# NUCLEAR REGULATORY COMMISSION ISSUANCES

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

January 1, 1984 - March 31, 1984

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# PREFACE

This is Book I of the nineteenth volume of issuances (1 - 936) of the Nuclear Regulatory Commission and its Atomic Safety and Licensing Appeal Boards, Atomic Safety and Licensing Boards, and Administrative Law Judge. It covers the period from January 1, 1984 to March 31, 1984.

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

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

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

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

Issuances are referred to as follows: Commission--CLI, Atomic Safety and Licensing Appeal Boards--ALAB, Atomic Safety and Licensing Boards--LBP, Administrative Law Judge--ALJ, Directors' Decisions--DD, and Denial of Petitions for Rulemaking--DPRM.

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

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**COMMISSIONERS:**

**Nunzio Palladino**, Chairman
**Victor Gilinsky**
**Thomas M. Roberts**
**James K. Asselstine**
**Frederick M. Bernthal**

**In the Matter of** **Docket Nos. 50-275**
**50-323**

**PACIFIC GAS AND ELECTRIC COMPANY**
**(Diablo Canyon Nuclear Power Plant, Units 1 and 2)** **January 16, 1984**

The Commission denies the intervenors' request for a stay of fuel loading and pre-criticality testing at the Diablo Canyon plant.

# ORDER

The Commission hereby denies Joint Intervenors' request for a stay of fuel loading and pre-criticality testing at Diablo Canyon. As the Commission noted in reinstating this limited authority, fuel loading and pre-criticality testing do not present significant public health and safety risks. *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-83-27, 18 NRC 1146 (1983). In addition, the presently authorized activities will not prejudice subsequent decisions or foreclose modification, if necessary, of the facility. *Id.* This decision is without prejudice to renewal of the stay request at subsequent stages of authorization.

Commissioner Gilinsky abstained from this decision. His separate views are attached.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 16th day of January 1984.

## SEPARATE VIEWS OF COMMISSIONER GILINSKY (DIABLO CANYON, CLI-84-1, SECY-83-512)

There is little point to the Commission's action since the request for a stay of fuel loading and pre-criticality testing has been overtaken by events. However, I would note that while fuel loading does not present the type of risk associated with reactor operation, its safety significance has nonetheless been excessively downplayed.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**COMMISSIONERS:**

**Nunzio J. Palladino,** Chairman
**Victor Gilinsky**
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**James K. Asselstine**
**Frederick M. Bernthal**

**In the Matter of** **Docket No. 50-275**

**PACIFIC GAS AND ELECTRIC COMPANY**
**(Diablo Canyon Nuclear Power Plant, Unit 1)** **January 25, 1984**

Acting on the applicant's request, the Commission authorizes further pre-criticality tests (hot system testing) at the Diablo Canyon plant on the ground that the tests will provide valuable information regarding plant design, construction and operation without presenting any significant public health and safety concerns.

# MEMORANDUM AND ORDER

This matter comes before the Commission on licensee Pacific Gas and Electric Company's ("PG&E" or "licensee") January 4, 1984 request for reinstatement of the authority to conduct further pre-criticality tests, operational modes 4 and 3 as described in the Technical Specifications, at Diablo Canyon Nuclear Power Plant, Unit 1.

On November 8, 1983, the Nuclear Regulatory Commission reinstated the licensee's authority under Facility Operating License No. DPR-76 to load fuel and conduct pre-criticality activities in modes 6 and 5, as described in the Unit 1 Technical Specifications. Memorandum and Order, CLI-83-27, 18 NRC 1146 (1983). The Commission based its decision upon the determination "that the results of the [Independent Design Verification Program (IDVP)] provide reasonable assurances of protection of the public health and safety insofar as [fuel loading and pre-criticality testing] are concerned." *Id.* at 1150-51.

On January 4, 1984, PG&E requested authority to proceed to operational modes 4 and 3, pre-critical hot system testing. This further stage of operation would enable hot system testing[1] and certain equipment calibration to be conducted while the nuclear fuel is still in a pre-critical condition; *i.e.,* there would be no self-sustaining nuclear chain reaction and no significant production of radioactive fission products.

The NRC staff and Governor George Deukmejian support PG&E's request, both because no safety hazards are presented by pre-criticality testing and because hot system testing can provide valuable information regarding the design, construction, and operation of the facility. The staff notes in particular that information obtained during hot system testing would be of assistance in evaluating a number of pending allegations regarding plant construction and design.

The Joint Intervenors in the licensing proceeding oppose PG&E's request, citing continuing concerns with design and construction quality assurance and numerous allegations of improper design or construction. Joint Intervenors have offered no explanation how the concerns they raise relate to pre-critical operations and have offered no reason why the rationale for our order authorizing modes 6 and 5 does not apply equally to modes 4 and 3.

Taking the foregoing into consideration, the Commission has decided to *grant* PG&E's request and hereby authorizes further pre-criticality testing at Diablo Canyon, Unit 1, in operational modes 4 and 3 as defined in the Technical Specifications. This further stage of operation will provide valuable information regarding plant design, construction and operation without presenting any significant public health and safety concerns. In accordance with the Commission's previous statement, this authorization in no way prejudices future decisions regarding the operation of this facility.

Commissioner Gilinsky abstained from this decision. His separate views are attached.

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[1] The term "hot" does not imply that any radioactivity is involved. It merely refers to the fact that certain plant systems would run at elevated temperature and pressure. The heat necessary for hot system testing is generated by the operation of pumps and other non-nuclear sources, not by any process involving the production of radioactive fission products.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 25th day of January 1984.

## SEPARATE VIEWS OF COMMISSIONER GILINSKY SECY-84-1; Modification of Diablo Canyon Order (CLI-83-27)

When the Commission granted Diablo Canyon the authority to load fuel last November, I withheld my own approval for two reasons: the Commission had not addressed a fundamental question related to the adequacy of the seismic design standard applied to this plant; and, I had not visited the plant since modifications were made to correct the design errors discovered in 1981 and wanted to have a first-hand look before I acted.

The seismic standard situation remains the same. Notwithstanding the recommendation of our Advisory Committee that the Commission conduct a comprehensive review of the seismic design in the first few years of operation, the Commission has done nothing. I continue to think that the Commission needs to commit itself to doing such a review, before authorizing further operation.

The second reason for my holding back approval today grows out of my visit to Diablo Canyon last December. It appears that none of the Diablo Canyon licensed operators have any prior licensed commercial experience at large power reactors. (Nor is there even a plant-specific simulator). My own view, which I have voiced in the face of similar problems at Shoreham and Grand Gulf, is that a requirement for full-power operation should be that at least one experienced supervisor will be assigned to each shift. At a bare minimum, the Commission should require that any ascension to power by an inexperienced crew be a good deal more gradual than otherwise, and subject to formal evaluations at each stage by the Company and the Commission. The Commission has yet to act on this issue.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Alan S. Rosenthal, Chairman**
**Gary J. Edles**
**Howard A. Wilber**

**In the Matter of** **Docket Nos. 50-443-OL**
**50-444-OL**

**PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, *et al.***
**(Seabrook Station, Units 1 and 2)** **January 24, 1984**

The Appeal Board affirms, on different grounds, the Licensing Board's denial of an untimely petition for leave to intervene in this operating license proceeding.

## OPERATING LICENSE PROCEDURES: RESPONSIBILITY OF NRC STAFF

It is the responsibility of the Director of Nuclear Reactor Regulation, and not the Licensing Board, to make the finding required by 10 C.F.R. 50.57(a)(1) as a precondition to the issuance of an operating license for a nuclear plant. *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-226, 8 AEC 381, 410-11 (1974).

## APPEARANCES

**John F. Doherty,** Boston, Massachusetts, petitioner *pro se.*

**Thomas G. Dignan, Jr.,** and **R.K. Gad III,** Boston, Massachusetts, for the applicants, Public Service Company of New Hampshire, *et al.*

**William F. Patterson, Jr.,** for the Nuclear Regulatory Commission staff.

# DECISION

Before us is the appeal of John F. Doherty from the Licensing Board's denial in a November 15, 1983 order (unpublished) of his untimely petition for leave to intervene in this operating license proceeding. The denial was founded on a balancing of the five factors governing the acceptance of a belated intervention petition.[1] Although not necessarily in full agreement with the Board's analysis, we affirm its result on entirely different grounds.

## I

A. This proceeding involves both Units 1 and 2 of the Seabrook nuclear facility located on the New Hampshire seacoast. On October 19, 1981, the Commission published the customary notice of opportunity for hearing on the operating license application that had been filed for both units.[2] The notice stated that petitions for leave to intervene were to be filed by November 18, 1981.[3]

Several intervention petitions were filed in the wake of the notice. Thereafter, the Licensing Board admitted a substantial number of organizations and governmental bodies to the proceeding, either as parties under 10 C.F.R. 2.714(a) or as interested states or municipalities under 10 C.F.R. 2.715(c).[4] Last August, the Licensing Board held evidentiary hearings on certain safety and onsite emergency planning issues. Offsite emergency planning issues remain to be heard.

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[1] Those factors are set forth in 10 C.F.R. 2.714(a)(1).
[2] 46 Fed. Reg. 51,330.
[3] *Id.* at 51,331.
[4] *See* 48 Fed. Reg. 32,417, 32,418 (1983).

B. Mr. Doherty filed his intervention petition on September 6, 1983 — almost two years after the prescribed deadline. The petition set forth a single contention: that the application for an operating license for Unit 2 is premature and should be denied for that reason. The assigned basis for the contention was that Unit 2 is only 22 percent completed and "many more than four years are likely to remain before the unit is substantially completed in conformance with N.R.C. rules and regulations."[5] According to Mr. Doherty, in these circumstances the filing of the Unit 2 application violated 10 C.F.R. 50.57(a)(1),[6] and its grant prior to the substantial completion of the unit would threaten his health, safety and economic interests.[7]

On the matter of the petition's lateness, Mr. Doherty explained that he had lacked standing to intervene before June 23, 1983, when he acquired his present residence in the general vicinity of the plant.[8] Between that date and August 26, he had assumed that, given the "decreased demand" for electricity and the "lack of [Unit 2] construction," the applicants would not be pressing for an operating license for that unit.[9] On August 26, he made a limited appearance statement before the Licensing Board, in which he presented his prematurity claim.[10] But the Board took no action on the claim, "evidently not being empowered to do so."[11] Turning then to the other four Section 2.714(a) lateness factors (*see* note 1, *supra*), Mr. Doherty maintained that there are no sufficient alternative means available for the protection of his interest; that no existing party to the proceeding has indicated an intent to raise the prematurity issue; that his participation would contribute to the development of a sound record, and that the prematurity issue was worthy of whatever time might be involved in its exploration.[12]

In their responses, the applicants and the NRC staff urged that, taken collectively, the Section 2.714(a) lateness factors tipped against a grant

[5] John F. Doherty's Petition for Leave to Intervene (September 6, 1983) at 2.

[6] That Section requires, as a precondition to the issuance of an operating license, a finding that:

> Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

[7] Doherty Petition at 2. Those interests were said to stem from the fact that Mr. Doherty resides 40 miles from the Seabrook site, travels and recreates very near the site, and is a rate payer of a utility that purchases power from a Seabrook co-owner. *Id.* at 1-2.

[8] *Id.* at 5. Prior thereto, he had resided for a number of years in Texas. *Id.* at 1, 5.

[9] *Id.* at 5.

[10] *Ibid.*

[11] *Ibid.*

[12] *Id.* at 6-7.

of the petition.[13] As previously noted, the Licensing Board agreed and, accordingly, denied the petition.[14]

## II

A. By his late petition, Mr. Doherty seeks to inject into this proceeding a single issue: whether, given the indisputable fact that Unit 2 of the Seabrook facility is still in a relatively early stage of construction, the operating license application for that unit must be dismissed as premature by virtue of 10 C.F.R. 50.57(a)(1).[15] Not only is this issue purely legal in character — and thus requires no evidentiary hearing for its resolution — but, since the time of Mr. Doherty's filing below, it has been both presented to the Licensing Board by a party to the proceeding and decided by the Board.

On September 26, 1983 (exactly 20 days after the submission of the untimely Doherty petition), intervenor Seacoast Anti-Pollution League (SAPL) filed a motion "for dismissal of the Seabrook Unit II operating license." The motion was founded on precisely the same assertion advanced by Mr. Doherty — namely, that the issuance of an operating license for Unit 2 at this juncture is foreclosed by 10 C.F.R. 50.57(a)(1) in light of the current status of that unit's construction.[16] Thereafter, on December 14, SAPL filed a further memorandum in support of the motion, together with a petition for leave to put forth the prematurity claim as a late-filed contention.

In a January 13, 1984 unpublished memorandum and order, the Licensing Board denied the SAPL motion for the following reasons:

> The Board finds no basis for it to consider dismissal of the application for an operating license for Unit II. The Board's authority in this instance is to submit its initial decision on controverted issues before it which may *authorize* the granting a license for the facility. 10 C.F.R. § 1.11, § 2.760(a). The *issuance* of a license is vested in the Office of Nuclear Reactor Regulation. 10 C.F.R. 1.61. The jurisdiction of the Board

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[13] Applicant's Response to John F. Doherty's Petition for Leave to Intervene (September 19, 1983) at 2-5; NRC Staff Response Opposing "John F. Doherty's Petition for Leave to Intervene" (September 26, 1983) at 2-6.

[14] November 15, 1983 Order at 4-8.

[15] On September 8, 1983, the lead applicant issued a press release in which it announced that the work on Unit 2 had been reduced to the "lowest feasible level" in order to permit, among other things, a "maximum effort to be put toward completing Unit I while maintaining Unit II in an active state." The press release noted that Unit 2 was "23.4 percent complete."

[16] Although the SAPL motion correctly quoted the text of Section 50.57(a)(1), it erroneously cited it as Section 50.47(1). SAPL's Motion to Dismiss the Operating License Application for Seabrook Unit II (September 26, 1983) at 1, 3. The other parties and the Licensing Board were fully aware, however, that SAPL meant to refer to Section 50.57(a)(1).

is established by 10 C.F.R. 2.721(a). Section 2.721(a) limits the Boards to presiding over such proceedings for granting, suspending, revoking, or amending licenses or authorizations as the Commission may designate, and to perform such other adjudicatory functions as the Commission deems appropriate. Nothing in the sections cited confers upon this Board the power to make a determination required under Section 50.57(a)(1). Such a finding is not, nor can it be, an issue which this board has before it. Any determination as to whether Unit II is substantially completed must be made by the Commission or its delegate, the Director of Nuclear Reactor Regulation.[17]

B. Accordingly, events clearly have overtaken Mr. Doherty's intervention effort. As matters now stand, his objective of having the Unit 2 prematurity issue placed before the Licensing Board has been achieved — albeit through the endeavors of someone else. True, on the SAPL motion the Board determined the issue against Mr. Doherty's position. There is no reason to suppose, however, that the Board would have decided it any differently had it considered his claim rather than SAPL's.

Given that Mr. Doherty has not identified any other issue that he would raise and pursue in this proceeding, it would thus appear that the denial of his petition has very limited, if any, practical significance. Indeed, all that he seems to have lost by that denial is the ability to seek our review of the Licensing Board's conclusion that the Unit 2 operating license application is now properly in adjudication. But, to have prevailed on any such review, he would have had to persuade us that the conclusion not merely was wrong but, in addition, concretely and adversely affected some personal interest of his own.

At the very least, the Licensing Board's analysis of the Unit 2 prematurity question in its January 13 memorandum and order is not manifestly (or even probably) erroneous.[18] In any event, Mr. Doherty has not attempted to explain how the adjudication of the Unit 2 application at this point might pose a threat to an interest possessed by him in the safe and environmentally acceptable operation of that unit. His late intervention

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17 Memorandum and Order at 5. The Licensing Board went on to deny the late-filed SAPL contention raising the same prematurity issue. *Id.* at 6-10.

18 To the contrary, this much is clear: First, the Licensing Board correctly held that it is not its responsibility, but that of the Director of Nuclear Reactor Regulation, to make the finding required by Section 50.57(a)(1) as a precondition to the issuance by the Director of an operating license. *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-226, 8 AEC 381, 410-11 (1974). Second, there is nothing in the Commission's regulations specifically providing that a reactor must have reached a particular stage of completion before an operating license application may be filed. Third, just 16 months ago the Commission denied a petition for rulemaking that sought amendments to the Rules of Practice that would have, *inter alia,* limited the scope of each operating license hearing to a single reactor unit even if that unit were one of several similar units constructed on a multi-reactor site. 47 Fed. Reg. 46,524 (1982). In support of his proposal, the petitioner had noted that the "time lag between inservice dates for individual reactors at multi-reactor nuclear plants has been increasing for many years." *Ibid.* In the Commission's view, however, that consideration did not provide a sufficient basis for requiring "an exclusive hearing on each reactor unit." *Id.* at 46,525.

petition raises the prematurity issue abstractly, without even a passing reference to any specific safety or environmental concern that could be tied to the inclusion of Unit 2 in the ongoing adjudication.[19]

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In the foregoing circumstances, we are disinclined to overturn the result below. The Section 2.714(a) lateness factors to one side, there is no discernible reason why Mr. Doherty should now be granted intervention for the sole purpose of enabling him to press a legal argument that (1) has already been rejected by the Licensing Board; (2) is of dubious merit; and (3) has not been shown to further any interest of his that might be affected by the operation of Unit 2.[20]

The denial of Mr. Doherty's intervention petition is therefore *affirmed.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

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[19] In his petition (at 4), Mr. Doherty took note of the possibility that Unit 2 might not be completed in the manner now contemplated because of such contingencies as regulatory changes, uncorrected construction errors or unavailability of materials. But there will be time enough for him to seek appropriate relief when (and if) his interests are concretely and adversely affected by some new development associated with Unit 2 construction. *See generally Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), ALAB-687, 16 NRC 460, 467-70 (1982), *modified,* CLI-83-19, 17 NRC 1041 (1983).

[20] In his appellate brief, Mr. Doherty asserts that the denial of his intervention petition by the Licensing Board offended due process. *See* Doherty Brief in Support of his Appeal (December 1, 1983) at 3. That assertion need not detain us long. For one thing, there was no claim below that, despite its tardiness, the petition had to be granted as a matter of due process. More important, Mr. Doherty's brief does not elaborate upon the foundation of the due process claim and we fail to see any substance to it in the context of this case.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Alan S. Rosenthal, Chairman**
**Thomas S. Moore**
**Dr. Reginald L. Gotchy**

**In the Matter of** **Docket No. 50-354-OL**

**PUBLIC SERVICE ELECTRIC AND GAS COMPANY, *et al.***
**(Hope Creek Generating Station, Unit 1)** **January 25, 1984**

Upon consideration of an order (referred to it by an administrative judge) denying an intervenor's motion that he recuse himself from further service as a member of the Licensing Board for this operating license proceeding, the Appeal Board rules that the judge must be replaced on the Licensing Board by another member of the Licensing Panel.

## DISQUALIFICATION: STANDARDS

Licensing Board members are governed by the same disqualification standards that apply to federal judges. *Houston Lighting and Power Co.* (South Texas Project, Units 1 & 2), CLI-82-9, 15 NRC 1363, 1365-67 (1982).

## DISQUALIFICATION: STANDARDS

An administrative trier of fact is subject to disqualification if he has a direct, personal, substantial pecuniary interest in a result; if he has a "personal bias" against a participant; if he has served in a prosecutive or investigative role with regard to the same facts as are in issue; if he has prejudged factual — as distinguished from legal or policy — issues; or if he has engaged in conduct which gives the appearance of personal bias or prejudgment of factual issues. *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-101, 6 AEC 60, 65 (1973).

### DISQUALIFICATION: STANDARDS

The current statutory foundation for the Commission's disqualification standards is found in 28 U.S.C. 144 and 455.

### DISQUALIFICATION: STANDARDS

The current Section 455(a) of 28 U.S.C. imposes an objective standard for recusal; *i.e.,* whether a reasonable person knowing all the circumstances would be led to the conclusion that the judge's impartiality might reasonably be questioned. *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), CLI-82-9, 15 NRC 1363, 1366, *citing Fredonia Broadcasting Corp. v. RCA Corp.,* 569 F.2d 251, 257 (5th Cir. 1978). And, as a general proposition, recusal under this section must rest upon extrajudicial conduct. 15 NRC at 1367.

### DISQUALIFICATION: STANDARDS

28 U.S.C. 455(b)(2) requires a judge to disqualify himself in circumstances where, *inter alia,* in private practice the judge served as a lawyer "in the matter in controversy." Disqualification in such circumstances may not be waived. *See* 28 U.S.C. 455(e); *SCA Services Inc. v. Morgan,* 557 F.2d 110, 117 (7th Cir. 1977).

### COMMISSION PROCEEDINGS: RES JUDICATA/ COLLATERAL ESTOPPEL

The doctrines of res judicata and collateral estoppel apply in operating license proceedings. *See Alabama Power Co.* (Joseph M. Farley Nuclear Plant, Units 1 and 2), ALAB-182, 7 AEC 210, *modified on other grounds,* CLI-74-12, 7 AEC 203 (1974).

## APPEARANCES

**R. William Potter** and **Susan C. Remis**, Trenton, New Jersey, for Joseph H. Rodriguez, Public Advocate of the State of New Jersey.

