part III, section A, states that petitions will be reviewed under section 2.206 if the request is for enforcement action, and that a request under section 2.206 should be distinguished from a request to deny a pending license application or amendment.

Because this request by the NRDC concerns licensing-type action, not enforcement-type action, the Staff has determined that, consistent with the guidance of Management Directive 8.11, this request is not within the scope of section 2.206.[4] To the extent that further facts may be developed that may warrant consideration of this request, the matter may be raised in an individual licensing proceeding; however, no such proceeding is presently pending, as there is no application pending for the issuance of a license to Envirocare.

IV. CONCLUSION

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

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

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

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

FOR THE NUCLEAR
REGULATORY COMMISSION

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

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Carl J. Paperiello, Director

In the Matter of **Docket Nos. 50-346**
72-1004

TOLEDO EDISON COMPANY, ***et al.***
(Davis-Besse Independent Spent Fuel Storage Installation) **February 5, 1997**

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

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

INTRODUCTION

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

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

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

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

The petition was referred to me pursuant to NRC regulations in 10 C.F.R. § 2.206.[2] Because the petition requested immediate relief (i.e., a halt to any loading of the DSCs at Davis-Besse), it was necessary for me to give an immediate response to that portion of the Petitioners' request. By letter dated December 18, 1995, I denied the Petitioners' request for immediate action on the petition on the basis of my judgment that there was (and continues to be) no imminent risk to health, safety, or environment such as to warrant the emergency relief sought by the Petitioners.[3]

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

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

BACKGROUND

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

DISCUSSION

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

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

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

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

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

In NRC's original evaluation, when it certified the NUHOMS and accepted VECTRA's SAR in 1994, the NRC Staff reviewed a variety of potential cask accidents (e.g., a cask drop or tipover, vent blockage leading to cask heatup, low temperatures, earthquakes and tornadoes, explosions, lightning, floods) that were thought to cover the range of cask accidents that might reasonably be assumed to occur. In the NRC review, the accident was assumed to occur (i.e., probability of occurrence was assumed to be one), and the consequences were evaluated. For each accident, the NRC Staff review found that the DSC would maintain confinement of the spent fuel without any breach or rupture of the DSC. Therefore, there could be no adverse impact on the public. As noted, the original NRC evaluation was based on a DSC nominal shell thickness of 0.625 inch.

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

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

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

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

VECTRA used the same ASME Code provisions for evaluating the DSC designs[10] and demonstrated that the Code provisions were met by a DSC shell thickness of 0.500 inch.[11] Thus, even with the reduction in shell thickness, VECTRA demonstrated that the ASME Code provisions will be met by the DSC shell thickness of 0.500 inch.[12]

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

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

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

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

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

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

Thus, while VECTRA failed to comply with its SAR commitment of 0.625 inch, its failure resulted in no compromise of safety. Nonetheless, the failure raised an issue of poor control during the fabrication process. This deficiency was identified by NRC during the June 1995 inspection; and VECTRA was cited for it in the NRC Notice of Nonconformance issued to VECTRA in August 1995.[13]

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

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

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

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

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

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

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

Petitioners' November 14 questions appear to be aimed at VECTRA's implementation of Condition 9 in the NRC COC issued to VECTRA in December 1994.[15] Condition 9 permits VECTRA to make changes in the DSC design without NRC approval provided, among other things, the change does not involve "an unreviewed safety question." Condition 9 states that a change shall be deemed to involve an unreviewed safety question "[i]f the margin of safety as defined in the basis for any technical specification or limit is reduced." After evaluation, VECTRA concluded that the wall thinning of the three DSCs at Davis-Besse did not involve a reduction in the "margin of safety" or "an unre-

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

It is not NRC's practice to review and approve a licensee's operating procedures. It is important to understand that, just with respect to dry cask storage activities, which are a very small fraction of the daily activities conducted at an operating nuclear power plant, the applicable written procedures of a general licensee are likely to be voluminous. Moreover, the written procedures prepared by a licensee typically are site-specific in nature and thus reflect the licensee's special knowledge of its plant and how dry cask storage activities interconnect with plant personnel, as well as other plant activities and procedures. The written procedures are prepared according to the formal procedure development process and exercised during the dry run. In my judgment, there would be very little additional value to be gained from a requirement of NRC review and approval of a licensee's written operating procedures, particularly given our existing inspection activities illustrated by the Davis-Besse example, below.

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

CONCLUSION

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

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

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

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

FOR THE NUCLEAR
REGULATORY COMMISSION

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

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

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Frank J. Miraglia, Jr., Acting Director

In the Matter of **Docket No. 50-245**
(License No. DPR-21)

NORTHEAST NUCLEAR ENERGY COMPANY
(Millstone Nuclear Power Station, Unit 1) **February 11, 1997**

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

ENFORCEMENT POLICY: SEVERITY OF VIOLATIONS

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

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

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

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

I. INTRODUCTION

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

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

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

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

II. DISCUSSION

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

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

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

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

communications" regarding the procedure has caused a "widespread problem of procedure noncompliance."[4]

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

III. CONCLUSION

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

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

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

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

As provided in 10 C.F.R. § 2.206(c), a copy of this Decision will be filed with the Secretary of the Commission for the Commission's review. This Decision will constitute the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes a review of the Decision in that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

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

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