**Troy B. Conner, Jr.**, Washington, D.C., for the applicants, Public Service Electric and Gas Company, *et al.*

**Lee Scott Dewey** for the Nuclear Regulatory Commission staff.

# MEMORANDUM AND ORDER

On November 18, 1983, intervenor Public Advocate of the State of New Jersey (Advocate) filed a motion under 10 C.F.R. 2.704(c) requesting, *inter alia,* that Administrative Judge James H. Carpenter disqualify or recuse himself from further service as a member of the Licensing Board for this operating license proceeding involving the Hope Creek nuclear facility. On December 27, 1983, Judge Carpenter entered an unpublished order in which he denied the motion. As required by Section 2.704(c), the order referred the matter to us.

We have fully considered the motion, the responses to it, the relevant portions of the record and the explanation given by Judge Carpenter for declining to recuse himself. We conclude that, for the reasons stated below, Judge Carpenter must step aside and another member of the Licensing Board Panel must be appointed to the Licensing Board for this proceeding.

## I

A. In February 1970, the Public Service Electric and Gas Company (Public Service) filed an application to build a two-unit nuclear facility on its Newbold Island site located on the Delaware River a few miles below Trenton, New Jersey. On November 1, 1973, while the application remained pending before a Licensing Board, Public Service amended its Preliminary Safety Analysis Report (PSAR) to reflect a relocation of the proposed facility to a site on Artificial Island and a change of its name from Newbold Island to Hope Creek. Although similarly on the Delaware River, Artificial Island is located a considerable distance from Newbold Island and is in Lower Alloways Creek Township, New Jersey, some 18 miles southeast of Wilmington, Delaware.[1]

In November 1974, pursuant to the Licensing Board's authorization the prior month,[2] the Atomic Energy Commission (this agency's predecessor) issued construction permits for the two Hope Creek units. Subsequently, a decision was made to complete Unit 1 alone and last year Public Service applied for an operating license for that unit. In response to the notice of opportunity for hearing on that application, the New Jersey Public Advocate filed an intervention petition and hearing request. On October 5, 1983, the Licensing Board granted both.

B. Approximately two weeks later, the Licensing Board for this proceeding issued a memorandum in which it provided the parties with certain information "regarding the qualifications and prior professional activities" of the Board members.[3] With respect to Judge Carpenter, the memorandum noted, *inter alia,* that during "the mid-60s" (when a member of the Johns Hopkins University faculty) he had performed "some dye studies" of Newbold Island for Public Service.[4] Further, in a separate statement attached to his resume accompanying the memorandum, Judge Carpenter pointed out that he had worked as a consultant to eight different electric utilities over a twenty-year period, "the general scope of the work being evaluation of the environmental effects of the operation of electricity generating plants, both nuclear and fossil fueled."

These disclosures prompted the Advocate's recusal motion.[5] In the view of the Advocate, the Newbold Island studies conducted by Judge Carpenter or the fruits of those studies might "well be tested" in the current proceeding.[6] Although explicitly disclaiming a belief that Judge Carpenter in fact had prejudged any issue that might be placed in controversy or had demonstrated actual bias against it or in favor of the applicants, the Advocate also maintained that:

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[1] *See Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), LBP-74-79, 8 AEC 745, 746-47 (1974). In a subsequent PSAR amendment, Public Service added the Atlantic City Electric Company as a co-applicant based upon the latter's 10 percent ownership of the site and the proposed facility. *Id.* at 747.

The Hope Creek facility shares the Artificial Island site with Public Service's Salem nuclear facility. The two Salem units went into commercial operation in 1977 and 1981, respectively.

[2] *Id.* at 768.

[3] October 18, 1983 Memorandum (unpublished) at 1.

[4] *Id.* at 2. In 1960, Judge Carpenter had developed a "very sensitive dye tracer technique." *Ibid.*

[5] No party has contested the timeliness of the November 18 motion. *Cf. Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-749, 18 NRC 1195, 1198-99 (1983). In any event, we are satisfied with the Advocate's explanation respecting why the motion had not been filed more promptly. *See* Affidavit of R. William Potter (November 21, 1983).

[6] Recusal Motion at 2.

> Dr. Carpenter's prior employment by PSE&G raises a reasonable apprehension that he may not be able to discharge his duties as a judge in this proceeding with complete impartiality. This apprehension is underscored by Dr. Carpenter's wide-ranging employment on behalf of electric utilities regarding matters of potential relevance to the Hope Creek application.[7]

At a November 22 prehearing conference, Judge Carpenter stated that he intended to respond in writing to the Advocate's motion.[8] He also requested that the counsel for Public Service furnish him with copies of any of his work product in the utility's possession.[9]

C. By letter of December 1, Public Service's counsel transmitted a number of documents to the Licensing Board representing all of Judge Carpenter's work product found in the utility's files. On the basis of these documents, it appears that Judge Carpenter's consultant relationship with the utility extended from at least 1967 to 1973 and embraced the following activities.

1. As previously noted (*see* note 1, *supra*), Artificial Island is the site of not only the Hope Creek facility but Public Service's two-unit Salem facility as well. In late 1967 or early 1968, Judge Carpenter and an associate (as "Pritchard-Carpenter, Consultants") performed a study of that site for Public Service, utilizing the Army Corps of Engineers' hydraulic model of the Delaware River.[10] The purpose of the study, which took approximately 10 days to complete,[11] was to evaluate the potential effects of thermal discharges into the river from the then proposed Salem facility.[12] It culminated in a fifty-seven-page report issued in July 1968.

2. In 1972, again in collaboration with his associate, Judge Carpenter performed a detailed model study, as well as a site tracer experiment, in connection with the then proposed (but later abandoned) Newbold Island site.

3. On October 24, 1973, J.T. Boettger, a Public Service official, sent an airmail-special delivery letter to Judge Carpenter.[13] The letter alluded

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[7] *Id.*, Affidavit of R. William Potter (November 18, 1983) at 3-4.

[8] Tr. 124.

[9] Tr. 125-26, 235-36.

[10] This mechanistic 750-foot scale model of the river and its branches and estuary is capable of simulating the major effects of currents and tides at any particular location.

[11] *See* Tr. 125.

[12] At the time the study was conducted, Salem had not yet received its construction permits. (The permits were issued in September 1968.) The study was not an integral part of the Atomic Energy Commission licensing process for that facility. Prior to the enactment of the National Environmental Policy Act of 1969, the AEC was not required to concern itself with thermal discharge matters. *See New Hampshire v. AEC*, 406 F.2d 170 (1st Cir.), *cert. denied*, 395 U.S. 962 (1969).

[13] By this time, Judge Carpenter had assumed a faculty position at the University of Miami. It is unclear whether his further undertakings for Public Service were in a strictly individual capacity rather than as a

*(Continued)*

to previous telephone conversations in which Mr. Boettger had informed Judge Carpenter of Public Service's intention to relocate the Newbold Island facility at the Salem (*i.e.,* Artificial Island) site. Judge Carpenter was supplied with information respecting the estimated thermal discharge from the relocated facility, the proposed location of the intake and discharge structures for that facility, and the actual location of those structures for the Salem facility. The letter then advised Judge Carpenter that:

> Commitments have been made to the A.E.C. Regulatory Staff for the submittal of additional information required by them to review the relocation of the project. In order to meet that commitment, we will need to know the behavior of the thermal plume in the bay by November 12, 1973.

Finally, the letter furnished him with "the design parameters to be considered for the analysis" and the assumptions that were to be incorporated into the study.

The record does not reflect the precise date upon which Judge Carpenter submitted his report. On December 8, 1973, however, Mr. Boettger wrote him to the following effect:

> Enclosed please find a copy of Appendix A and C of the Hope Creek Environmental Report. Appendix A is your report on the Thermal Discharge . . . .[14]
>
> We thank you for your timely response to the critical schedule requirements in relocating the Units to Artificial Island.
>
> As previously discussed, silting in the area adjacent to Artificial Island must be investigated in depth. We would appreciate any recommendations and comments you may have regarding the minimization of siltation. Restrictions upon jetty placement so as to not affect the thermal plume may be required.

Ten days later, Mr. Boettger sent yet another letter to Judge Carpenter with respect to the Hope Creek study, seeking his further analysis in aid of the utility's response to a specific AEC inquiry.

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principal in Pritchard-Carpenter, Consultants (which appears to have been Maryland-based). The record similarly does not disclose whether Judge Carpenter's associate in that firm also performed studies for Public Service in connection with the Hope Creek facility.

[14] Mr. Boettger's reference was to Appendix A in Amendment 7 to the Hope Creek Environmental Report — Construction Permit Stage, titled "Report on Thermal Discharge — Hope Creek Generating Station by Dr. James H. Carpenter." In the body of Amendment 7, Dr. Carpenter and his conclusions and recommendations were specifically mentioned several times. *See, e.g.,* ER Sections III.A.2 and IV.B.1; Supp. ER Section 6.4. It is worthy of note that the initial decision authorizing the issuance of construction permits for Hope Creek cited those three sections in accepting certain of Judge Carpenter's thermal discharge conclusions. LBP-74-79, *supra* note 1, 8 AEC at 758 n.62.

On December 27, 1973, Judge Carpenter submitted his invoice for "Consulting services — Hope Creek Station Discharge design and dilution study." The invoice reflected that he had spent two days, and an unidentified associate one-half day, on the project.

Finally, on December 28, 1973, Mr. Boettger transmitted several documents to Judge Carpenter, including the Draft Environmental Statement (DES) that had been prepared by the AEC staff for the Hope Creek facility. Judge Carpenter was asked to supply to Public Service by January 11, 1974, any comments he might have with respect to the DES.

D. On December 8, 1983, Judge Carpenter issued a memorandum in which he announced his tentative conclusion, based upon an examination of the documents supplied by Public Service and other materials contained in his own files, that no basis existed for his recusal.[15] Noting that his undertaking for Public Service had been "limited to evaluating the potential thermal effects of discharges of heated waters," and that none of the studies he had performed in that regard would be "in issue, or material to any fact in issue in this case," Judge Carpenter went on to state:

> I have not prejudged any issues in dispute in this case, nor do I have any bias with respect to the proper determination of those issues. My work performed over a decade ago for [Public Service] presents no objective basis for assuming the existence of any appearance of bias or prejudgment.[16]

Nonetheless, Judge Carpenter elected to defer an ultimate ruling on the motion in order to afford all parties an opportunity to review the materials discovered in his files. The Advocate was given 10 days within which to "amend his motion and supporting affidavit to present any additional information he deems relevant."[17] The other parties were provided an equal period to respond to any such supplemental filing.[18]

The Advocate chose not to amend his motion. Accordingly, on December 27 Judge Carpenter entered the order denying it for the reasons stated in his December 8 memorandum.

Upon receiving the required referral of the matter, we requested and obtained the views of the applicants and the NRC staff.[19] The applicants

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[15] The materials in Judge Carpenter's files were appended to the memorandum and, to the extent here relevant, consisted of (1) an undated draft of the Artificial Island study proposal submitted by Pritchard-Carpenter to Public Service; and (2) a November 17, 1967 draft of the proposal submitted by Pritchard-Carpenter to the Corps of Engineers for the rental of the Delaware River model. The documents in Public Service's files had been served upon the other parties by its counsel at the same time they were supplied to the Licensing Board.

[16] December 8, 1983 Memorandum at 3.

[17] *Id.* at 4.

[18] *Ibid.*

[19] *See* December 28, 1983 Order (unpublished).

state that there is no foundation for requiring Judge Carpenter to step aside but that they have "no way of predicting what conclusion an appellate tribunal ultimately reviewing decisions in this proceeding might reach, at least regarding the possible appearance of a conflict of interest on . . . [his] part."[20] They then observe that their interests would be critically affected were a Licensing Board decision in favor of the issuance of an operating license for Hope Creek later vacated on the ground that Judge Carpenter should have been disqualified. That being so, the applicants believe that "a cautious approach to the situation at hand dictates the replacement of [Judge] Carpenter with another member as to whom no question exists."[21] For its part, the staff concludes that Judge Carpenter did not abuse his discretion in denying the recusal motion.[22] Although not explicitly so stating, the staff apparently thus would have us affirm his December 27 order.

## II

A. In its decision in *South Texas* two years ago, the Commission made clear that Licensing Board members are governed by the same disqualification standards that apply to federal judges.[23] Our own *Midland* decision almost a decade earlier summarized the standards in these terms:

> [A]n administrative trier of fact is subject to disqualification if he has a direct, personal, substantial pecuniary interest in a result; if he has a "personal bias" against a participant; if he has served in a prosecutive or investigative role with regard to the same facts as are an issue; if he has prejudged factual — as distinguished from legal or policy — issues; or if he has engaged in conduct which gives the appearance of personal bias or prejudgment of factual issues.[24]

The current statutory foundation for the standards is found in 28 U.S.C. 144 and 455. Section 144 requires a federal judge to step aside if confronted with the "timely and sufficient affidavit" of a party to the effect that the judge has a personal bias or prejudice either against that party or

[20] Applicants' Response to Order Dated December 28, 1983 Regarding Recusal Motion by the Public Advocate (January 9, 1984) at 3-4.

[21] *Id.* at 4-5.

[22] Staff's Response to the Public Advocate's Request for Recusal of Judge Carpenter (January 13, 1984) at 13-14.

[23] *Houston Lighting and Power Co.* (South Texas Project, Units 1 & 2), CLI-82-9, 15 NRC 1363, 1365-67 (1982). *Cf. Duffield v. Charleston Area Medical Center, Inc.*, 503 F.2d 512, 517 (4th Cir. 1974). In invoking those standards, the Commission fulfills the Administrative Procedure Act mandate respecting the conduct of adjudicatory proceedings "in an impartial manner." 5 U.S.C. 556.

[24] *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-101, 6 AEC 60, 65 (1973).

in favor of an adverse party. And, to the extent here relevant, Section 455 provides:

> (a) Any justice, judge, or magistrate of the United States shall disqualify himself in any proceeding in which his impartiality might reasonably be questioned.
> (b) He shall also disqualify himself in the following circumstances:
> (1) Where he has a personal bias or prejudice concerning a party, or personal knowledge of disputed evidentiary facts concerning the proceeding;
> (2) Where in private practice he served as lawyer in the matter in controversy, or a lawyer with whom he previously practiced law served during such association as a lawyer concerning the matter, or the judge or such lawyer has been a material witness concerning it;
>
> * * * * *
>
> (e) No justice, judge, or magistrate shall accept from the parties to the proceeding a waiver of any ground for disqualification enumerated in subsection (b). Where the ground for disqualification arises only under subsection (a), waiver may be accepted provided it is preceded by a full disclosure on the record of the basis for disqualification.[25]

B. In his December 8 memorandum, Judge Carpenter did not specifically refer to any of the foregoing authorities. His familiarity with the governing standards may be inferred, however, from the fact that he correctly identified most of the questions upon which the outcome of the recusal motion hinged: (1) would the Licensing Board be called upon in this proceeding to pass judgment upon the validity of any of the studies that he conducted for Public Service in connection with the Hope Creek, Salem and proposed Newbold Island facilities; (2) had he prejudged any of the issues in controversy in this proceeding; (3) was he biased with respect to the proper determination of those issues; and (4) did his undertakings for Public Service "over a decade ago" present an "objective basis for assuming the existence of any appearance of bias or prejudgment?"[26] Because each of these inquiries received a negative response, Judge Carpenter declined to recuse himself.

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[25] Section 455 was substantially revised in December 1974. Pub. L. No. 93-512, § 1, 88 Stat. 1609. Prior to that revision, the Section read in full as follows:

> Any justice or judge of the United States shall disqualify himself in any case in which he has a substantial interest, has been of counsel, is or has been a material witness, or is so related to or connected with any party or his attorney as to render it improper, in his opinion, for him to sit on the trial, appeal, or other proceeding therein.

28 U.S.C. 455 (1968 ed.).

[26] Because Judge Carpenter was the target of the recusal motion, he was required to pass upon it by himself. *See Houston Lighting & Power Co.* (South Texas Project, Units 1 and 2), ALAB-672, 15 NRC 677, 683-85 (1982), *rev'd on other grounds,* CLI-82-9, *supra.* Nonetheless, with regard to any issue of law presented by the motion, he was free to solicit the advice of his colleagues or of the legal counsel available to the Licensing Board Panel. *Id.* at 685 n.19. Whether he did so is not known. All that appears from the record is that, at the November 22 prehearing conference, the Licensing Board Chairman expressed

*(Continued)*

1. As to the first three questions, there can be no quarrel with Judge Carpenter's conclusions. To begin with, the Advocate is the sole intervenor in the proceeding and none of his contentions admitted to the proceeding has anything to do with thermal discharges.[27] Thus, absent an accepted late-filed contention concerned with thermal discharges (an improbable occurrence), the Licensing Board will not have the task of examining and evaluating any of Judge Carpenter's studies, whether performed in connection with the Hope Creek facility or otherwise. And, as earlier noted (p. 16, *supra*), the Advocate took pains to make it clear that he was not asserting *actual* prejudgment or bias on the part of Judge Carpenter.

2. Equally apparent, however, is that 28 U.S.C. 455(a) and (b)(2) require Judge Carpenter's disqualification because of his prior involvement with this facility as a consultant in Public Service's employ.

a. As the Commission noted in *South Texas, supra,* the current Section 455(a) substituted for the prior subjective standard[28] "an objective standard for recusal, i.e., whether a reasonable person knowing all the circumstances would be led to the conclusion that the judge's impartiality might reasonably be questioned."[29] Thus, what must be decided in the application of that subsection is whether Judge Carpenter's prior association with Public Service might lead a fully informed reasonable person to question his impartiality in the present proceeding.

Had that association not involved the Hope Creek facility, and more particularly the construction permit application for that facility, it might well be that no such doubt could legitimately arise. The fact is, however, that the last project that Judge Carpenter undertook in his consultant relationship with Public Service not merely was directly tied to the Hope

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the view of the third member of the Board and himself that the recusal motion lacked merit and accordingly denied it. Tr. 127. Even though without prejudice to Judge Carpenter's own appraisal of the motion after receiving the material in Public Service's files, this action was in derogation of our *South Texas* instructions. 15 NRC at 685 n.19.

[27] *See* December 21, 1983 Special Prehearing Conference Order (unpublished). The admitted contentions relate to intergranular stress corrosion cracking of the recirculation piping; management competence; environmental qualification; and the environmental effects on cropland and groundwater of the salt deposition produced by cooling tower operation. (Although several of the Advocate's contentions were rejected by the Licensing Board, none of them dealt with thermal discharges.)

[28] The pre-1975 version of Section 455 called for recusal when, *in the judge's opinion,* continued participation would be improper because of his or her relationship to or connection with a party or attorney. *See* note 25, *supra.*

[29] 15 NRC at 1366, *citing Fredonia Broadcasting Corp. v. RCA Corp.*, 569 F.2d 251, 257 (5th Cir. 1978) ("Section 455(a) is a general safeguard of the appearance of impartiality and establishes a 'reasonable factual basis - reasonable man' standard").

The Commission also determined in *South Texas* that, as a general proposition at least, recusal under Section 455(a) must rest upon extrajudicial conduct. 15 NRC 1367. That limitation is unimportant here inasmuch as, in contrast to the situation in *South Texas,* the recusal of Judge Carpenter is sought exclusively on the basis of nonadjudicatory activities.

Creek construction permit application but, more than that, culminated in a work product that was cited by the Licensing Board in its decision in favor of the application. *See* note 14, *supra*. As we view the matter, it is simply idle to suggest that a reasonable person could not entertain the suspicion that, because he had played a role in the obtaining by Public Service of a construction permit for Hope Creek, Judge Carpenter might be partial to the current endeavor to acquire an operating license for it.[30]

b. We have seen that, even in the absence of a perception question, Section 455(b)(2) requires disqualification in circumstances where, for example, in private practice the judge served as a lawyer "in the matter in controversy."[31] The staff maintains that, inasmuch as Judge Carpenter did not serve the utility in the capacity of a lawyer, this provision does not come into play here.[32] It takes this position notwithstanding the earlier acknowledgement in its brief that "[p]ast work by a consultant would be analogous to past work by a lawyer on behalf of a client."[33]

Agreeing that such an analogy is appropriately drawn, we encounter no difficulty in further concluding that, in the instance of an adjudicator versed in a scientific discipline rather than in the law, disqualification is required if he previously provided technical services to one of the parties in connection with the "matter in controversy." Beyond doubt, this portion of Section 455(b)(2) was cast in terms of service as a lawyer simply because the members of the federal judiciary are lawyers and it was thus a natural assumption that any prior contact that a federal judge might have had with the "matter in controversy" would have been in a legal

[30] Even though perhaps not enough in themselves to require his recusal, Judge Carpenter's earlier undertakings for the utility in connection with the Salem and proposed Newbold Island facilities would not, of course, decrease the possibility that such a suspicion would arise.

In this regard, there is a marked difference between the present case and *Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear 1), ALAB-76, 5 AEC 312 (1972), in which the recusal of a Licensing Board member was likewise sought on the ground that he had previously had a consultant relationship with an electric utility seeking a nuclear license. In affirming the denial of the recusal motion in *Bailly*, we emphasized, *inter alia*, that that relationship had been with a different utility and, moreover, had not involved its license application. *Id.* at 313.

[31] Disqualification in such circumstances may not be waived. *See* 28 U.S.C. 455(e); *SCA Services Inc. v. Morgan*, 557 F.2d 110, 117 (7th Cir. 1977).

We are satisfied that the other bases for recusal under Section 455(b) are not even arguably relevant here.

[32] Staff Response, note 22 *supra*, at 14 n.6.

[33] *Id.* at 7-8. For its part, the applicants' four-page brief did not even refer to Section 455, let alone discuss the relevance of the various provisions of that Section to the situation at bar. All in all, we found that brief to be singularly unhelpful and, as such, far short of what we have a reason to expect from counsel whose views have been requested. True, recusal motions are matters of some delicacy from the standpoint of the parties to the proceeding. But that consideration did not relieve applicants' counsel of the obligation to brief the disqualification question more fully. In this connection, we might additionally have been provided with an explanation respecting the *legal* basis upon which we could order Judge Carpenter to step aside (as applicants would have us do) if (as applicants also maintain) there is nothing in the record to require his disqualification. *See* pp. 19-20, *supra*.

capacity. It is equally manifest to us that it would thwart the salutary purpose underlying the provision were it held inapplicable to a non-lawyer NRC adjudicator simply because his earlier involvement in the "matter in controversy" was necessarily in some other capacity.

The question remains, of course, whether the construction permit proceeding and the present operating license proceeding should be deemed to be the same "matter" for Section 455(b) purposes. The term is not defined in the statute and a canvass of the myriad judicial decisions concerned with the Section has provided no direct assistance to our resolution of that question.[34] There is, however, a hint in some of the decisions that the pivotal inquiry is whether the matter with which the judge had prior contact is "wholly unrelated" to that in adjudication before him or her.[35]

As the Seventh Circuit has observed, one of the Congressional objectives undergirding the 1974 revision of Section 455 was "to overrule the concept that close cases involving disqualification should be resolved on the ground that a judge had a 'duty to sit'."[36] Given this aim, as well as the legislative purpose of bringing the federal statutory disqualification standard into line with the "appearance of justice" standard for judicial disqualification set forth in the 1972 ABA Code of Judicial Conduct,[37] we think the "wholly unrelated" test appropriate for adoption here.[38] Accordingly, it is of no present moment that, strictly speaking, an operating license proceeding is not a continuation of the construction permit proceeding for the facility in question but, instead, is a separate proceeding initiated by its own notice of opportunity for hearing.[39] For, notwithstanding this consideration, the two proceedings have a decided

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[34] The decisions do establish the obvious: that a judge who was directly or indirectly professionally associated with a particular case while in private life is barred by Section 455(b) from sitting in judgment on the same case even if totally different issues are presented to him or her. *See, e.g., In re Rodgers*, 537 F.2d 1196 (4th Cir. 1976). It is therefore not crucial here that the issues presented in this operating license proceeding differ from those that Judge Carpenter addressed when serving as a consultant in connection with the construction permit proceeding.

Because no useful purpose would be served by discussing them individually, suffice it to note that we have examined with particular care each judicial decision involving Section 455 that was cited in the staff's brief. We uncovered nothing in any of them that might be thought to run counter to the conclusions we reach in this opinion.

[35] *See, e.g., National Auto Brokers v. Gen. Motors Corp.*, 572 F.2d 953, 958 (2d Cir. 1978), *cert. denied*, 439 U.S. 1972 (1979). *See also*, under the former version of Section 455, *Darlington v. Studebaker-Packard Corp.*, 261 F.2d 903, 906-07 (7th Cir.), *cert. denied*, 359 U.S. 992 (1959), *quoting Carr v. Fife*, 156 U.S. 494, 498 (1894) to the effect that a judge need not disqualify himself because he had previously represented some of the parties on matters "not connected" with the case on which he is sitting.

[36] *SCA Services*, note 31 *supra*, 557 F.2d at 113.

[37] *Ibid. See*, in particular, Canon 3(c) of the ABA Code to the effect that "a judge should disqualify himself in a proceeding in which his impartiality might reasonably be questioned . . . ."

[38] *Cf.* 5 C.F.R. 737.5(c)(4) concerning conflicts of interest involving former government employees.

[39] Section 189a of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2239; 10 C.F.R. 2.104, 2.105.

and undeniable relationship; not only is the same facility involved but, additionally, the ultimate purpose of each is identical: the eventual operation of the facility under NRC license.[40]

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For the foregoing reasons, we hold that Judge Carpenter should have recused himself and, accordingly, *direct that another member of the Licensing Board Panel be appointed to serve on this Licensing Board.* In taking this action, we stress that it is based *wholly* on the mandate of 28 U.S.C. 455(a) and (b)(2). Neither of those provisions requires a finding of actual bias or prejudgment on the part of Judge Carpenter and, in common with the Advocate, we expressly disclaim a belief that Judge Carpenter is, in fact, partial to one of the parties to the proceeding or has prejudged one or more of the issues that will be decided by the Licensing Board.[41] Rather, all that the statute requires is that Judge Carpenter's 1973 undertaking for Public Service with regard to the Hope Creek facility either (1) created an *appearance* of partiality (within the meaning of Section 455(a)) toward a party to this proceeding; or (2) was sufficiently related to this proceeding to bring the Section 455(b)(2) "matter in controversy" proviso into play. As seen, we have found both to be so.[42]

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

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[40] In these circumstances, it is hardly surprising that the doctrines of res judicata and collateral estoppel have long been applied in operating license proceedings. *See Alabama Power Co.* (Joseph M. Farley Nuclear Plant, Units 1 and 2), ALAB-182, 7 AEC 210, *modified on other grounds,* CLI-74-12, 7 AEC 203 (1974).

[41] *See In re Rodgers,* note 34 *supra* (recusal of a federal district judge under Section 455(b)(2) ordered despite the absence of a claim or finding of actual personal bias or prejudice against or in favor of a party).

[42] In *South Texas,* CLI-82-9, *supra,* the Commission observed that the proceeding was "now well along" and that the judge there involved had "acquired a valuable background of experience." 15 NRC at 1367. The situation here is quite different. The proceeding at bar is still in a very early stage; there has yet to be any evidentiary hearings on the Advocate's contentions. Thus, the Licensing Board Panel member assigned to replace Judge Carpenter should not be at a disadvantage.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Alan S. Rosenthal,** Chairman
**Dr. John H. Buck**

**In the Matter of** **Docket Nos. STN 50-519**
**STN 50-521**

**TENNESSEE VALLEY AUTHORITY**
**(Hartsville Nuclear Plant,**
**Units 1B and 2B)** **January 27, 1984**

On motion of the applicant following the cancellation of Units 1B and 2B of its proposed four-unit (1A, 2A, 1B and 2B) Hartsville Nuclear Plant, the Appeal Board terminates, with respect to those two cancelled units, the limited jurisdiction previously retained over this construction permit proceeding involving all four units.

## APPEARANCES

**Herbert S. Sanger, Jr., Lewis E. Wallace and James F. Burger,** Knoxville, Tennessee, for the applicant, Tennessee Valley Authority.

# MEMORANDUM AND ORDER

The Tennessee Valley Authority has advised us of the cancellation of Units 1B and 2B of its proposed four-unit Hartsville nuclear plant. On its motion, we therefore *terminate,* with respect to those two units, the limited appellate jurisdiction previously retained over this construction permit proceeding involving all four units.[1] *See Tennessee Valley Authority* (Phipps Bend Nuclear Plant, Units 1 and 2), ALAB-752, 18 NRC 1318 (1983), and *Duke Power Co.* (Cherokee Nuclear Station, Units 1, 2 and 3), ALAB-745, 18 NRC 746 (1983).

It is so ORDERED.[2]

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

[1] *See* ALAB-554, 10 NRC 15, 16 n.2, and ALAB-558, 10 NRC 158, 159 (1979). The retained jurisdiction was with regard to a single generic issue as to which an ultimate Commission determination has not as yet been reached: the environmental effects associated with the release of radioactive radon gas (radon-222) to the atmosphere as a result of the mining and milling of uranium for reactor fuel.

[2] This order has no application to Units 1A and 2A of the Hartsville facility. The appellate jurisdiction over the radon issue retained in ALAB-554 and ALAB-558, *supra,* thus is not affected insofar as those units are concerned.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Sheldon J. Wolfe, Chairman**
**Dr. George C. Anderson**
**Dr. Hugh C. Paxton**

**In the Matter of** | **Docket No. 50-482**
**(ASLBP No. 81-453-03-OL)**

**KANSAS GAS & ELECTRIC COMPANY, *et al.***
**(Wolf Creek Generating Station, Unit 1)** | **January 5, 1984**

The Licensing Board issues a memorandum and order which, *inter alia,* grants Intervenors' motion to add a contention out-of-time.

## RULES OF PRACTICE: NONTIMELY SUBMISSIONS OF CONTENTIONS

As to late-filed contentions, all five factors in 10 C.F.R. § 2.714(a)(1) should be applied by a Licensing Board, including the Appeal Board's three-part test for good cause.

## RULES OF PRACTICE: ADMISSIBILITY OF CONTENTION

While the basis of a contention must be set forth with reasonable specificity, the contention need not allege noncompliance with a regulation and need not specify how that regulation has been violated in the absence of any explanation by, as here, emergency planning authorities that determinations had been made in compliance with the regulation.

**RULES OF PRACTICE: ADMISSIBILITY OF CONTENTION**

It is not the function of a licensing board to reach the merits of a contention at the time the admissibility of a contention is being considered.

**RULES OF PRACTICE: ADMISSIBILITY OF CONTENTION**

A basis for a contention is set forth with reasonable specificity if the applicants are sufficiently put on notice so that they will know, at least generally, what they will have to defend against or oppose, and if there has been sufficient foundation assigned to warrant further exploration of the proposed contention.

# MEMORANDUM AND ORDER

**(Granting Intervenors' Motion to Add Contention and Witnesses)**

## Memorandum

On December 8, 1983, Intervenors Christy and Salava filed a Motion to Add Contention and Witnesses.[1] Therein the Intervenors requested leave to file out-of-time the following additional contention:

> The emergency planning zone for the plume exposure pathway does not include the Town of Waverly and the Waverly Unified School District No. 242 schools located in Waverly. The city and that part of the school district should be included in the plume exposure pathway emergency planning zone.

The Intervenors also requested that they be permitted to call one or more of the following as witnesses: the County Commissioners and

[1] The Notice of Opportunity for Hearing was published in 45 Fed. Reg. 83,360 (1980). Thereafter, in the Memorandum and Order of March 13, 1981 (unpublished), the Board admitted Ms. Christy and Ms. Salava as party-intervenors. In the Special Prehearing Conference Order of June 3, 1981 (unpublished), the Board admitted a similar contention of each intervenor which, in substance, questioned whether the evacuation plan was workable. The Memorandum and Order Ruling on Scope of Emergency Planning Issues dated July 28, 1983 (unpublished), ordered that, pursuant to agreement among the parties, a collation prepared and submitted by the Applicants on June 13, 1983 and as modified by that Order, would serve as the contentions at the hearing. The forthcoming hearing will be held in two stages — between January 17 and January 26, 1984, and between February 14 and February 23, 1984.

the Emergency Planning Coordinators for Franklin, Allen, Anderson and Lyon Counties.

In an Answer filed on December 23, 1983, the Applicants opposed Intervenors' Motion. In a letter of December 27, 1983, the Staff stated that it and FEMA had no objection to the granting of the Motion. In an Amended Answer filed on December 27, 1983, Applicants withdrew their opposition to that portion of the Motion seeking permission to call additional witnesses because Intervenors had advised "that the only two witnesses which Intervenors would call were the two individuals [Messrs. Sattler and McCracken] for whom subpoenas were issued by the Licensing Board on December 19, 1983."

## I. THE BALANCING OF THE FIVE FACTORS IN 10 C.F.R. § 2.714(a)(1) CALLS FOR THE ADMISSION OF THE LATE-FILED CONTENTION[2]

In *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983), the Commission determined that as to late-filed contentions, all five factors in § 2.714(a)(1) should be applied by a licensing board, including the Appeal Board's three-part test for good cause which had been set forth in ALAB-687, 16 NRC 460, 469 (1982).[3] We proceed to balance these five factors in light of the circumstances in the present case.[4]

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[2] These five factors set forth in § 2.714(a)(1) are as follows:

(i) Good cause, if any, for failure to file on time.

(ii) The availability of other means whereby the petitioner's interest will be protected.

(iii) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.

(iv) The extent to which the petitioner's interest will be represented by existing parties.

(v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding.

[3] The Appeal Board's three-part test for determining the good cause factor of a late-filed contention is whether it:

a. is wholly dependent upon the content of a particular document;

b. could not therefore be advanced with any degree of specificity (if at all) in advance of the public availability of that document; and

c. is tendered with the requisite degree of promptness once the document comes into existence and is accessible for public examination.

[4] Applicants' Answer relies heavily upon inapposite decisions in *Washington Public Power Supply System* (WPPSS Nuclear Project No. 3), ALAB-747, 18 NRC 1167 (1983) and *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-743, 18 NRC 387 (1983). At issue in those two cases was whether or not late-filed petitions for leave to intervene should have been permitted. The circumstances in the present case are entirely different. Here the Intervenors had timely petitioned for leave to intervene, had been admitted as parties, and their respective contentions upon emergency planning had been admitted as issues in controversy. At issue here is whether the Intervenors should be permitted to file out-of-time one additional contention questioning the adequacy of emergency planning.

With regard to factor (i), the Intervenors assert that they did not file this contention earlier because the City of Waverly and the part of the school district in question were previously included in the 10-mile Emergency Planning Zone (EPZ) as reflected in the Coffey County Plan of November 1982. Apparently agreeing that, prior to the revision of the Coffey County Plan in September 1983, which excluded that city and a portion of the school district, and prior to service of that document on October 5, 1983, the Intervenors had no cause to request the addition of this contention, the Applicants urge that no justification was offered for waiting more than two months before filing the instant motion. Thus, while apparently agreeing that the added contention was wholly dependent upon the content of the Coffey County Plan revised in September 1983 and could not therefore be advanced prior to the public availability of that document after October 5, 1983, the Applicants urge that the third part of the Appeal Board's three-part test has not been met — namely, that the contention had not been tendered with the requisite degree of promptness once the revised Coffey County Plan came into existence and was accessible for public examination.

Absent any statement by the Applicants that they had specifically notified the Intervenors that the revised plan had excluded the City of Waverly and a portion of its school district from the effective 10-mile EPZ, we conclude that the Intervenors filed their motion with a requisite degree of promptness. Beginning October 10, 1983, when we assume the Intervenors received the revised plan, they had to search the multipage document to determine what revisions had been effected. We do not think an inordinate amount of time passed before the Intervenors completed their combing of the revised plan and the filing of the instant motion on December 8, 1983. We are not persuaded by Applicants' argument that this two-month delay "severely prejudices" them. Our Order, *infra*, reflects: that discovery upon this contention shall begin immediately and be completed by January 30, 1984; that written direct testimony need not be submitted upon this contention until February 8, 1984; and that cross-examination will not begin prior to February 20, 1984, during the second stage of this hearing. We are certain that the Applicants have adequate personnel and resources to comply with our Order without undue hardship. We weigh this factor in the Intervenors' favor.

With respect to factors (ii) and (iv), the Applicants argue that the Intervenors baldly assert but do not support their assertions that "[i]f the motion is not granted, there will be no other way to protect the Intervenors' interests" and their "interests will not be represented by other parties in this matter." We have not been presented with any reason which would cause us to believe that, if we were to deny the instant Motion, FEMA and the NRC Staff would not represent the public interest as well as the Intervenors' interest in this matter. Again, if we were to deny this Motion, pursuant to 48 Fed. Reg. 44,332 (1983) (to be codified at 44 C.F.R. §§ 350.10, 350.15), the Intervenors could protect their interests by attending a FEMA public meeting in order to suggest improvements or changes in the State and related local plans, and, if necessary, by appealing the final FEMA decision. However, we weigh these two factors in favor of the Intervenors because the fact of the matter is that neither FEMA nor the NRC Staff oppose the motion. Moreover, we note that, without explanation, the revised County Plan deleted the City of Waverly and a part of its school district that had previously been included in the effective 10-mile EPZ. The Board is quite interested in hearing the reasons that prompted this deletion.

With respect to factor (iii), the Applicants argue that the Intervenors have only barrenly asserted that their participation would assist in developing a sound record upon this additional contention. We weigh this factor in favor of the Intervenors. First, as observed in note 4, *supra,* the cases cited in support of Applicants' argument are inapposite. Second, since numerous contentions of the Intervenors, as collated and agreed upon by the parties, have been admitted already as issues in controversy with respect to the basic issue of emergency planning, it ill-behooves Applicants to argue that there must be a showing that Intervenors' participation may reasonably be expected to assist in developing a sound record upon this additional emergency planning contention.

Finally, Applicants assert that the Intervenors failed to address factor (v) and thus tacitly admit the obvious potential for delay. The Intervenors were remiss, but not fatally so, since we do not anticipate that the proceeding will be delayed. (See discussion, *supra,* with respect to the first factor.) We weigh this factor in Intervenors' favor.

## II. THE BASES FOR THE ADDITIONAL CONTENTION HAVE BEEN SET FORTH WITH REASONABLE SPECIFICITY

Applicants cite *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), LBP-82-106, 16 NRC 1649, 1656 n.7 (1982) for the proposition that particularity requires not only an allegation of the fact of noncompliance with a specified regulation, but also sufficient detail to permit the Board to determine how the regulation is supposedly being violated. They assert that the Intervenors made no attempt to address the requirements of 10 C.F.R. § 50.33(g) or how these requirements have been violated.[5]

Under ordinary circumstances we would agree with the Seabrook Licensing Board. Here, however, without explication in both instances, the Coffey County Plan of November 1982 did include, but the revised Plan of September 1983 did not include, the City of Waverly and a part of its school district in the emergency planning zone for the plume exposure pathway. Absent such explication, it was not incumbent upon the Intervenors to allege noncompliance with the regulation and detail how the regulation had been violated. Moreover, pursuant to § 2.714(b), the Intervenors have given the bases (reasons) for their concern.[6] While Applicants have sought to refute these reasons in their Answer, it is not the function of a licensing board to reach the merits of a contention at this stage of the proceeding. *Mississippi Power & Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 AEC 423, 426 (1973). Finally, we conclude that the Applicants are sufficiently put on notice so that they will know at least generally what they will have to defend against or oppose, and that there has been sufficient foundation assigned to warrant further exploration of this additional contention. *Philadelphia Electric Co.* (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-216, 8 AEC 13, 20-21 (1974).

## Order[7]

1. The Intervenors' Motion to Add Contention and Witnesses is granted.
2. Discovery upon the admitted additional contention shall begin immediately and be completed by January 30, 1984.

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[5] Section 50.33(g) provides in pertinent part:

> Generally, the plume exposure pathway EPZ for nuclear power reactors shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius. The exact size and configuration of the EPZs surrounding a particular nuclear power reactor shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries.

[6] The reasons for their concern advanced by the Intervenors are that (a) the City of Waverly is only 11 miles from the Wolf Creek plant, (b) Waverly is almost due north of the plant, with south winds being the prevailing wind in Coffey County between March and November; and that (c) confusion would reign if Waverly and part of its school district were not included in the plume exposure pathway EPZ because parents would likely assume that a substantial number of their children, who live within the EPZ and attend school in Waverly, would be evacuated in the event of an emergency, and because residents of Waverly might assume that they are included in the evacuation plan.

[7] In a conference call on December 30, 1983, the Chairman read the contents of the Order to certain of the parties' counsel and requested that this information be relayed to other counsel who had not been available at the time of the conference call.

3. Written direct testimony upon this additional contention shall be submitted by February 8, 1984, and cross-examination will begin at some time in the last four days of the February session — *i.e.*, at some time between February 20 and 23, 1984.[8]

Judges Anderson and Paxton join but were unavailable to sign this Memorandum and Order.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Sheldon J. Wolfe, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 5th day of January 1984.

[8] *See* ¶ 10 of the Prehearing Conference Order of March 18, 1983 (unpublished).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Ivan W. Smith,** Chairman
**Dr. Dixon Callihan**
**Dr. Richard F. Cole**

**In the Matter of** | **Docket Nos. STN 50-454-OL**
**STN 50-455-OL**
**(ASLBP No. 79-411-04-OL)**

**COMMONWEALTH EDISON COMPANY**
**(Byron Nuclear Power Station,**
**Units 1 and 2)** | **January 13, 1984**

## LICENSING BOARDS: RESOLUTION OF ISSUES; DELEGATION TO STAFF

When governing statutes or regulations require a licensing board to make particular findings before granting an applicant's requests, a board may not delegate its obligations to the Staff. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 & 2), ALAB-298, 2 NRC 730, 737 (1975). The post-hearing approach should be employed only in clear cases — for example, where minor procedural deficiencies are involved (*Consolidated Edison Co. of New York* (Indian Point Station, Unit 2), CLI-74-23, 7 AEC 947, 951-52 (1974)), but not where the issue involved is a very extensive quality assurance reinspection program for which the Staff and the applicant have yet to agree on a full set of standards.

## OPERATING LICENSES: DENIAL

The remedy most responsive to the circumstances of this case where, though construction nears completion, the Board finds that the Applicant has not demonstrated that it has met its quality assurance obligations, and the remedy least harsh to the Applicant, yet still appropriate, is to decide the issue now. This permits the parties to test immediately on appeal the quality of the decision. To reserve jurisdiction and to postpone final decision, in face of the impending completion of construction, would impose unilaterally upon the parties, particularly the Applicant, the Board's own view of the facts, law and appropriate remedy. Unless Applicant could mount a difficult interlocutory appeal from such a determination (to postpone the decision), it would have been denied due process.

## RULES OF PRACTICE: *RES JUDICATA*/COLLATERAL ESTOPPEL

The Board avoided describing the reach of the denial of license on quality assurance grounds, as *res judicata* or collateral estoppel with respect to the quality assurance issues because neither concept, as ordinarily understood, neatly fits the unusual situation to be found in the continuum of a licensing proceeding with many aspects. The Board did not foreclose future proceedings on the quality assurance issue and had no jurisdiction to do so.

## EMERGENCY PLANNING: TRAFFIC TIME ESTIMATES; AVERAGE GENERIC SHELTERING VALUES

The Board did not agree with the Applicant that its intentional overestimation of assumed traffic times under adverse weather conditions in an emergency and intentional underestimation of average generic sheltering values of the structures in the EPZ are conservative. Therefore the Board required the Applicant to make realistic estimates of these factors. Any variance from realistic estimates of these factors could lead a decisionmaker away from actions affording radiological dose savings.

## TECHNICAL ISSUES DISCUSSED

Quality assurance program
Steam generator tube integrity

Flow-induced vibrations
Bubble-collapse water hammer
Occupational radiation exposure As Low As Reasonably Achievable (ALARA)
Linear hypothesis about health effects of radiation
Supralinear hypothesis about health effects of radiation
Severe accident analysis
Groundwater contamination
Groundwater velocity
Seismic design
Capability of faults
Strain gage tests
Emergency plans
Evacuation times
Average generic sheltering values.

## APPEARANCES

On behalf of Applicant, Commonwealth Edison Company:

**Alan P. Bielawski, Joseph Gallo, Michael I. Miller, Victor G. Copeland, M. Gwen Herrin, Bruce D. Becker, JoAnne G. Bloom, Mark C. Furse, Catherine T. Crowley, Michael R. Goldfein, Robert G. Fitzgibbons, Jr., and Lisa C. Styles, Esquires.**

On behalf of the Nuclear Regulatory Commission:

**Steven C. Goldberg, Richard J. Rawson, and Mitzi A. Young, Esquires.**

On behalf of the Intervenors, DAARE/SAFE, and Rockford League of Women Voters:

**Bryan Savage, Esq., David C. Thomas, Esq., Ms. Diane Chavez, Mr. Paul Holmbeck, Ms. Betty Johnson, Jane M. Whicher, Esq., Mrs. Patricia Morrison, Stanley E. Campbell, Joel Greenberg, Esq., and Allen Goldberg, Esq.**

Special Appearance: **Mrs. Ethel McGreevy.**

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# INITIAL DECISION
## (Operating License)

## I. SUMMARY AND COMMENTS

On November 30, 1978, Commonwealth Edison Company ("Commonwealth Edison," "CECO," or "Applicant") applied for a facility operating license for Byron Station, Units 1 and 2, two Westinghouse pressurized water reactors of 1120 megawatts electric output each. The Byron Station is located in Ogle County, Illinois about 17 miles southwest of Rockford, Illinois.

Petitions to intervene were filed by the League of Women Voters of Rockford, Illinois and jointly by the DeKalb Area Alliance for Responsible Energy and the Sinnissippi Alliance for the Environment (the "League" and "DAARE/SAFE," respectively; collectively, "Intervenors"). The Intervenors were admitted as parties to the proceeding along with the Applicant and the NRC Staff. Hearings were conducted in Rockford, Illinois during March through May and August 1983.

The issues heard, arising out of the Intervenors' contentions, pertained to the seismology of the Byron site, occupational radiation exposure (the ALARA principle), the consequences of severe accidents, steam generator integrity including the water hammer phenomenon, groundwater pathways for the release of radiation, and emergency planning.

The most important aspect of this decision is that we withhold authorization for an operating license for the Byron Station because of failures in the Applicant's quality assurance program. The application is, therefore, denied. If, however, operation is otherwise ultimately permitted, it would be in accordance with Board-imposed conditions relating to emergency planning and in consideration of various other commitments made by the Applicant.

### Quality Assurance

The Applicant demonstrated that it has, both on site and off site, an overall organization and several components within the corporate organization well-designed to provide quality assurance services in accordance with 10 C.F.R. Part 50, Appendix B. The quality assurance organizations have structural independence, and channels of reporting are at a sufficiently high level to preserve the independence of the quality assurance function.

Applicant has a very long record of noncompliances with NRC requirements, but it is also a very large nuclear utility. By the end of 1982 Applicant had been fined a total of $313,000 in civil penalties. For the most recent period evaluated by the NRC Staff, 1979-82, Applicant's civil penalty record was substantially better than the national average for other utilities. However, if $200,000 in additional civil penalties proposed in 1983 are finally assessed against Applicant, the favorable comparison with other utilities would be in doubt. As a tabulation of numbers, Applicant's record of noncompliances not involving civil penalties compares favorably with other nuclear plants in the NRC's Region III. The NRC Staff in its latest Systematic Assessment of Licensee Performance (SALP) rated Commonwealth Edison operating plants in the average range of Region III plants. The Board, however, found that none of these indicia — the dollar amounts of civil penalties, the number of other noncompliances and the SALP ratings — are reliable in assessing Applicant management's commitment to safety and quality assurance for the purposes of this proceeding.

Criteria for quality assurance programs are set out in Appendix B to 10 C.F.R. Part 50. Criterion I permits applicants to delegate to their contractors the execution of the quality assurance program but applicants retain responsibility for the program. Commonwealth Edison has freely availed itself of its prerogative to delegate, but failed in its responsibility to assure that its contractors carried out their delegated quality assurance tasks.

Every contractor doing safety-related work at Byron was required by the Applicant, upon the insistence of the Region III Staff, to conduct a large reinspection of their work because the Applicant had failed to assure that the contractors' quality assurance and quality control personnel were properly trained, qualified and certified. The results of these plant-wide reinspection programs have not yet been evaluated and, in some instances, the program is incomplete.

The Hatfield Electric Company is the electrical contractor at Byron. It has a long and bad quality assurance record there. The Board has no confidence in the quality of Hatfield's work. We recognize that a reinspection program could be an empirical demonstration that Hatfield's work is satisfactory, but we have no confidence in the reinspection program either. The NRC Staff was unable to assure the Board, as we requested, that the reinspection program will provide the necessary assurance that any deficiencies in Hatfield's quality assurance program will be identified and corrected. The Applicant made a weak showing bordering on default in response to the Board's order to present evidence respecting Hatfield. The Hatfield aspect of the proceeding alone requires that we deny the application for the Byron operating license.

The NRC Staff would have the Board leave it to the Staff to resolve post-hearing the problems surrounding Hatfield, but prevailing Commission law prohibits such a large delegation in contested issues.

Hatfield is not the only Byron contractor causing concern about the effectiveness of Applicant's control over its contractors. Systems Control Corporation, for example, is a supplier of safety-related electrical and control equipment. This contractor had a fraudulent and ineffective quality assurance program and the Department of Justice is investigating the matter. Reliable Sheet Metal, the heating, ventilating and air conditioning contractor needed a 100 percent independent over-inspection of its work. The piping contractor, Hunter Corporation, failed to maintain a reliable method of identifying nonconforming conditions, preferring instead to resolve nonconformances during a final walk-down. An effective reinspection of Hunter's work is essential because the sloppy documentation cannot assure reliable control and trending of faulty work. Quality assurance problems with other contractors surfaced during the hearing but were not thoroughly explored.

Worker allegations against various contractors constituted a large part of the quality assurance litigation but produced relatively little in reliable results. Most worker allegations were not substantiated despite reasonable inspection efforts by Region III. Occasionally at the hearing the Board was faced with unpersuasive worker allegations countered by unconvincing explanations by witnesses for the Applicant. Most of the reliable evidence adverse to Applicant was produced through Region III inspections.

Despite our strong criticisms of Applicant's quality assurance performance, and our firm reaction, we have not concluded that Applicant is institutionally unable or unwilling to maintain a reliable quality assurance program. A better explanation, we believe, is that Applicant began to deal effectively with its contractors' problems too late, but is catching up. Finally, although the Board has found that there were widespread failures in the contractors' quality assurance programs, we have not found, nor has the NRC Staff reported, widespread hardware or construction problems. But we are not confident that such problems would have been discovered.

**Steam Generator Integrity**

The Board's awareness of a long-standing weakness in the steam generators of pressurized water reactors was enhanced by the litigation of contentions charging violations of one of the important barriers to the dispersal of radionuclides from steam-electric generating stations. Additionally, though not litigated, is the economic loss incurred as a consequence of the steam generator problem. Westinghouse, the vendor of the Byron steam generators, through a rather extensive research and development program, has proposed a number of modifications of its generators intended to mitigate the several specific problems. The modifications affect the design and materials of construction and procedures for maintenance and operation. Although extensive field testing of these proposed modifications is absent, the Applicant and the Staff have enthusiastically supported them and the former has committed itself to their adaptation. The Board recognizes the potential of the proposals as at least a partial solution to a very significant problem of the industry.

For now at least, Applicant's proposals, when implemented, will provide reasonable assurance that the Byron steam generators will maintain their integrity. Applicant's commitments to the proposals form an essential basis for our favorable findings on the relevant contentions.

The Board is not necessarily convinced that all serious weaknesses in steam generators have been identified and we sense a spirit of optimism with a twinge of overconfidence. Further investigations along the lines of the recent activities should be supported and encouraged.

### Water Hammer

In August 1981, at the KRSKO nuclear power plant in Yugoslavia, pipe damage was discovered which indicates that in July of the same year, a "bubble-collapse water hammer" of considerable force took place in one of the pipes which feeds water into one of KRSKO's Westinghouse Model D steam generators: Apparently, steam leaked back from the steam generator, through faulty check valves, and into the feedwater bypass line; then with steam in the bypass line, cold water was fed into the line by the auxiliary feedwater pumps, which were being tested in July 1981. Some of the steam must have been trapped in one or more "bubbles" in the cold water; the bubbles would have condensed rapidly, causing slugs of water to rush into the void left by the condensing steam and thereby produce that sudden increase in pressure called "water hammer."

Despite the water hammer, the systems affected by it continued to function without impairment. But the water hammer did have enough force to move pipes, and even to make a bulge in one. Such water hammers can have enough force to rupture pipe. DAARE/SAFE Contention

9(c) says that the Applicant "should be required to demonstrate that a similar event will not occur at Byron."

Westinghouse has made several recommendations on how to prevent KRSKO-type water hammer at Byron. The Applicant affirms that it will follow all of these recommendations.

Their efficacy will be tested before the plant goes into operation. Implementation of these recommendations will not completely eliminate the chance that a KRSKO-type water hammer will occur at Byron, but that implementation will make such a water hammer very unlikely. The Board therefore concludes that there is reasonable assurance that there will be no KRSKO-type water hammer at Byron.

## ALARA

The Intervenors contend, with a number of alleged deficiencies as bases, that the Applicant will be unable to operate the Byron Nuclear Station with occupational radiation exposure as low as is reasonably achievable. Some of the bases are: (1) inadequate equipment design; (2) improper translation of small radiation doses into health effects; (3) a health physics staff deficient in equipment and in both quantity and quality of personnel which is incapable of adequate radiation monitoring and record keeping; and (4) the use of temporary employees to distribute the exposure load.

In the course of the hearing these items were addressed by witnesses sponsored by all parties. Westinghouse reported improved designs to reduce accumulations of foreign radioactive material and to lessen time requirements for maintenance. The Board was confronted by the testimony of two individuals of substantial standing in the radiation health effects community whose evaluations are strongly at variance with each other. The Board adopted the linear hypothesis relating radiation dose to resulting health effects over the concept of relatively more severe effects from the same dose experienced at a lower rate. Both the Applicant's corporate and site organizations responsible for radiation protection were described as to function, staff and equipment. The Board found them adequate. Absent failures in the prescribed security, training, monitoring and record keeping of transient or "temporary" employees, the Board considered the practice to be acceptable.

## Severe Accidents

The question of severe ("Class 9") accidents was entered into this proceeding through assertions by the Intervenors of the inadequacies of the Reactor Safety Study, WASH-1400 (the Rasmussen Report). In their case the Intervenors overemphasized a popular interpretation of the conclusions of the [WASH-1400] Independent Risk Assessment Group, NUREG/CR-0400 (the Lewis Report). Whereas the Assessment did fault some parts of the Study, including the statistical analysis and its presentation and the error bounds on the accident probabilities, it did, nonetheless, pay tribute to the overall effort. The Assessment found, for example, that the Study was a substantial advance over earlier attempts at its goal. The Executive Summary to WASH-1400 was judged not to emphasize sufficiently the consequences of reactor accidents and the uncertainties in the calculation of their probabilities.

The Board concluded that the Staff and Applicant have reviewed and analyzed possible severe accidents and their consequences in a manner consistent with the Commission's regulations and policies.

**Groundwater Pathway**

In a consolidated contention on groundwater contamination, the League of Women Voters claimed that the Staff and the Applicant had not adequately characterized the groundwater system at Byron and therefore had probably underestimated the velocity with which radionuclides which might be released into the groundwater system by certain possible accidents would travel to points where humans use the water. The League argued that the formula which the Staff and the Applicant used to estimate that velocity could not be applied to the highly fractured bedrock at Byron because the formula, Darcy's equation, had been developed in studies of uniform porous media. The League also argued that a study of the migration of cyanide from a chemical waste dump near the Byron site showed that contaminated water in the Byron bedrock could travel with a velocity an order of magnitude greater than the velocity the Staff and the Applicant had calculated.

The League contended that until the system of fractures at Byron had been adequately studied, there could be no confidence in the Staff's and the Applicant's analyses of the consequences of those accidents which could contaminate the groundwater system. The higher the velocity of the contaminated water, the less time there would be for radionuclides to decay to safe levels, or for the spread of the contaminated water to be stopped.

The Board concluded that the Staff and the Applicant had adequately characterized the groundwater system at Byron. The Applicant's investigations of the geology at Byron were thorough, and they revealed circumstances which even the League's expert witness said would permit the use of Darcy's equation. The alternative means which the League recommended for estimating velocity would entail a cost in money and time all out of proportion to the benefits those means might bring. Last, the cyanide migration study is no proof countering the Staff's and the Applicant's results, for there are many strong indications that much of the cyanide migrated by surface routes.

### Seismic Analysis

League of Women Voters alleges that there exist serious seismic-related site deficiencies discovered subsequent to the issuance of the construction permit. The League asserts that because of a lack of information regarding the cause of earthquakes in northern Illinois, the Applicant should be required to perform strain gage tests on faults cutting basement rock. The League also contends that it is not known if a recently discovered fault (found after the CP issuance) is a capable fault and that neither the Safe Shutdown Earthquake (SSE) nor the Operating Basis Earthquake (OBE) are sufficiently conservative. Evidence was received on each of the issues and the Board found for the Applicant.

The strain gage application recommended as necessary by the League was found to be beyond the current state of technical feasibility and, even if strain gages could be installed, techniques for translating the strain measurements to predicting faults are beyond the current state of knowledge. The lack of observed movement in the zone in at least the last 125,000 years is further reason to question the need for such measurements. As to the presence or absence of capable faults in the plant vicinity, the evidence was substantial and convincing. There is no evidence of a capable fault in the vicinity of the Byron plant. In fact, there are no known capable faults east of the Rockies in the United States. Regarding the SSE and OBE, based on the evidence presented, the Board finds that the seismic design of the Byron plant is sufficiently conservative and in accordance with the requirements of 10 C.F.R. Part 57 and 10 C.F.R. Part 100, Appendix A.

### Emergency Planning

Emergency planning for the Byron Station was in relatively early stages at the time of the hearing. Most of the Intervenors' concerns about the Byron emergency plans were resolved or deferred by a stipulation in which Applicant committed to many emergency planning actions following the Federal Emergency Management Agency final report. The

Board approved the stipulation and, at the Board's request, the Commission extended the Board jurisdiction to hear disputes over Applicant's commitments even after an initial decision and any full-power operation.

As to those issues which were litigated, high-ranking Illinois emergency and disaster agency officials appeared at the hearing and, together with the cognizant Commonwealth Edison employees and Federal witnesses, provided convincing assurances that careful attention is being paid to the Byron emergency plans and that the plans will satisfy regulatory requirements.

However, the Board was constrained to impose three conditions with respect to protective measures during an emergency. We have required the Applicant to clarify or amend its Evacuation Time Study to identify employers in the emergency planning zone with extended shutdown times — a rather minor adjustment.

We have also taken issue with Applicant on the use of so-called conservatisms in emergency planning and have imposed corrective conditions. The Evacuation Time Study intentionally overestimates assumed traffic times under adverse weather conditions in an emergency and intentionally underestimates average generic sheltering values of the structures in the Byron EPZ. Neither variance from realism is conservative.

Applicant's witnesses could not explain why overestimating the traffic time assumptions for evacuations would be conservative, nor can the Board identify any such conservatism. Incorrect evacuation times could lead a decisionmaker away from actions affording radiological dose savings. Underestimating the average sheltering value of the structures near the Byron Station is a reflection of a policy of the State of Illinois emergency planning officials to favor evacuation over sheltering no matter how slight the potential dose savings. Considering the risks to the public in any evacuation, this Board does not believe that the Illinois policy is a good one, but that consideration is a matter beyond the Board's jurisdiction. It is within our jurisdiction, however, to require Applicant to provide accurate evacuation time assumptions in the event that emergency planning officials choose to use them. We have done so.

## II. FINDINGS OF FACT AND CONCLUSIONS OF LAW

### A. Steam Generators

#### *1. Introduction*

A-1. The Byron Nuclear Power Station is comprised of two pressurized water reactors and auxiliary equipment necessary to generate electrical energy in steam-driven turbines. The thermal energy derived in the reactor core from the uranium fission process is transported to the turbine by two fluid circuits, the primary and secondary reactor coolants, separated by the steam generator tubing. After traversing the turbine, the steam must be condensed by removal of a not insignificant quantity of heat before this secondary coolant can be recycled to the steam generator. This heat removal from the discharge from the turbine, through the condenser tubes, is by still another fluid circuit called the circulating water system and is constituted by the condensers, the cooling towers, a capability for adding and discharging water and for adding chemicals, and a storage reservoir together with necessary pumps, controls, etc., for operation. The primary and secondary coolants operate at quite high temperature and pressure.

A-2. A steam generator is simply a heat exchanger. It is comprised of a number of thin-walled tubes through which hot water, the primary coolant, flows. With use the primary coolant becomes radioactively contaminated, mainly with fission products from leaky fuel pins.

A-3. These tubes are contained within a vessel into which the secondary coolant is pumped to be vaporized to drive the turbine. Two of the contentions in issue relate to steam generators in the following ways. A combination of physical, chemical and metallurgical actions have individually and collectively contributed to a history of failure of the tubes in generators. These failures have lead to leaks from the primary to the secondary side of the tubes with concomitant dispersal of radionuclides throughout the secondary coolant and its flow circuit. A remedial measure to this highly undesirable situation is to first identify the faulty tubes and to take them separately out of service by plugging their ends. There is, thereby, a decrease in the effective heat transfer area.

A-4. A particular challenge has to do with potentially destructive forces arising in a natural physical phenomenon called a "water hammer." A water hammer is the production of pressure within a moving liquid as its speed is suddenly altered. The collapse of a steam bubble, for instance, allows a surge of liquid which, when stopped, produces such a pressure. The forces resulting from the pressure are potentially damaging to the containing structure.

A-5. Another issue involving the nuclear generation of steam concerns radiation exposures possibly incurred by operating, maintenance and management personnel during the course of normal operations, of necessary repairs, and during and following some unanticipated event or accident. It is the policy of the Nuclear Regulatory Commission to require nuclear plant operations to be conducted so that this exposure to onsite persons is kept as low as reasonably achievable (ALARA). This means that not only is any discharge of radionuclides carefully monitored and limited but also that control be exercised over the access of employees to areas where the structure may itself have become radioactive under irradiation during operation. Adequate and proper radiation detection instrumentation coupled with strict administrative practices are necessary, and usually can be adequate, to achieve the ALARA goal.

A-6. The contentions, in whole or in part, related to steam generators, posed by the Intervenors and admitted for litigation are:

*League Contention 22* (Steam Tube Integrity):

An extremely serious problem occurring at other plants such as Consumers' Palisades plant and C.E.'s Zion plant, and likely to occur at C.E.'s Byron plant, is presented by degradation of steam generating tube integrity due to corrosion-induced wastage, cracking, reduction in tube diameter, and vibration-induced fatigue cracks. This affects, and may destroy, the capability of the degraded tubes to maintain their integrity, both during normal operation and under accident conditions, such as a LOCA or a main steam line break. The Commission Staff has correctly regarded this problem as a safety problem of a serious nature, as evidenced both by NUREG-0410 and the Black Fox testimony cited above [sic]. As a result of this serious and unresolved problem, the findings required by 10 C.F.R. §§ 50.57(a)(3)(ii) and 50.57(a)(6) cannot be made.

*DAARE/SAFE Contention 9(c)* (Steam Tube Integrity):

Intervenors contend that there are many unresolved safety problems with clear health and safety implications and which are demonstrably applicable to the Byron Station design, but are not dealt with adequately in the FSAR. These issues include but are not limited to:

> Steam generator tube integrity. In PWRs steam generator tube integrity is subject to diminution by corrosion, cracking, denting and fatigue cracks. This constitutes a hazard both during normal operation and under accident conditions. Primary loop stress corrosion cracks will, of course, lead to radioactivity leaks into the secondary loop and thereby out of the containment. A possible solution to this problem could involve redesign of the steam generator, but at FSAR, Section 10.3.5.3 the Applicant notes its intent to deal with this as a maintenance problem, which may not be an adequate response given the instances noted in Contention 1, above [sic].

*DAARE/SAFE Contention 9(a)* (Water Hammer):

During recent startup tests at the KRSKO plant in Yugoslavia, which has steam generators which are similar in design to those at Byron, the plant experienced a bubble collapse water hammer event in the feedwater bypass line. Applicant should be required to demonstrate that a similar event will not occur at Byron.

*League Contention 111.B(1)* (ALARA in Steam Generators):

C.E. has not met the requirements of NEPA and the Regs, including but not limited to 10 C.F.R. §§ 50.34(a) and 50.36(a) because C.E. has not adequately monitored and provided a design base for the Byron plant which will keep radiation levels as low as achievable as required for operation of the plant to protect the health and safety of the public. To keep radiation levels as low as achievable, C.E. should provide and utilize:

B. More accurate calculation of design doses which can be accomplished by utilizing information from the improved monitoring suggested above and also by:
   (1) Providing for and constant update and replacement of equipment and analysis to respond to new experimental and analytical results. Byron was licensed for construction, for example, when some (including C.E.) asserted improperly that there was a threshold to radiation effects;

*League Contention 112(a)* (ALARA in Steam Generators):

C.E. has not met the requirements of NEPA and 10 C.F.R. Part 20 because it has not adequately assessed the effect of radiation on plant workers and provided a design base for the Byron plant which will provide radiation levels as low as achievable. To keep radiation levels as low as achievable there is a need for better use of preventive measures to reduce radiation, including neutron, exposure levels to regular plant personnel and transient workers. These include but are not limited to:

(a) Plant designs for reducing amount of radiation exposure which take into account new evidence on low levels of radiation which were not considered in design of the plant.

A-7. A steam generator is comprised, in major part, of a few thousand inverted-U tubes each about 60 feet in length. Conway, ff. Tr. 4126, at 7. The tubes are of relatively thin wall to facilitate heat transfer.

A-8. The tubes of a steam generator provide a barrier between the radioactive materials in the primary coolant, largely arising from leaks in the cladding of fuel pins, and the normally uncontaminated steam supply to the turbine.

A-9. As a requisite for the issuance of a license to operate a nuclear steam supply system, it is incumbent upon an applicant to show reasonable assurance that the operation will not endanger the health and safety of the public. 10 C.F.R. 50.57(a)(1). Specifically in this context, conformity shall be shown to the Commission's General Design Criteria 14, 30, 31 and 32 of Appendix A of 10 C.F.R. Part 50.

A-10. In a like manner General Design Criterion 4, 10 C.F.R. Part 50, Appendix A, demands that the reactor systems be protected from dynamic effects, such as a water hammer, and 10 C.F.R. 20.1(c) requires that personnel exposure be kept as low as reasonably achievable.

A-11. A history of one or more malfunctions in various forms and degrees of severity within steam generators at pressurized-water nuclear power stations has led the NRC to include steam generator integrity among its list of continuing generic problems termed "Unresolved Safety Issues" (USI), thereby making it a candidate for especial attention in research and development programs. The subject of steam generator tube integrity has been designated USI: A-3. Intervenors have used that classification to support their position in these contentions. The Commission, however, through its Appeal Boards has taken a position on the matter whereby the Staff is to make clear in its Safety Evaluation Report (Staff Ex. 1) its "perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny." *Gulf States Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977) at 775. The Staff has now met that requirement. Staff Ex. 1, at 5-19 through 5-22.

A-12. Further to the question of unresolved safety issues, it is noted that the Staff has devoted considerable attention to these matters in its Safety Evaluation Report. Staff Ex. 1, Appendix C. Specific attention is given to thirteen of the issues which have at least a potential bearing on the Byron Station.[1]

A-13. In the litigation of these contentions the Applicant presented fourteen witnesses, all qualified scientific and technical personnel. John C. Blomgren and Lawrence D. Butterfield, of Commonwealth Edison, addressed minimization of steam generator tube degradation and modifications directed to the flow-induced vibration phenomenon, respectively. Mahendra R. Patel, of Westinghouse Electric Corporation, addressed the "leak-before-break" principle and steam generator tube plugging criteria. Six additional Westinghouse persons testified on generator integrity: Daniel Malinowski on inspection measures used to detect steam generator tube degradation, Michael J. Wootten on water chemistry, Lawrence Conway on design changes in steam generators at Byron that both enhance resistance to tube degradation and minimize occupational radiation exposure, and Thomas F. Timmons on flow-induced vibration phenomenon; Wilson D. Fletcher provided an overview of the steam generator tube integrity issue; and Michael Hitchler quantitatively assessed the probability of steam generator tube ruptures under various conditions. R.W. Carlson of Westinghouse, Richard Pleniewicz of Commonwealth Edison, and Kenneth J. Green of Sargent & Lundy testified on water hammers; Rodolfo Paillaman of Ebasco Services, Inc. addressed the pre-service inspection of the steam generator tubes. J.R. Van Laere discussed the effect of generator modifications on the ALARA goals.

[1] Among these thirteen issues are three which are included in the contentions litigated in these proceedings. They are water hammer, steam generator tube integrity and seismic design criteria. Applicant Ex. 1, at C-7 through C-26.

A-14. The NRC Staff presented the testimony of Aleck W. Serkiz on water hammer events and of a panel composed of Jai Raj N. Rajan on flow-induced vibration phenomenon and the probability of tube rupture under various conditions, Ledyard B. Marsh on tube degradation, Louis Frank on water chemistry and in-service inspections, and Conrad McCracken on steam generator design and secondary coolant water chemistry. The combined experience of the members of the Staff panel in nuclear engineering and related subjects is more than 80 years.

A-15. For the League, Dale G. Bridenbaugh, a nuclear engineer with more than 25 years experience and President of MHB Technical Associates, testified on various aspects of the steam generator tube integrity issues. No Intervenor evidence was presented on water hammers. The testimony of K.Z. Morgan made reference to past experiences with radioactive accumulations in steam generators and their bearing on the ALARA principle.

## 2. *Steam Generator Tube Integrity*

A-16. The historical degradation experienced by Westinghouse steam generators in non-accident operation has taken several forms including tube-wall thinning, pitting, cracking, intergranular attack, and tube wear. Malinowski, ff. Tr. 4126, at 15-21.

A-17. An extended analysis of steam generator tube degradations and their consequences, in preparation by Science Applications, Inc. (SAI) under contract with the Commission[2] (Intervenors Ex. 9), is often referred to in this record as the "SAI Report." This report presents a value-impact analysis of twelve regulatory requirements related to steam generators, which are under consideration by the NRC Staff for imposition on the operators of pressurized water reactors.[3] Tr. 4573 (Marsh). Of the twelve candidate requirements posed in this analysis, four are considered to have potential for effectively improving the outlook for more successful operation of pressurized water reactors. Those four are: inspection of the secondary side of generators for loose parts, overall in-service inspection, the eddy-current method for tube examination, and the secondary coolant chemistry coupled with in-service inspection of the secondary condenser. Intervenors Ex. 9, at ES-4 and ES-5. These four topics were extensively discussed in this proceeding.

[2] A "draft final report" of this effort is Value-Impact Analysis Recommendations Concerning Steam Generator Tube Degradation and Rupture Events, December 23, 1982, prepared by Science Applications, Inc.

[3] *See* Intervenors Ex. 9, Section IV for details of these proposals.

A-18. Four ruptured-tube events in domestic Westinghouse steam generators at Point Beach 1, Surrey 2, Prairie Island 1 and Ginna have been reported.[4] The Staff examined the operator response and the radiological consequences in each of these instances. In the first three the consequences were less than the expectations from a design basis generator tube rupture accident. The results from Ginna are characterized as "were as expected" without quantification. Tr. 4801-02 (Marsh).

A-19. Although the Intervenors' witness, Mr. Bridenbaugh, had no reason to disbelieve the evaluation by the Staff of the radiological effects, he, too, was uncertain of the meaning of the statement about Ginna. He had made no independent analysis. Tr. 6495-96 (Bridenbaugh).

A-20. The Staff has analyzed the effect of generator tube rupture on such low-probability events as main steam-line and feedwater-line breaks. When a break occurs outside the containment there may be a loss of primary coolant directly to the environment. Under action by emergency core coolant, the core would remain covered. Even smaller consequences result when the break is inside the containment. Marsh, ff. Tr. 4473, at 4.

A-21. Concurrent rupture of tubes and a cold-leg loss of coolant has been examined analytically and a few related experiments have been performed at the Idaho Nuclear Engineering Laboratory, assuming various conditions of departure. None led to a core melt. *Id.* at 5.

A-22. These computational analyses may lead to a realistic statistical in-service inspection program to counter the occurrence and consequences of concurrent steam generator failure and large-scale accidents. *Id.*

A-23. A typical present-day Westinghouse steam generator contains about 4500 inverted-U tubes of 60-foot overall length, 0.75 inch in diameter with a 43-mil-thick wall. The ends of the tubes are expanded into a tube sheet.[5] The material is type 600 Inconel, a nickel-chromium-iron alloy. The tubes are supported laterally by ¾-inch-thick support plates spaced vertically every 3 to 5 feet. The surface area of the tubes in a generator is on the order of $10^4$ $ft^2$. Conway, ff. Tr. 4126; Patel, ff. Tr. 4126.

A-24. Internal to the tubes is the high-temperature, high-pressure primary coolant; external to the tubes is the secondary coolant which is

[4] Additional information on these events appears at Paragraph A-51, *infra*. *See also* Frank, ff. Tr. 4473, at 8.

[5] A discrepancy exists in the testimony: the tubes are expanded into the tube sheet (Conway, ff. Tr. 4126, at 12) and welded into the tube sheet (Conway, ff. Tr. 1309, at 6).

vaporized to the steam necessary to propel the turbines. The nominal outwardly directed pressure differential across a tube wall is 1250 psi. The burst pressure of new 43-mil-thick tubing is 10,000 to 12,000 psi. Tr. 4393 (Patel).

A-25. Over recent years, Westinghouse steam generators have experienced an evolutionary development with progressive changes intended to surmount the structural, material and operational weaknesses which surfaced with use.[6] Correspondingly, the designations of the models have changed. The most recently designed model at the time of the hearing was D5, the preceding one was D4, etc. Byron Unit 2 is equipped with Model D5; Byron Unit 1 with D4. Conway, ff. Tr. 4126, at 8, 14. These weaknesses have been essentially confined to the secondary side of generators, *i.e.,* there has been no erosion or wear inside tubes. Tr. 4790 (McCracken).

A-26. Although observation of radionuclides in the secondary coolant is a *prima facie* indication of tube damage, the principal method for monitoring steam generator tube wear and damage utilizes measurement of eddy currents induced in the tube by alternating magnetic fields as a probe traverses the tube internally. Four magnetic fields are impressed on the tube simultaneously, allowing differentiation of signals arising from expected sources, such as support plates, from those truly indicative of the tube itself. Obtainable are data on the thickness of the tube, on the dimensions of a defect, and within limits, the composition of the defect. Calibration of the apparatus is by observation of a hole in the material drilled to specified dimensions. Additionally, direct examination of damaged tubes removed from generators has verified earlier eddy-current determinations. The method is sufficiently sensitive to detect a decrease in wall thickness in the range of 20 to 40 percent depending on the type of degradation, its extent and location. Malinowski, ff. Tr. 4126, at 15. Intervenors agree that the multi-frequency eddy-current

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[6] Subsequent to the closure of the record in this proceeding, the Board received from the Applicant a report on modifications proposed by Westinghouse of the current generation of steam generators. The report, titled "Independent Evaluation of Proposed Modifications of Westinghouse D4, D5 and E Steam Generators," CSGORG-002, dated July 29, 1983, was developed by the Counterflow Steam Generator Owners Group, a technical committee, independent of the Byron Station, composed of representatives of utilities and consulting engineering firms both domestic and overseas. As a goal "[t]he purpose of this report is to issue the group's evaluation of the problem definition and the acceptability of proposed solutions to problems related to the implementation of design modifications and full-power operation of the Model D4, D5 and E steam generators."

The Board is also aware of the issue by the NRC Staff of "Safety Evaluation Report Related to the D4/D5/E Steam Generator Design Modifications," NUREG-1014, dated October 1983. The Board noted a statement in the November 21, 1983 letter of transmittal to this Board from T.M. Novak that the "report is consistent with the Staff's testimony during the Byron hearing that the proposed D4/D5 modification is acceptable."

method is a significant improvement and may prove to be adequate. Tr. 6461 (Bridenbaugh).

A-27. Recent changes effected in the design, construction and operation of Westinghouse steam generators are intended to improve their performance, to decrease losses due to outages and the removal of defective tubes from service, and to lessen the occupational radiation exposure to maintenance personnel.

A-28. Early observations were made of caustic stress corrosion cracking of generator tubes, which had resulted from the absence of careful control of the concentration of phosphates added to the secondary flow system to offset inleakage from the tertiary coolant through the condenser. Additionally sludge was an undesired product. Unsuccessful remedial measures led to the use of the All Volatile Treatment (AVT) in which volatile chemicals having volatile reaction products are employed. The chemicals in the treatment are ammonia and hydrazine. Experience is showing a reduction in tube "thinning" and in sludge formation with the AVT. Wootten, ff. Tr. 4126, at 6-10.

A-29. Tube-wall thinning results from accumulations of sludge arising from corrosion by phosphates. Remedy is by AVT and adjustment of flow speeds to disperse the solids. Malinowski, ff. Tr. 4126, at 15, 16.

A-30. Pitting is the formation of about 100-mil-diameter discrete circular regions believed to be corrosion attributable to actions by acidic copper and chloride ions. It has been observed even in Inconel and may be mitigated by improved water chemistry (AVT) and removal of copper alloys from the secondary coolant system, from the condenser for example. Wootten, ff. Tr. 4126, at 13, 14.

A-31. Denting is a type of deformation resulting, for example, in a decrease in a tube radius at support and baffle plates caused, in turn, by an accumulation of solid corrosion products, including magnetite[7] in the annulus between the tube and the support plate establishing an undesirable stress. This condition is expected to be lessened by improved water chemistry; by replacing the carbon steel support plates with stainless steel;[8] by substituting quartrefoil holes for the tubes in the support plates for the earlier circular ones, thereby diminishing crevices and improving the flow pattern and scavenging; and by expanding the tubes at the support plates, thereby further removing crevices. Conway, ff. Tr.

[7] Applicant's witness Wootten attributed denting to an accumulation of solids resulting from an interrelationship among chlorides, copper compounds and oxygen. Wootten, ff. Tr. 4126, at 11. He also identified the deposit as magnetite. Tr. 4177. The Board assumes the presence of both solid forms to be reasonable. Magnetite is readily detected by the eddy-current technique. Tr. 4395 (Malinowski).

[8] Stainless steel support plates and baffles are in Byron Unit 2, but not in Unit 1.

4126, at 10, 11; Tr. 4364 (Conway). Denting is considered by Intervenors not to be an accident-inducing problem. Tr. 4507 (Bridenbaugh).

A-32. Wear results from abrasion of generator tubes upon contact with other items of the structure, exemplified by fixed baffle plates and adjacent tubes set into vibration, and with extraneous objects inadvertently left in the generator shell.[9] Remedies to be included at Byron are the installation of noise detectors and provision for visual inspections for loose parts. Tr. 4424 (Blomgren). An additional remedy to wear is the modification of the secondary coolant flow pattern in the shell augmented by installing more securely those tubes susceptible to vibrations. The modification to the secondary flow into the shell is essentially a 10 percent reduction in the input flow through the main feedwater nozzle and the diversion of that flow into the generator through an auxiliary nozzle, location not specified. Tr. 4266 (Conway); Tr. 4273 (Blomgren); Timmons, ff. Tr. 5908, at 23. The evolution of the above prescription appears below.[10]

[9] Two instances have been recorded of damage to steam generator tubes caused by foreign objects left in the shell. In 1979 a leak at Prairie Island Unit 1 was found to have been due to tube wear by a coiled spring remaining in the shell following a maintenance shutdown. In 1982 a sizeable leak occurred at the Ginna plant through a long-wear scar in an active tube produced by the impingement of a section of a previously plugged tube which had been severed by an extraneous ½-pound metal plate left behind following a maintenance operation. Frank, ff. Tr. 4473, at 8; Fletcher, ff. Tr. 5908, at 14.

[10] An extraction from the rather lengthy and laborious testimony, both written and oral, gives somewhat the following genesis of the tube vibration problem and its purported remedy. These remarks concern the "counter flow" generators (Models D4 and D5), not the earlier "split-flow" types (Models D2 and D3). The ill to be cured is generator tube wear. Whether this wear results from mutual abrasion of adjacent vibrating tubes or from the impingement on the tube of a high-speed water stream, or both, has not been made clear and is likely unimportant. The generators considered are of the "preheater" type where at least a part of the incoming secondary stream is first directed across the portion of the tubes where the primary coolant exits the generator. In this manner the thermal efficiency of the generator is enhanced. Conway, ff. Tr. 4126, at 8.

About two years ago a Model D3 Westinghouse steam generator in Sweden developed a 2.5-gallon/minute primary-to-secondary leak. Examination showed a single hole in a tube, at a baffle plate, facing the feedwater inlet. Wear was observed in other tubes in outer rows near the nozzle. Like examinations within Westinghouse of other generators thereby were initiated. Subsequently an inspection at the KRSKO reactor in Yugoslavia with a D4 generator, which had operated at 50 percent power, showed undetectable tube wear. Nonetheless accelerometers were installed which subsequently showed insignificant tube vibration at 70 percent power and less. Entrance of the remaining 30 percent of the design full-load-flow into the generator via a bypass* did not increase vibrations beyond acceptable limits. [*The location of this bypass entrance is not made clear in the evidence of the Byron hearing.]

Further to the KRSKO experience, about a year ago one tube in the generator was expanded at the baffle plate location.** No subsequent wear was observed with eddy-current measurements, and vibration was a factor of five below that before expanding. Timmons, ff. Tr. 5908, at 10-16. [**In his testimony, Applicant's witness Timmons refers continually to "baffle plate intersection" as the place tubes are expanded, while the location of baffle plates is not carefully identified in the testimony. They are, however, in the preheater section of the secondary side of the generator and are functional in the distribution of the inflowing secondary coolant among the tubes. Malinowski, ff. Tr. 4126, at 21. At least one baffle is not tube-supporting. Tr. 4767 (McCracken). *See* Timmons, ff. Tr. 5908, at 15; also Timmons' Attachments 2, 3 and 4 thereto. On the other hand, Conway speaks of tube expansion at their intersection with tube support plates. Conway, ff. Tr. 4126, at 15; also Tr. 4409 (Blomgren). An obscure description is given at Tr. 6180 (Timmons). The Board recognizes that tube expansion at both places is desirable.]

A-33. An operations procedure to be established by the Applicant is to monitor leakage of coolant from the primary to secondary system. This monitoring can most likely be done by the appearance of radioactive species in the secondary. The ductility of Inconel 600, the tube material, allows the development of short cracks, and hence minor leaks, well before rupture can occur. The maximum generator-tube leak rate permitted at Byron by the Technical Specifications is 500 gallons/day (0.35 gallon/minute) per generator and will occur through a single 0.43-inch-long crack. The critical crack length corresponding to the maximum accident condition pressure during a postulated feedwater-line break/steam-line break was conservatively determined to be 0.51 inch using the results of burst pressure tests. Since the maximum permissible crack length of 0.43 inch for continued operation is less than the critical crack length of 0.51 inch, the unit is safeguarded against tube rupture during a postulated feedwater-line break/steam-line break accident. Patel, ff. Tr. 4126, at 12-13. In the application of this limiting leak rate, a single penetration is assumed, thereby introducing a significant element of conservatism. Leak detection sensitivity as low as 0.001 gpm is not impractical (Tr. 4339 (Malinowski)) and 0.05 gpm is common (Tr. 4338 (Patel)). This characteristic, called "leak-before-break," allows opportunity for remedial measures before a severe event. The usual remedy for leaky tubes is removal from service by plugging. As installed, the thickness of the Inconel generator tubes is 43 mils, corresponding to a burst pressure of 10,000 psi to be compared to normal operating pressure differential of 1250 psi. The ASME Boiler and Pressure Vessel Code recommends a safety factor of only three. The factor of three describes a tube of about half the wall thickness of that installed. This oversize allows considerable tube degradation, such as thinning, before remedies need be effected. In fact the "tube plugging criterion," the wear before repair or removal from service, is 40 percent of the original thickness; that is, a 26-mil-thick wall is acceptable for operation. Tr. 4369 (Patel).

A-34. Staff witnesses report that a tube uniformly reduced to 20 percent of its original wall thickness,[11] *i.e.*, to about 8 mils, will withstand the pressure arising from a steam-line break, a pressure differential of about 2650 psi. Tr. 4600-03 (Frank, Rajan).

A-35. A Model D4 Westinghouse steam generator is installed in Byron Unit 1 and a D5 is in Unit 2. Tr. 4388-89 (Blomgren). Into each

[11] Both the transcript at Tr. 4603 and Staff's Finding D-100 say "[a] tube . . . deviated to about 80 percent of the nominal wall thickness . . . ." In the context of each citation, however, the Board understands the statement as meaning an 80 percent loss of the wall.

of these models have been incorporated improvements over earlier installations.[12] The improved secondary water chemistry, the absence of copper in the secondary stream[13] (Tr. 4180 (Malinowski)), the use of fresh water as the tertiary coolant (Tr. 4389, 4390 (Wootten)), the reduction in tube vibration by diverting some 10 percent of the secondary flow to the generator through the bypass, a more rigid mounting of approximately 100 peripheral tubes,[14] and the installation of heat-treated Inconel 600 as tube material (Tr. 4348 (Conway)) are common to both Units 1 and 2.

A-36. The direct measurement of tube vibration as a function of steam generator feedwater flow rate, at KRSKO and at various test facilities, showed, for that reactor installation, diversion of 30 percent of the required total full-power input to be optimum. Because of differences in total flow, and the fact that the speed of the water into the preheater, not quantity, is controlling, that division will correspond to a 75/25 division at Byron. Tr. 6262-63 (Timmons).

A-37. The maximum diversion of full-power feedwater obtainable through the fully open bypass valve at Byron is 10 percent. In the considered judgment of Westinghouse, with the concurrence of the Applicant, effecting the program of more rigidly supporting approximately 100 peripheral tubes will allow satisfactory operation of the generators with this 90/10 division of feedwater flow. Any lesser ratio will require significant changes in the piping, such as installation of orifices and even sizing. Tr. 6224-28 (Timmons).

A-38. Additionally, the Model D5 in Unit 2 will have more corrosion-resistant stainless steel tube-support plates replacing carbon steel (Tr. 4351 (Conway)), tubes expanded by a pressurized hydraulic-fluid device (rather than the usual mechanical-roller method) to improve the primary flow pattern and reduce stresses at the tube sheet, and quatrefoil support-plate holes instead of circular to reduce solids deposition and denting. *Id.* Each of these changes is expected to further improve generator performance. The Intervenors express concern that these changes are not being made in Unit 1. Tr. 6462 (Bridenbaugh).

A-39. Westinghouse is sufficiently confident of the expected improved performance of the Model D4 generator as a consequence of the modifications noted in Paragraph A-35, *supra,* to proceed into the operation of Unit 1 even though the benefit of further changes already made

[12] No operating experience has been accumulated with Model D5; the KRSKO plant has a D4.

[13] The Byron condensers are tubed with stainless steel. Tr. 4240, 4428 (Blomgren).

[14] Although the exact number and location of the tubes to be expanded have not yet been established (Tr. 6240, 6306 (Timmons)), they will likely be those determined by tests to be the ones most susceptible to wear (Tr. 4767 (Rajan)).

in D5 (Paragraph A-38, *supra*) is recognized. Tr. 4389 (Wootten); Tr. 4435 (Conway).

A-40. Of importance in this array of modifications is water treatment. The Steam Generator Owners Group (SGOG) has established guidelines for water chemistry stricter than even those of the NSSS. The Applicant has incorporated the SGOG guidelines in its chemistry monitoring program at Byron. Blomgren, ff. Tr. 4126, at 8-11.

A-41. SGOG is an association of PWR operators, established in 1977, which, in concert with the Electric Power Research Institute, has worked toward improvement in steam generator performance. The Applicant holds membership in the Group and actively supports it. Bridenbaugh, ff. Tr. 6406, at 7, 8, 9; Blomgren, ff. Tr. 4126, at 8.

A-42. The twelve viable investigative efforts presented in Intervenors Ex. 9 likely had their genesis, at least in part, in a presentation to the NRC by the SGOG in July 1982.

A-43. Intervenors' witness selected eight of the twelve items he considered of sufficient importance to be "imposed" by the Staff on Byron. These are, by abbreviated title: (1) loose parts control, (2) degraded tubes, (3) in-service inspection, (4) water chemistry, (5) condenser inspection, (6) additional inspection ports. Equally important are (7) pressure control during tube rupture — already imposed — and (8) eddy-current techniques considered an industry-wide responsibility to be applied in their most advanced form at Byron. Tr. 6442-43 (Bridenbaugh).

A-44. Westinghouse has made extensive studies of the effect of the vibration of generator tubes on wear leading, in the limit, to their failure and necessary removal from service. Westinghouse has developed proprietary empirical relations among directly measurable quantities describing such vibrations which can be correlated with the wear suffered during vibration.[15] In summary, rather dramatic results were presented from this study both of the extent of the reduction in vibration upon expansion of tubes into baffle and support plates and in the concomitant decreased wear. Intervenors' witness expressed an expectation of success of tube expansion in the reduction of tube wear. Tr. 6507 (Bridenbaugh).

A-45. There shall be a 100 percent pre-service inspection of the steam generator tubes.[16] In this inspection the eddy-current measuring

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[15] The details of the measurements and their correlations were presented to the Board and parties in an *In camera* session on April 27, 1983. Tr. 6162-6203. This transcript is not in the public record.

[16] The inspection of the tubes in the four generators of Unit 1 has in fact been completed by the four-frequency eddy-current measurements following procedures conforming to applicable ASME and ASNI requirements. Of the some 18,000 tubes examined, two were apparently mechanically blocked to the *(Continued)*

device will be inserted into a tube at the hot-leg tube sheet and then pushed past the U-bend entirely, down the cold leg to the exit of the tube.[17] The initial in-service inspection shall occur at the first refueling outage or within 24 full-power months equivalent. The interval to subsequent inspections will be determined by experience. Malinowski, ff. Tr. 4126, at 6; Tr. 4282 (Blomgren).

A-46. Intervenors contend that the procedures for in-service inspection of tubes should include plugged tubes, and that an adequate method for such inspection should be developed. Visual inspection internal to the generator shell, by television and fiber optics, does not completely suffice. Additionally, in-service inspections of generators are not made sufficiently frequently. A biennial schedule for each generator was suggested as a minimum rate. Tr. 6445-46 (Bridenbaugh). The thrust of the witness was not directed so much at the specific details of requirements, such as those delineated here, but more to the historic interval when their problems were known, yet solutions to those problems were not derived and no firm conclusionary enforcement actions were taken to give confidence that generators would function in an acceptable manner. Tr. 6444 (Bridenbaugh).

A-47. Additionally, the in-service inspection procedures, reflecting as a minimum the requirements of applicable codes and regulations, should be prepared in advance of plant operation. Tr. 6500 (Bridenbaugh).

A-48. Other inspection programs, particularly with respect to water chemistry, are presented in the record. Applicant Ex. 17, "PWR Secondary Water Chemistry Guidelines," DPRI-2704-SR; Tr. 4252, Tr. 4200-68 (Blomgren, Malinowski, Wootten).

A-49. There exists an ongoing research and development effort directed to every facet of steam generators — geometry, materials, techniques, inspections — largely under the aegis of the Steam Generator Owners Group (SGOG) associated with the Electric Power Research Institute. Tr. 4406-08 (Conway, Malinowski, Blomgren).

A-50. The Applicant has committed to the several proposals made by Westinghouse for the modifications of the Byron steam generators to

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degree that the probe would not pass through; about 500 tube dents were observed in each generator, having occurred, no doubt, during manufacture, shipment and installation; in about 200 tubes, areas of off-standard magnetic permeability were detected indicating local impurities in the Inconel; although some tubes showed variations in wall thickness, there was none greater than the 20 percent loss requiring reporting. The two blocked tubes were to be plugged. Additionally to the general description of the tubes and their quality, this inspection provided a data base for subsequent in-service observation. Paillaman and Malinowski, ff. Tr. 4816, at 4-10; Tr. 4821 (Paillaman).

[17] Regulatory Guide 1.83 recommends the survey to traverse the tube around the bend to a point in the cold leg adjacent to the upper support plate. The Applicant's test is more informative.

improve their service; further, these modifications are to be incorporated prior to the operation of the Station. Blomgren, ff. Tr. 4126, at 17, as modified at Tr. 4118, 4119; Butterfield, ff. Tr. 5906, at 5; Tr. 6056 (Butterfield). This commitment (preoperational implementation) by the Applicant was reaffirmed by its counsel. Tr. 6385 (Gallo). Although the Staff is generally familiar with the proposed modifications and, particularly, believes they will take care of the tube vibration problem, it will review the proposals before operation of the Byron Station is begun. Rajan, ff. Tr. 4473, at 5; Tr. 4674-75 (Rajan). The Staff estimates the tube wear not to reach 40 percent of initial wall thickness during the 50-year life of the plant. Tr. 6328 (Rajan).

A-51. A quantitative model[18] for the assessment of the expected performance of steam generator tubes, expressed as expected frequency of occurrence, was presented by the Applicant. There are on record only five instances of severe tube leaks in operating Westinghouse steam generators.[19] Each event is treated in this assessment as though the defect was a complete tube rupture. Overall operating experience encompasses 2.5 million tube-years, or $2 \times 10^{-6}$ events/tube-year. On this basis, among the 18,000 tubes at a Byron unit, a tube rupture will occur at a Byron unit about every 30 years. The witness placed a range factor of five on this value at a 90 percent confidence level. The model does not recognize the recently presented modifications intended to improve tube performance. The results of the assessment predict frequencies of single and multiple tube ruptures of the order of $10^{-2}$ and $10^{-5}$ per operating-year respectively. These results combined with or taken as a consequence of loss-of-coolant or major fluid-line break events give frequencies of the order of $10^{-5}$ to $10^{-7}$ per year. Moreover, severe core damage as a consequence of tube rupture is expected at a frequency of

[18] The model was developed for the Byron Probabilistic Risk Assessment, a document not in this record except in pertinent part as an attachment to this witness' testimony. Tr. 2322 (Gallo).

[19] These events with 0.875-inch-diameter tubes are:

| No.: | Occurrence Date: | Plant (startup date): | Attributed Cause: | ELR[a] (gpm) |
|---|---|---|---|---|
| 1 | 2/26/75 | Point Beach 1 (10/70) | Phosphate Wastage + SCC[b] | 125 |
| 2 | 9/15/76[c] | Surry 2 (1/73) | Denting + SCC | 80 |
| 3 | 6/25/79 | DOEL 2 (Belgium, 6/75) | Ovality + SCC | 135 |
| 4 | 10/2/79 | Prairie Island (8/73) | Loose part (spring) | 390 |
| 5 | 1/25/82 | Ginna (9/69) | Loose part (plate) | 634 |

[a] ELR = Estimated Leak Rate (gpm)

[b] SCC = Stress Corrosion Cracking

[c] This is the most recent domestic tube rupture due to corrosion. Tr. 6454 (Bridenbaugh).

one in 10 million operating years. Hitchler, ff. Tr. 5908, at 5-8 plus Attachment A; Tr. 6231, 6235 (Hitchler). Intervenors' witness did not dispute the once-in-30-years probability estimated. Tr. 6509 (Bridenbaugh).

A-52. Intervenors' witness opines that there will be an increased likelihood of the occurrence of accidents attributable to steam generator tube failure at a given installation. This increase is predicated on the recent development of new and increasing incidences of tube degradation. Bridenbaugh, ff. Tr. 6406, at 5; Tr. 6510 (Bridenbaugh). The witness, however, did not support his opinion with any quantitative measure of the increased accident frequency. Tr. 6475 (Bridenbaugh).

A-53. No Staff witness disagreed with the proposals for tube modification put forth by the Applicant's panel. Tr. 4792 (Rajan, Frank, Marsh, McCracken).

A-54. The Staff looks with optimism at recent developments towards solutions of steam generator problems, including those testified to in this proceeding, which have plagued the industry. Staff concludes that the tube degradation Unresolved Safety Issue A-3 is now not sufficiently severe to warrant delay in licensing new PWRs for operation. Frank, ff. Tr. 4473, at 7.

A-55. Recent operating experiences are showing regulatory requirements to be generally satisfactory to control corrosion sufficiently, when supplemented by frequent and intense inspections to reveal symptoms, to retain structural integrity adequate to prevent tube rupture which could violate the health and safety of the public. Tr. 4714 (McCracken). Corrosion will be reduced but cannot be prevented. Tr. 4713 (McCracken, Frank).

A-56. Continuing improvement in the control of corrosion of generator tubes, which is the solution to the steam generator Unresolved Safety Issue A-3, together with the necessary reviews within the NRC will remove A-3 from the list. Tr. 4714, 4798 (McCracken); Tr. 4478 (Marsh).

A-57. Mr. Bridenbaugh emphasized the importance to the future of the industry of remedies to steam generator problems, such remedies being represented, at least in part, by the modifications proposed by Westinghouse, accepted by the Staff and committed to by the Applicant. He made particular reference to expected improvements in the secondary water chemistry, to which such high hopes and extreme importance as a major contributor to the hoped-for betterment have been assigned. He recommended that licensing of Byron be delayed until the "new" water chemistry be thoroughly reviewed by some to-be-established group of knowledgeable and experienced individuals who are separate from the utility and the regulating agency. The review should emphasize the chemical procedures themselves and, equally, the operating procedures by which they will be effected. Tr. 6462-69 (Bridenbaugh).

A-58. The above reference to a review of the secondary water treatment at Byron is one of the witness' eight recommended actions before the plant is made "radioactive." Bridenbaugh, ff. Tr. 6406, at 21.

A-59. In the witness' opinion it is prudent to perform in the best possible manner. In this context he faults the Staff and the Applicant for doing less than their alleged best in not installing the current ultimate modifications in the D4 generator in Byron Unit 1. Tr. 6468 (Bridenbaugh).

A-60. From his own knowledge and experience and from the testimony of the other parties presented in this proceeding, Mr. Bridenbaugh concluded that some progress has been made toward resolution of the long-standing difficulties with steam generators. There has been an investment of considerable effort by concerned parties in understanding the technical issues and in the proposed remedies. Those remedies, however, have not yet been cleared even within the NRC. They have not been enforced in the field. Tr. 6477-79 (Bridenbaugh). The Board accepts this categorization of the present status of the problem.

A-61. The Board recognizes that the modifications proposed for construction, operation and maintenance of the steam generators is no panacea for all of their ills. Rather, the modifications represent a step toward improving generator performance, particularly a resistance to corrosion and, consequently, toward lessening the potential for endangering the public's health and safety. Staff makes a rough estimate of a decrease in corrosion by a factor of three. Tr. 4772 (McCracken).

A-62. The Board further recognizes that the proof of those remedies to steam generators rests in the success of their application which, on a practical scale, is supposedly yet to come. The Board recommends that the industry and the Staff continue to be vigilant in the application, inspection, review and evaluation of the proposals, and others hopefully and likely to surface as experience is accumulated, in a concerted effort towards a true ultimate solution of what has, in the past, appeared as a Sisyphean task. There is no justification for complacency.[20]

A-63. The Board recognizes that many historic and troublesome misoperations leading principally to tube leakage and rupture in the past have been addressed under the comprehensive heading Unresolved

[20] These remarks are independently shared in spirit by the Advisory Committee on Reactor Safeguards. *See* ACRS letter to the Chairman of the Commission, dated October 18, 1983, subject Unresolved Safety Issue A-3.

Safety Issue A-3, and that the outlook towards future generator performance is brighter because of the responses, and the evidence supporting them, made here to Intervenors' questions. A multitude of specifics of construction, operation and degradation were addressed in this proceeding, including water chemistry, searching for mobile objects potentially detrimental to tubes, means of reducing tube wear, the probability of future damaging events, and the selection of better basic materials together with surveillances, both qualitative and quantitative, which may predict events. Whereas this list may be of those weaknesses now known and experienced, the Board is confident that the future holds the identity of still more difficulties potentially of equal severity and importance.

A-64. In the foregoing findings, the Board has noted many improvements in design and procedures already implemented which will enhance the integrity of the steam generators at Byron. In addition, we found that the Applicant has represented to the Board that several prospective actions, equipment modifications, design changes and procedures will be implemented at Byron. The Board relies upon Applicant's representations in deciding the steam generator integrity issue in Applicant's favor. Specifically the following commitments, identified according to Applicant's respective proposed findings, are essential to the Board's conclusions.

1. Before operation, in each steam generator at the Byron Station, about 100 tubes will be expanded where they intersect the baffle plate in the preheater. Applicant's Proposed Findings 166 and 178.
2. Feedwater flow will be split. To ensure that about 10 percent of the feedwater flow will enter the steam generator through the auxiliary feedwater nozzle, changes will be made to the control circuitry of the feedwater preheater bypass valve, and a flowmeter will be installed on the feedwater bypass line. These modifications, too, will be completed before operation. Applicant's Proposed Findings 172 and 178.
3. An AVT water chemistry program, based on strict adherence to Westinghouse and EPRI guidelines, will be implemented on the secondary side of the reactor systems at Byron. Applicant's Proposed Findings 180, 182, 185, and 189. To mitigate the denting experienced by plants which have operated only on AVT, the Byron program will include reduction of the ingress of oxidizing agents such as copper and oxygen, and further restriction on the introduction of chloride ions into the secondary system. Applicant's Proposed Finding 186.

4. To detect any degradation of the walls of the steam generator tubes, a 100 percent pre-service inspection following NRC Regulatory Guide 1.83 will be performed on the tubes in Unit 2. Applicant's Proposed Finding 192. This inspection will establish a baseline against which later in-service inspections will be compared. Applicant's Proposed Finding 196.
5. These in-service inspections will be performed according to the Byron Technical Specifications and NRC Regulatory Guide 1.83. Applicant's Proposed Finding 195. Eddy-current testing will be the primary inspection technique. Applicant's Proposed Finding 200. The eddy-current inspection program set out in Attachment A to the prefiled testimony of John C. Blomgren, an employee of the Applicant, is the minimum inspection the Applicant will conduct. Blomgren, ff. Tr. 4126, at 11. Moreover, the Applicant will update its eddy-current testing techniques and equipment as technology advances. *Id.* at 12.
6. To guard against damage caused by loose parts and foreign objects in the secondary side of the steam generators, the Applicant will have a loose parts control program which will consist of tool and material inventory control procedures and a Loose Parts Monitoring System (LPMS). Applicant's Proposed Findings 205-07. (1) The inventory control procedures will require that all materials and tools used in the secondary side of the steam generators during maintenance and inspection be accounted for before the steam generators are returned to operation. In addition, the maintenance procedures will require hold points for cleanliness operations. Applicant's Proposed Finding 206. (2) The LPMS for Byron will be a monitoring, alarm, and diagnostics system which provides real-time information to the operator on a variety of mechanical vibration phenomena. Applicant's Proposed Finding 207. Moreover, the secondary side of each steam generator will be visually inspected from time to time. *Id.*
7. The Byron reactor systems will also be monitored for primary-to-secondary leakage. Applicant's Proposed Finding 208.
8. Last, to provide an overall measure of tube integrity, periodic hydrostatic pressure testing will be performed on the steam generators. Applicant's Proposed Finding 213.

A-65. Accordingly the Board finds that, contrary to League Contention 22 and DAARE/SAFE Contention 9(c), there is reasonable assurance that the steam generators at Byron will maintain their integrity.

### 3. *Water Hammer in Preheat Steam Generators*

A-66. The water hammer contention, set out above, refers to Yugoslavia's KRSKO plant, whose steam generators are similar in design to those at Byron. KRSKO experienced a bubble-collapse water hammer event, and Intervenors would require Applicant to demonstrate that a similar event will not occur at Byron. This contention was pursued independently by DAARE/SAFE. The steam generators at both Byron and KRSKO are Model D steam generators by Westinghouse. Carlson, ff. Tr. 930, at 6. The primary and secondary sides of the steam generator are the fluid volumes inside and outside the steam generator tubes. Primary water from the reactor enters the steam generator at the bottom and goes up into the inlet half of the bundles of inverted U-shaped tubes. The first half of the bundle where the primary water flows upward is referred to as the hot-leg side. The second half, where the primary water flows downward, is referred to as the cold-leg side because the primary water has given up some of its thermal energy to the feedwater surrounding the tube bundle. The cooled water exits the steam generator at the bottom and goes back to the reactor. Secondary coolant water from the condenser is fed into the steam generator, and during normal operation the lower part of the tube bundle is surrounded by water and the upper part by steam. *Id.* at 5-6.

A-67. To make the transfer of heat from the reactor water to the feedwater more efficient, a Model D steam generator has a region of baffles, called a preheater, located near the bottom of the cold-leg side of the tube bundle. Feedwater passes through the preheater on entering the steam generator. *Id.* at 7.

A-68. The preheater may be subject to a potentially damaging event called "water hammer." A water hammer is a sudden increase in water pressure caused by a decrease in water velocity. For instance, when flowing water in a pipe is abruptly stopped by the rapid closing of a valve downstream of the source of the flow, the inertia of the flow produces an increase in pressure in the pipe. The same effect can result when slugs of water meet from opposite directions. In steam bubble-collapse water hammer, a pocket, or "bubble," of steam trapped by cold water condenses rapidly, "collapses," causing slugs of water to rush from opposite directions into the void left by the condensing steam. Carlson, ff. Tr. 930, at 3-4. Water hammer can occur in the preheater of the Model D steam generator because steam can become trapped in the preheater and the feedwater can be cold enough to cause the steam to condense rapidly. *Id.* at 7.

A-69. To prevent bubble-collapse water hammer in the preheater, the Model D generator has a feedwater bypass system, which automatically prevents water cold enough to cause water hammer from entering the preheater. If it should be necessary to feed water which is too cold into the steam generator, the bypass system pipe carries the feedwater from the main feedwater line up to an auxiliary nozzle near the top of the steam generator, above the preheater. *Id.*

A-70. But apparently, the bypass piping can suffer from what it is designed to prevent. From pipe damage discovered at KRSKO during an inspection in August 1981, it is inferred that a bubble-collapse water hammer was caused in July 1981 by the introduction of cold water into the feedwater bypass pipe while the pipe was full of steam which had leaked back from the steam generator. *Id.* at 8. The Intervenors fear that something similar could happen at Byron.

*The Scope of the Contention*

A-71. DAARE/SAFE has sometimes argued, and at other times has appeared to argue, that its Contention 9(a) is about more than bubble-collapse water hammer in a feedwater bypass line into a Model D steam generator preheater. Some of DAARE/SAFE's Proposed Findings suggest that DAARE/SAFE thinks that the contention covers water hammer in the preheater steam generator. Nothing in DAARE/SAFE's Proposed Findings explicitly says that the contention extends beyond water hammer in the bypass piping, but DAARE/SAFE's Proposed Findings 49-56 do talk a great deal about preheater water hammer, and DAARE/SAFE's representative, Ms. Chavez, claimed during the hearings that the contention extended to preheater water hammer (Tr. 946-47 (Chavez)). The Applicant is persuaded that DAARE/SAFE's Proposed Findings 49-56 try to stretch the scope of the contention. *See* Applicant's Reply Findings at 7.

A-72. Whatever DAARE/SAFE may have intended by its Proposed Findings 49-56, Contention 9(a) does not cover preheater water hammer. Not only does the language of the contention say nothing about preheater water hammer, the history of the contention shows that the contention's silence about any kind of water hammer but the kind which is thought to have occurred at KRSKO rules out litigation in these proceedings of any other kind of water hammer. By our Memorandum and Order of September 10, 1982 (unpublished), we in effect decided by summary disposition all issues of water hammer at Byron except KRSKO-type water hammers. "Left unresolved in Contention 9(a) by the summary disposition motions was whether a water hammer event in the feedwater bypass similar to the type of occurrence that was believed to have happened at the Yugoslavian plant could take place at Byron."

Memorandum and Order Ruling on Applicant's Motion for Clarification, January 7, 1983 (unpublished), at 2.

A-73. DAARE/SAFE, by talking about preheater water hammer in its Proposed Findings 49-56, apparently wants not only to stretch the contention to cover preheater water hammer, but also to raise the question of metal fatigue. DAARE/SAFE's Proposed Findings 55-56. Applicant's witness Carlson says water hammer can cause metal fatigue in the preheater. Tr. 1076. To justify raising the question, DAARE/SAFE cites Applicant's witness Carlson's testimony that bubble-collapse water hammers have common elements wherever they occur. Tr. 1075-76.

A-74. But the common elements Carlson refers to are the *causes* of the water hammers — a "confined volume of steam being rapidly condensed by cold water . . . brought into contact with it" (Tr. 1075) — not the *effects,* as DAARE/SAFE would have it. Carlson went on to testify that bubble-collapse water hammer could cause metal fatigue in the preheater because it has a "rather complicated . . . structure, the combination of tubes and baffle plates and partition plate," but that such water hammer would not cause metal fatigue in the simpler geometry of the bypass piping. Tr. 1085-86, 1111. Preheater water hammer is neither an issue in its own right in Contention 9(a), nor a sign that we should be concerned about metal fatigue in the bypass piping.

A-75. DAARE/SAFE also suggests, in its Proposed Findings 51, 52, 54, and 55, that Contention 9(a) covers what is called "acoustical" or "classical" water hammer in the bypass piping. It is true that the Applicant and Westinghouse do think that classical water hammer could occur in the bypass line (Carlson, ff. Tr. 930, at 7), and it is true that bubble-collapse water hammer and classical water hammer can have the same effects on piping. Tr. 986-87 (Serkiz). But we reject the suggestion that the contention extends to classical water hammer. Again, the words of the contention say nothing about such water hammer, and as our September 10, 1982 Memorandum and Order and the January 7, 1983 Clarification show, Contention 9(a) is to be taken at face value. Even if Contention 9(a) did cover classical water hammer, DAARE/SAFE has presented no evidence that the Applicant has not taken adequate measures against such water hammer or its effects.

A-76. But DAARE/SAFE's most important argument about the scope of Contention 9(a) is this: "The issue before us then is whether the auxiliary feedwater system at the Byron plant is appropriately protected against the dynamic effects of a KRSKO-type water hammer event so as to meet Standard Review Plan criteria." Proposed Findings at 3. The auxiliary feedwater system pumps water to the steam generator when the main feedwater line breaks, or some other accident happens which causes the heat sink to be lost. The feedwater from the auxiliary system flows into the feedwater bypass line and then into the generator through the auxiliary nozzle. Carlson, ff. Tr. 930, at 8. The grounds of this last of DAARE/SAFE's claims about the scope of Contention 9(a), and our rejection of the claim, are best discussed in setting out the law which applies to the resolution of the contention.

A-77. Although the last sentence of DAARE/SAFE's Contention 9(a) is categorical — "Applicant should be required to demonstrate that a similar event will not occur at Byron" — DAARE/SAFE concedes that 10 C.F.R. Part 50, Appendix A, General Design Criterion 4, which applies to the resolution of this contention, "does not require a guarantee that a KRSKO-type water hammer event will not occur at Byron." DAARE/SAFE's Proposed Findings at 3.

A-78. Instead, 10 C.F.R. 50.57(a)(3) requires that before an operating license may be issued, there must be "*reasonable assurance* . . . that the activities authorized by the operating license can be conducted without endangering the health and safety of the public . . ." (emphasis added). One way to satisfy this regulation is to build the plant according to the General Design Criteria set out in Appendix A to 10 C.F.R. Part 50. The parties agree that Applicant's measures to prevent bubble-collapse water hammer must meet Criterion 4. In pertinent part, General Design Criterion 4 says, "[s]tructures, systems, and components important to safety shall be . . . appropriately protected against dynamic effects . . . that may result from equipment failures and from events and conditions outside the nuclear power unit." A feedwater bypass line is, in the words of Criterion 4, a structure "important to safety" in a number of ways, and it must be "appropriately protected" against a bubble-collapse water hammer, which is a "dynamic effect," thought to have resulted at KRSKO in part from an "equipment failure" to be described later.

A-79. DAARE/SAFE argues that besides 10 C.F.R. 50.57(a)(3) and General Design Criterion 4, Section 15.2.8 of the Standard Review Plan (SRP) (NUREG-0800), should be applied to the resolution of the contention. DAARE/SAFE says that Section 15.2.8 "describes the review pertinent to the evaluation of potential water hammer effects." DAARE/SAFE's Proposed Findings at 2. Section 15.2.8 sets out criteria for the performance of the auxiliary feedwater system under certain accident conditions and says that the auxiliary feedwater system should be put through preoperational tests to verify that it can function after a feedwater-line break. NUREG-0800 at 15.2.8-4. DAARE/SAFE makes two major claims about Section 15.2.8 of the SRP. The first is that "Applicant has not demonstrated that the preoperational procedures or operation plans being developed [to deal with KRSKO-type water hammer events] can specifically address feedline breaks or accident sequences so as to maintain auxiliary Feedwater System integrity to meet Standard Review Plan criteria." Proposed Findings at 9. DAARE/SAFE's second claim is that the Applicant has not shown that its "preoperational procedures and operation plans" for KRSKO-type water hammer events will address the changes which the Staff has testified (Tr. 1012 (Serkiz)) that it is recommending be made in the Standard Review Plan to reflect experience with water hammer. From these two claims DAARE/SAFE concludes that the auxiliary feedwater system at Byron is not, in the language of General Design Criterion 4, "appropriately protected against the dynamic effects" of a bubble-collapse water hammer in a feedwater bypass line. *Id.* at 9-10.

A-80. As to DAARE/SAFE's first claim about Section 15.2.8 of the SRP, that Section has no place in the litigation over Contention 9(a). The Section merely guides the Staff in determining whether the Applicant is complying with criteria set out elsewhere — among those criteria are General Design Criteria 27, 28, 31, and 35 (*see* SRP at 15.2.8-3), but not 4, and the parties agree that 4 is the only Criterion which is applicable here.

A-81. Perhaps more important, Section 15.2.8 of the SRP says nothing about measures designed to *prevent* water hammer in a bypass line. It speaks only about measures designed to keep the auxiliary feedwater system functioning after a feedwater-line break. In fact, Section 15.2.8 says almost nothing about water hammer, only that potential water hammer effects on safety valve integrity in the event of a feedwater-line break should be evaluated. *Id.* at 15.2.8-2. But Contention 9(a) calls for prevention of water hammer, not measures for dealing with its consequences. In DAARE/SAFE's own words, "Applicant should be required to demonstrate that a similar event will not occur at Byron." The adequacy of the Applicant's plans for preventing a KRSKO-type water hammer event is determined by the degree to which they assure that a KRSKO-type water hammer will not occur, not by the degree to which they assure that the auxiliary feedwater system will function after a feedwater-line break caused by water hammer. Of course, the Applicant must show that the auxiliary system will function after a feedwater-line break, but the showing is not necessary as a response to this contention.

A-82. DAARE/SAFE's second claim about Section 15.2.8 of the SRP, that the Applicant cannot make a proper showing against Contention 9(a) until the changes the Staff contemplates making in the Standard Review Plan are announced, makes sense neither legally nor practically. On DAARE/SAFE's theory that we cannot judge the Applicant's measures against KRSKO-type water hammer event until we have a revised SRP in hand, issues we dealt with by summary disposition in our Memorandum and Order of September 10, 1982, might be before us again, for the changes the Staff recommends deal with more than just KRSKO-type water hammer events. Thus once again, as in DAARE/SAFE's pleadings opposing the Applicant's September 28, 1982 Motion for Clarification of our September order, DAARE/SAFE is in effect asking us to reconsider our summary disposition, and once again, we decline.

A-83. But more important, whatever those changes may be, they will not affect the Applicant's handling of bubble-collapse water hammer in feedwater bypass lines. Mr. Serkiz, the Staff's witness on Contention 9(a), testified that, taken generically, water hammer is no longer a significant safety issue (Tr. 1033), that although water hammer was still listed by the NRC as an Unresolved Safety Issue, the technical problems associated with it were solved and a report was likely to be issued in November or December of 1983. Tr. 1013-14. It is not surprising then that the Staff is not waiting for a revised Plan before it passes judgment on the Applicant's measures for preventing KRSKO-type water hammer events. The Staff is already satisfied that the Applicant "has [taken] or will take sufficient precautions to assure that a bubble-collapse water hammer such as occurred at the KRSKO plant will not occur at Byron." Staff's Proposed Finding B-45. In arriving at our own judgment of the Applicant's measures, it is this representation of the Staff we must consider, not the Applicant's conformity to unknown changes in a document designed to guide the Staff. Moreover, these changes will probably reflect the Staff's favorable judgment of Applicant's measures against KRSKO-type water hammer, and therefore, the revised SRP will probably contain no water hammer standard the Applicant has not already met.

A-84. In sum, Contention 9(a) is about what it says it is about: Not bubble-collapse water hammer in the preheater of the steam generator, not "classical" water hammer in the feedwater bypass line, not the integrity of the auxiliary feedwater system after a break in a feedwater line — rather, the prevention of bubble-collapse water hammer in the feedwater bypass line. Moreover, we need not wait for changes in the Standard Review Plan before we have all the law which is to be applied to the solution of the contention. To contribute to that "reasonable assurance" which 10 C.F.R. 50.57(a)(3) requires that the Applicant give to the public, the Applicant must show that it has conformed to General Design Criterion 4 by taking "appropriate measures" to prevent bubble-collapse water hammer in a feedwater bypass line. We now examine those measures.

*KRSKO Water Hammer Inference*

A-85. The Applicant presented the testimony of Robert W. Carlson, Principal Engineer in the Balance of Plant Systems Design Group of the Nuclear Technology Division of Westinghouse Corporation. In 1975 and 1976 he took part in a test program Westinghouse conducted at its Research and Development Center in Pittsburgh to study bubble-collapse water hammers, and in 1977 he took part in a program of study of such water hammers in steam generators like Byron's. Representing Westinghouse, he is helping the Applicant with the design and operation changes which Westinghouse recommends to prevent KRSKO-type water hammer, and which are discussed below in Board Findings A-94 to A-103. Mr. Pleniewicz, Assistant Superintendent of Operations at Byron, also testified. He is a member of the Onsite Review Committee, which reviews plant operating procedures and test results; and he is responsible for writing the procedures for operation of the feedwater bypass system. He testified about what the Applicant is doing to prevent a KRSKO water hammer from occurring at Byron.

A-86. The NRC Staff presented the testimony of Aleck W. Serkiz, a Senior Task Manager in the Generic Issues Branch of the NRC Office of Nuclear Reactor Regulation, and the Task Manager for Unresolved Safety Issue A-1 (water hammer). Mr. Serkiz testified about the documented information on the KRSKO water hammer, and about what the Applicant is doing to prevent a similar event at Byron.

A-87. The Intervenors presented no testimony on Contention 9(a).

A-88. Before a KRSKO-type event can be prevented, what happened at KRSKO must be reasonably clear. Apparently, no one witnessed the bubble-collapse water hammer which is thought to have occurred there. Instead, largely from the damage which was discovered during routine inspection of the feedwater piping at KRSKO in early August 1981, it is inferred that a bubble-collapse water hammer occurred in the feedwater bypass piping there in July 1981, a month before the discovery of the damage. Carlson, ff. Tr. 930, at 8; Tr. 1087 (Carlson). Mr. Carlson stated that "it's not possible to specify exactly what the conditions were when the bubble-collapse . . . occurred." Tr. 1087.

A-89. The first indications that something had gone wrong were the discoveries that paint on the auxiliary feedwater piping was blistered as far back as the auxiliary feedwater system pumps, and that the feedwater bypass piping was damaged. From the blistered paint, it is inferred that steam leaked out of the auxiliary feedwater nozzle of the generator into the feedwater bypass piping, and then into the auxiliary feedwater piping as far back as the auxiliary pumps. Carlson, ff. Tr. 930, at 8; Tr. 1087-88 (Carlson).

A-90. But steam could have leaked back to the pumps only if the water level in the steam generator had been below the discharge end of the auxiliary nozzle — the normal operating level being above (Carlson, ff. Tr. 930, at 9) — and if the check valves in the bypass and auxiliary piping had leaked. The check valves are designed to keep steam or water from leaking back out of the steam generator. *Id.* Thus, it is inferred both that the steam generator water level was below the discharge end of the auxiliary nozzle, and that the check valves leaked. In fact, the valves were known to leak (*id.* at 13), but we have no evidence except the blistered paint to show that the water level was low. *See* Tr. 1028-29 (Serkiz).

A-91. There was damage to piping both inside and outside the containment. Outside the containment building, there was negligible pipe movement; but inside the containment, hanger embedment plates were moved, hanger bolts loosened, and pipe clamps loosened and moved. Also inside the containment, the feedwater bypass piping was moved some, and there was a bulge on the upper surface of the bypass piping near the secondary shield wall. The bulged section was about 6 to 8 inches long and the bulged pipe about one-quarter of an inch greater in diameter than undamaged piping. Carlson, ff. Tr. 930, at 11-12. All this damage to the piping could be expected from a water hammer.

A-92. But for there to have been a water hammer, there must have been cold water in the piping at the same time the steam was there. It is known that cold water was intermittently fed into the piping by the auxiliary feedwater pumps during hot functional testing of the pumps in July 1981. *Id.* at 9-10.

A-93. Thus it is inferred that during hot functional testing in July 1981, the water level in a steam generator dropped below the discharge end of the auxiliary nozzle allowing steam to leak into the bypass and auxiliary system piping through leaky check valves, and that while steam was in the piping, cold water was fed into the piping by the auxiliary feedwater system pumps, which were turned on as part of hot functional testing. Some of the steam must have been trapped in one or more bubbles, which condensed rapidly and thus produced the hammer. *Id.* at 8, 9-10; Tr. 1086-90 (Carlson). *See also* Evaluation of Water Hammer Potential in Preheat Steam Generators, NUREG/CR-3090, Board Ex. 2, at 2-1 to 2-2. One sign that the inference is correct is that an impact noise was heard during the testing of the auxiliary feedwater pumps, during the hot functional testing in July 1981. Tr. 1088 (Carlson).

*The Westinghouse Recommendations*

A-94. On the basis of the foregoing account of what happened at KRSKO, Westinghouse has made four recommendations to the Applicant on how to avoid KRSKO-type water hammer at Byron:

(1) the steam generator water level should be maintained above the auxiliary nozzle discharge pipe as much as possible so that if backleakage does occur, water and not steam will leak back into the pipe;
(2) the auxiliary feedwater system check valves should be maintained to minimize backleakage;
(3) temperature sensors should be installed on the bypass piping close to the auxiliary nozzle to detect backleakage of hot water or steam;
(4) if backleakage is detected, the piping should be slowly refilled or the plant brought to a cold shutdown condition, depending on the circumstances; the recommended flow rate is on the order of 15 gpm.

Carlson, ff. Tr. 930, at 16.

A-95. Applicant affirms that it will follow all of these recommendations. Pleniewicz, ff. Tr. 896, at 4-8. None of them calls for a significant change in the way the Applicant had planned to operate the plant. Tr. 1119-20 (Pleniewicz). The next several findings discuss the recommendations separately.

A-96. Before Westinghouse's recommendations, it was already Applicant's plan to keep the level of the water in the steam generator above the discharge end of the auxiliary nozzle during all normal operations. However, during a turbine trip or a reactor trip, the steam generator water level could drop below the discharge end of the auxiliary nozzle; but these trips are infrequent. Pleniewicz, ff. Tr. 896, at 6.

A-97. Even if the water level drops below the auxiliary nozzle, there will usually be a continuous flow of feedwater through that nozzle, during which steam will not be able to leak back into the bypass piping. Feedwater will flow through the auxiliary nozzle continuously during power operations. Below 20 percent of full power, feedwater will enter the steam generator only through the bypass system. From 20 percent to 100 percent power, feedwater will enter the steam generator through the lower, main nozzle, but it will also continue to flow through the auxiliary nozzle. Witnesses for the Applicant gave inconsistent testimony on the purpose and amount of this flow through the auxiliary nozzle.[21] But both witnesses testified that there would be such a flow. During the normal nonpower operations of heatup, cooldown, and hot standby, feedwater enters the steam generator through only the auxiliary nozzle; but only a relatively small amount does so, not always enough for a continuous flow, and thus not always enough to keep steam from leaking back into the bypass piping; but plant operators are instructed to keep the flow of feedwater as continuous as possible, to reduce the likelihood of steam backleakage. Carlson, ff. Tr. 930, at 10-11; Pleniewicz, ff. Tr. 896, at 5-6.

A-98. In the unlikely event that the water level should fall at the same time that the feedwater is flowing only intermittently through the auxiliary nozzle, steam backleakage is still unlikely; for in keeping with a Westinghouse recommendation which is not among those set out in our Finding A-94, the Byron Station will have redundant check valves in each flow path by which steam or hot water could leak back into the auxiliary feedwater system.

A-99. Following the second of the Westinghouse recommendations, the Byron Maintenance Department has agreed to set up a regular schedule for testing these check valves for backleakage. Pleniewicz, ff. Tr. 896, at 7. When the reactor is shut down for maintenance and refueling, one of the two 6-inch valves and two of the eight 4-inch valves will be inspected, and if the inspected valves show problems, the rest of the valves will be inspected. Tr. 1108-09 (Pleniewicz).

A-100. During operation of the plant, continuous monitoring of the check valves in the bypass piping will be provided by the temperature sensors which Applicant, following Westinghouse's third recommendation for preventing a KRSKO-type water hammer at Byron, will install on each feedwater bypass line at Byron. *Id.* at 1109.

A-101. The sensors will be adjacent to the auxiliary nozzles and will detect backleakage of steam or hot water by sensing any increase in temperature in the bypass pipes. The plant process computer will be pro-

[21] Carlson, who testified on water hammer, said that from 20 percent to 100 percent power the flow through the auxiliary nozzle would be at 1 to 2 percent of the flow through the main nozzle, and that the 1 to 2 percent flow was designed to keep the auxiliary nozzle at feedwater temperature so that when feedwater had to be transferred from the main nozzle to the auxiliary nozzle, the thermal stress on the auxiliary nozzle would be low. Carlson, ff. Tr. 930, at 10-11. But Timmons, who testified on steam generator tube integrity, said that the flow through the auxiliary nozzle would be 10 percent of the total feedwater flow, and that Westinghouse was recommending this 10-90 percent flow split as a way to reduce steam generator tube vibration in the preheater area of the steam generator. Timmons, ff. Tr. 5908, at 23. Edison's Butterfield testified that the Applicant would be following this flow split recommendation. Ff. Tr. 5908, at 4. A 10 percent flow would, of course, also reduce thermal stress on the auxiliary nozzle. Both Carlson and Timmons work for Westinghouse, though in different parts of the Nuclear Technology Division. Apparently, Carlson, when he testified, did not know about the flow split recommendation. *See also* Tr. 6212-18 (Green).

grammed to set off an alarm whenever a sensor detects an abnormally high temperature in a bypass pipe. Pleniewicz, ff. Tr. 896, at 4.

A-102. In the unlikely event that the steam generator water level drops below the discharge end of the auxiliary nozzle while too little feedwater is flowing through the auxiliary nozzle and the check valves are leaking excessively, the bypass piping can be refilled slowly enough to prevent a bubble collapse, according to a study by the Westinghouse Research and Development Center. Pleniewicz, ff. Tr. 896, at 5. Thus, Westinghouse's last recommendation on how to prevent a KRSKO-type water hammer at Byron is that when the temperature sensors detect backleakage, the bypass should be refilled, or the plant brought to a cold shutdown. Westinghouse recommends a refill rate of 15 gpm. *Id.*

A-103. To implement this recommendation, Byron Station's Operating Department, under witness Pleniewicz's management, is developing procedures for the reactor operator to follow to purge backleaked steam from the bypass piping with a feedwater flow as close to 15 gpm as possible. The procedures will require that the low flow rate be maintained until the temperature of the bypass piping has returned to normal, and that a continuous flow be maintained until the cause of the high temperature is determined. *Id.*

*Preoperational Testing*

A-104. Before hot functional testing, every check valve which is installed to prevent backleakage into the auxiliary feedwater system will be tested for excessive backleakage. During hot functional testing, the ability of the tempering flow system to achieve the low, 15-gpm flow rate Westinghouse recommends for purging steam from the bypass piping will be tested, and Applicant will test for backleakage by stopping all feedwater flow into the steam generator and then monitoring the temperature of the bypass piping. *Id.* at 7. Also, Section 10.47 (p. 10-14) of the Byron Safety Evaluation Report (Staff Ex. 1) outlines other preoperational testing Applicant is committed to perform to determine the actual susceptibility of the Byron steam generator to water hammer.

A-105. It is the opinion of Applicant's witness Carlson that Applicant's implementation of Westinghouse's recommendations will reduce to an acceptable level the chance that a KRSKO-type water hammer will occur at Byron. Ff. Tr. 930, at 17. He could not say that the chance would be eliminated. Tr. 1104. He said that a KRSKO-type water hammer at Byron was not completely impossible. Tr. 1130. He had no probability value to assign to its occurrence. Tr. 1112. However, he said that it should not occur. *Id.*

A-106. The Staff agrees with the Applicant's conclusion. The Staff's witness, Mr. Serkiz, testified that the Applicant's implementation of Westinghouse's recommendations should preclude what happened at KRSKO. Ff. Tr. 940, at 5; Tr. 1015. He said that water hammer would occur at Byron, and with an unpredictable frequency (Tr. 982), but that he did not expect a KRSKO-type water hammer at Byron. Tr. 981-82.

A-107. But DAARE/SAFE would have the Board draw two conclusions opposed to the one the Applicant and the Staff agree on. The first is that because the Applicant has not shown that its plans for implementing and testing Westinghouse's recommendations "will specifically address feedline breaks or accident sequences," or changes the Staff contemplates making in the Standard Review Plan, the Applicant has not shown that the bypass piping is going to be "appropriately protected" against the effects of a KRSKO-type water hammer. Intervenors' Proposed Findings at 9-10.

A-108. We have already considered the first of DAARE/SAFE's conclusions. *See* our Findings A-80 to A-84. In brief, we found that Contention 9(a) was about the causes, not the consequences, of water hammer, and that it made neither practical nor legal sense to wait for changes concerning water hammer to be made in the Standard Review Plan, changes which, to the degree they are about KRSKO-type water hammers, are only likely to reflect the Westinghouse recommendations, for the Staff approves of those recommendations.

A-109. Perhaps more important is DAARE/SAFE's second proposed conclusion, that there is not enough known about what happened at KRSKO, and about what its generic implications may be, to conclude that there is no reason to have a significant health and safety concern about a similar occurrence at Byron. Intervenors' Proposed Findings at 10. DAARE/SAFE wanted there to have been more investigation by the Staff: The "Staff's investigation of the KRSKO event was neither more thorough or [sic] independent of that conducted by Westinghouse or Edison." Intervenors' Proposed Findings at 7. DAARE/SAFE points out that the Staff has made no direct investigation of the KRSKO water hammer but rather has relied on information from Westinghouse and the Applicant. Intervenors' Proposed Findings 27, 29-30; Serkiz, ff. Tr. 940, at 2; Tr. 949-52, 957 (Serkiz).[22] Yet despite the Staff's lack of knowledge about certain conditions at the time the water hammer occurred — what kinds of testing were being done, and what kinds of

[22] A member of the Staff has visited the KRSKO plant since the water hammer damage was discovered, but his visit was part of an International Atomic Energy Agency investigation of flow-induced vibration in steam generator tubes at the plant. *See* "Nuclear Power Safety Report to the Government of Yugoslavia," IAEA Report WP/5/1937, TA Report 1937, July 2, 1982. (Not in evidence.)

procedures followed (Tr. 1027-29 (Serkiz)) — the Staff concluded that the KRSKO water hammer was plant-specific (Serkiz, ff. Tr. 940, at 5; Tr. 1028 (Serkiz)) and without any generic implication other than that, when steam and cold water mix in a feedwater bypass line, there can be a water hammer. Tr. 1029 (Serkiz); Intervenors' Proposed Finding 34. DAARE/SAFE says, in effect, that more details about what happened at KRSKO might reveal some generic implications relevant to the public's health and safety, that the Staff's conclusion that the KRSKO water hammer is plant-specific too much resembles, "what we don't know can't hurt us."

A-110. However, the Staff's witness Mr. Serkiz is satisfied that the Staff knows enough about the KRSKO event to draw the conclusion that the KRSKO event should not occur at any United States plant which implements the Westinghouse recommendations. Tr. 1014-15.

A-111. The Staff bases its judgment that the KRSKO water hammer does not present a generic problem partly on NUREG/CR-3090, Evaluation of Water Hammer Potential in Preheat Steam Generators, December 1982 (Board Ex. 2), a study by Quadrex Corporation and E.G.&G. Idaho, Incorporated, under contract to the NRC. The Staff gave the consultants its information about the KRSKO event and has reviewed the study and judged it to be sound. Tr. 1015 (Serkiz). The consultants concluded that the Byron implementation of the Westinghouse recommendations makes "the occurrence of a KRSKO-type event" at Byron "not appear to be credible" (NUREG/CR-3090, at 3-13), and that "the KRSKO event was a plant-specific incident involving unusual test conditions and what appears to be multiple component failures (check valve gross leakage). . . . [E]xperience has shown that check valves are an effective means of preventing backleakage." *Id.* at 4-1. From the Staff's viewpoint, its claim that the KRSKO water hammer has no generic implications is not a claim that "what we don't know can't hurt us," but first that, not knowing in great detail what the circumstances were, the Staff could hardly specify any generic implication (Tr. 1029 (Serkiz)), and second, that enough *is* known about the KRSKO event to prevent anything like it from happening somewhere else.

A-112. DAARE/SAFE has given us no grounds to think otherwise. It has offered no evidence which casts doubt on either the accuracy of the information Westinghouse and the Applicant have provided the Staff, or the capacity of that information to support the inferences about the KRSKO event the Staff, the Applicant, and Westinghouse have drawn. DAARE/SAFE has not proposed other inferences from the same information, nor has it suggested hypotheses about plant conditions at the time of the KRSKO event which, if true, would support other inferences.

A-113. Equally important, DAARE/SAFE does not propose that we adopt a finding that Westinghouse's recommendations are not sound. DAARE/SAFE has expressed concern over check valve maintenance and the reliability of the temperature sensors, which monitor the valves (*see* Intervenors' Proposed Findings 12-17);[23] and DAARE/SAFE has urged that the Byron plant has a "test nature" (Intervenors' Proposed Findings at 10), for it is the first United States plant with a preheat steam generator of the sort KRSKO has now, a Model D steam generator modified to limit flow-induced vibration in the steam generator tubes. Intervenors' Proposed Finding 60; *see also* Tr. 1099-1101 (Carlson).

A-114. But DAARE/SAFE's concerns over maintenance and reliability imply that the Intervenors think the Westinghouse recommendations are sound rather than not, for unless the Intervenors think that maintaining valves and using reliable sensors help prevent water hammer, there is little point to their concerns. Moreover, to say that the Byron plant is something of a test case is not to say that what is being tested is not sound. DAARE/SAFE does not propose its own set of recommendations for preventing KRSKO-type water hammer.

A-115. We find that there is enough known about the KRSKO event to support sound inferences about what that event was and to support sound recommendations for preventing a similar event at Byron. We also find the KRSKO event has no generic implications for a plant which adopts the Westinghouse recommendations, and that Applicant's implementation of those recommendations at Byron makes bubble-collapse water hammer in the Byron feedwater bypass lines very unlikely contrary to the contention. We find, therefore, that the Byron bypass piping will be "appropriately protected" against bubble-collapse water hammer, as General Design Criterion 4 requires.

A-116. Nevertheless, the Applicant concedes, and the Staff agrees, that, although improbable, it is not impossible for a KRSKO-type water hammer event to occur at Byron. *See* our Findings A-108 and A-109, and references therein. Although strictly speaking, Contention 9(a) does not call for a consideration of the consequences of a KRSKO-type water hammer at Byron, it is not inappropriate to point out that even if a

[23] Although temperature sensors have been known to fail (Tr. 1106 (Pleniewicz)), we do not think that the failures justify DAARE/SAFE's claim that the sensors have "some unreliability" (Intervenors' Proposed Findings at 6). There is no evidence in the record to justify such a broad claim. Reliable machines can fail. Moreover, in his written testimony, Applicant's witness Pleniewicz speaks of installing "temperature sensor*s* on the feedwater bypass piping adjacent to the auxiliary feedwater nozzle." Ff. Tr. 896, at 4 (emphasis added). Redundancy of sensors should compensate for occasional failures.

bubble-collapse water hammer occurs in a feedwater bypass line at Byron, the health and safety of the public are not likely to be in danger — not at all in danger if the KRSKO event were precisely repeated at Byron, for despite the damage the KRSKO water hammer incident caused, KRSKO's auxiliary feedwater system and feedwater bypass system continued to function without impairment. Carlson, ff. Tr. 930, at 12; Tr. 1091, 1118 (Carlson).

A-117. But even assuming that a KRSKO-type water hammer could have enough force to rupture a bypass pipe — as Applicant's witness Carlson concedes it could have (Tr. 1110) — no radiation would be released, for the rupture would not be in a pipe which carries water from the reactor. A rupture in the bypass piping would not even indirectly lead to the release of radiation. The Applicant, as part of the consideration it was required to give to design basis accidents in the Byron FSAR, calculated the consequences of a total feedwater-line break and the consequent loss of secondary cooling from the steam generator which suffered the break. The calculations show no radiological release. Tr. 1020 (Serkiz). Also, a rupture in the bypass line is considered and prepared for under the heading of a main steam-line break, for the rupture would release steam. Tr. 1118, 1120-21 (Carlson). If a main steam line were to break, the auxiliary feedwater system would still provide cooling water to at least two effective steam generators. Tr. 1118 (Carlson).

### *4. ALARA as Related to Steam Generators*

A-118. Portions of Intervenors' Contentions 111 and 112 allege an absence by the Applicant of continuing attention to updating and replacing equipment at the Byron Station whereby the occupational exposure of onsite personnel will be kept *A*s *L*ow *A*s is *R*easonably *A*chievable (ALARA).

A-119. Intervenors' witness supported the contention by statements on undesirable characteristics of steam generators leading to the accumulation at various locations of solids bearing radionuclides which, in turn, induce exposures of maintenance personnel during the course of repairs and replacements. Morgan, ff. Tr. 1515, at 19-20.

A-120. A series of modifications to Westinghouse Model D4 and D5 steam generators proposed by the supplier and committed to by the Applicant can have, both directly and indirectly, a favorable consequence on the operational radiation exposure incurred by the Byron work force. These modifications have already been discussed at length in this decision. Some of the proposals, like the selection of structural materials, purport to reduce the radiation source term; others improve access and may lessen the time required for maintenance; still others are expected to enhance the performance and thereby increase the interval between maintenance functions.

A-121. These modifications as they relate to promulgation of the ALARA principle were addressed by Applicant's witnesses.

A-122. The primary source of activity leading to occupational exposure is fragments of used reactor fuel arising from damaged fuel pins and circulated through the primary coolant system. Tr. 1352 (Conway). These fragments are prone to settle out in the lower, divided, hemispherical plenum of the generator which accommodates, in its two sections, supply to and discharge from the generator tubes. This plenum is called the channel head.[24]

A-123. A second obvious source of radiation exposure, only obliquely alluded to in the testimony, is any manner of accumulations of solids, in the secondary side of the generator, carrying activated corrosion products from structures in that flow vicinity. An example is Co-60 from Co-59 in stainless steel. Tr. 4324 (Conway).

A-124. The modifications to the generators and to various attendant procedures and the expected bearing on the ALARA principle are:

(a) removal or reduction in the number of crevices and other potential pockets for the deposition of solids and in their capacity: termination of tubes no lower than the lower face of the tube sheet;

(b) selection of construction materials less subject to corrosion: installation of heat-treated Inconel-600 tubes and replacement of carbon steel tube support plates by stainless, both of low Co-59 content, in Model D5 in Unit 2;

(c) placement of a drain port at the lowest level of the primary channel head;

(d) provision of ring seals for the nozzles in the primary circuit at the channel head to shield against radiation from the primary piping;

(e) installation of sufficient and sufficiently sized ports for personnel access and egress, located to take advantage of internal structure as shielding and with quickly operated covers;

[24] Two schematics attached to the testimony are sorely devoid of relevant descriptive legends. The witness attempted to describe the generators by pointing, at the request of his counsel, to various portions depicted on a projected image of these drawings and verbally naming them. Any correlation between the tip of the pointer and the recording in the transcript is, needless to say, difficult. Conway, ff. Tr. 1309, following 12; Tr. 1311-14 (Conway).

(f) supply of the minimal quantity of instrument and other small access holes optimally placed to provide shielding yet allow for visual inspection, cleaning, etc.

(g) redesign of the topmost tube support place for increased rigidity of the U-bend, especially of those tubes with least spread between the strokes of the "U," thereby reducing the potential for strains in the bends;

(h) streamlining of the secondary flow to promote greater scavenging of solids.

Conway, ff. Tr. 1309, at 5-11; Tr. 1321, 1351, 4348-49, 4353-54 (Conway).

A-125. Additional and substantial reduction in occupational radiation exposure will be achieved by Applicant's commitment to complete before operation the modifications necessary to minimize tube wear due to flow-induced vibration. Blomgren, ff. Tr. 4126, at 17; Tr. 4385 (Gallo).

A-126. Removal of substandard tubes from service is accomplished by plugging one end of the tube. Using "Westinghouse mechanically patented plugs" [sic], this operation can be accomplished from without the generator "extremely rapid[ly]" with "almost non-existent" radiation exposure. Tr. 1350 (Conway).

A-127. A witness and employee of the Applicant referred to the availability and use of filters for removal of air [vapor] borne contamination within the generators upon opening of devices for expeditious removal and installation of "manways" covers necessary to the use of those 16-inch-diameter ports, and of the remotely operated eddy-current tube-inspection equipment. Van Laere, ff. Tr. 1707, at 22, 23; Tr. 1740-42 (Van Laere).

A-128. Through sufficient and judiciously located ports, deposits of solids, of composition unknown by the witness, can be removed from the secondary side of the generator, particularly the upper face of the tube sheet, by a method known as "lancing" by a two-stream device which simultaneously adds and removes liquid. The witness was unable to specify the flushing solution. Tr. 1344, 1349 (Conway).

A-129. The Board has developed some confidence in the position of the Applicant and the Staff that the modifications in the design, material, construction and operation of Westinghouse Model D4 and D5 steam generators will improve their performance. Superior performance should result in reduced maintenance and that, in turn, should lead to a lower radiation dose, within a given interval, to the onsite population. This confidence, however, implies no cause for any relaxation by the Applicant and Staff in their respective responsibilities in this important aspect of radiation protection. The Board believes the commitments by the Applicant to a broad reduction in the radiation source term to achieve the principle of ALARA is vastly superior to the distribution of any dose among an inordinate number of temporary workers.

A-130. The Board concludes therefore that the changes and modifications to be made to the steam generators will, in the long haul, reduce occupational radiation exposure and will strengthen the Applicant's conformity to the principle of ALARA.

## B. The Regulation of Industrial Exposure to Radiation As Low As Reasonably Achievable (ALARA)

### *League of Women Voters' Contentions 42, 111 and 112*

B-1. The contentions as litigated were stated as:

*Contention 42:*

As the Staff has recognized in NUREG-0410 and in the Black Fox testimony previously cited, occupational radiation exposure to Station and contractor personnel has generally been increasing in recent years, and violation of the limits of 10 C.F.R. Part 20 has been avoided by C.E.,[25] as by other licensees, by obtaining the temporary services of transient workmen rather than by devoting adequate effort to reducing exposures. Among other things, this practice results in using larger numbers of people and thereby increasing the risk of sabotage, operator error and similar safety-related hazards. Furthermore, new information on low-level radiation effects indicates that the Byron design basis will not provide safe operation. Accordingly, both because of the lack of assurance that proper exposure levels will be maintained and because of the practice of using transient workers, as a result of this serious and unresolved problem the findings required by 10 C.F.R. § 50.57(a)(3) cannot be made.

*Contention 111:*

C.E. has not met the requirements of NEPA and the Regs,[26] including but not limited to 10 C.F.R. §§ 50.34(a) and 50.36(a) because C.E. has not adequately monitored and provided a design base for the Byron plant which will keep radiation levels as low as achievable as required for operation of the plant to protect the health and safety of the public.[27] To keep radiation levels as low as achievable, C.E. should provide and utilize:

A. More adequate environmental and discharge monitoring of radioactive emissions from the Byron plant, which include:

[25] C.E. designates the Applicant, Commonwealth Edison Company.

[26] 10 C.F.R. Part 50, *et al.*

[27] It is noted that "as low as achievable" is not a requisite to operation of a nuclear power reactor. The regulation requires the occupational exposure to be kept "as low as is reasonably achievable." 10 C.F.R. 20.1(c).

(1) Monitoring devices at more locations within and without the plant site.
(2) Provisions for more frequent reading of monitors by independent analysts.
(3) Better monitoring devices which include:
(a) An automatic system of monitoring that notifies local authorities by an alarm when discharge emissions exceed design limits;
(b) Monitoring devices that measure differences in alpha, beta and gamma dose levels, which presently are not proposed to be considered and measured;
(c) Monitoring and recording of emissions of all dangerous long-lived radionuclides, including especially I-129 and plutonium;
(d) Bioaccumulative testing in a tiered system to assess the uptake of radioactive and chemical pollutants from bottom sediments or soil to lower organisms and to contamination of the food chain of man and other life.

B. More accurate calculation of design doses which can be accomplished by utilizing information from the improved monitoring suggested above and also by:
(1) Providing for and constant update and replacement of equipment and analysis to respond to new experimental and analytical results. Byron was licensed for construction, for example, when some (including C.E.) asserted improperly that there was a threshold to radiation effects;
(2) Including in calculation of doses the large transient populations in the low population zones around the plant, including schoolchildren when present in schools and others participating in recreational facilities;
(3) Including internal radiation doses caused by inhaled and/or ingested radionuclides which are deposited in different parts of the body where they give repeated radiation or until they are eliminated from the body;
(4) Including in calculation of radiation doses, cumulative doses to the general population outside the site boundary caused by overlapping circles of radiation from any nuclear facility (whether on or off the site), including Zion, Dresden, LaSalle, Quad Cities, and Braidwood Stations, as well as any new proposed facility and disposal facilities such as the Morris Waste Disposal Site; and
(5) Including in the calculation, calculation of doses to people by utilizing actual radionuclides for and in food, animals, plants, soil, water, and in other parts of the environment in and around the Byron site.

As a result, the applicable findings required by the Act,[28] NEPA, and the Regs, cannot be made herein. [By stipulation dated December 6, 1982, this contention was limited to in-plant radiation monitoring.]

*Contention 112:*

C.E. has not met the requirements of NEPA and 10 C.F.R. Part 20 because it has not adequately assessed the effect of radiation on plant workers and provided a design base for the Byron plant which will provide radiation levels as low as achievable. To keep radiation levels as low as achievable there is a need for better use of preventive measures to reduce radiation, including neutron, exposure levels to regular plant personnel and transient workers. These include but are not limited to:

[28] Atomic Energy Act of 1954, as amended.

(a) Plant designs for reducing amount of radiation exposure which take into account new evidence on low levels of radiation which were not considered in design of the plant.
(b) Improved record keeping of radiation exposures, including cumulative exposures both at the plant site and at other facilities.
(c) Better training of personnel to prevent radiation exposures, including more use of regular trained personnel rather than transient or temporary workers with little experience and training.
(d) Limiting exposure to high levels of radiation to volunteers and/or only older workers beyond the child-bearing age or others incapable of biological reproduction.
(e) Better education about radiation dangers to ensure cooperation of workers in keeping radiation exposures to a minimum.

As a result, the applicable findings required by the Act, NEPA, and the Regs, cannot be made therein.

B-2. Collectively these three contentions address the potential for exposure of both the employees at the Byron site and the public to radiation arising from the operation of the plant. Emphasis was put on the challenge to the Applicant established by the regulatory condition that occupational exposures should be kept *A*s *L*ow *A*s *R*easonably *A*chievable (ALARA). Although a number of specific issues were named in the contentions, many were not posed for litigation through evidence presented by the Intervenors.

B-3. Under requirements set forth in Commission regulations and elsewhere, the operator of a nuclear power station is obligated to protect its employees and those of other entities who are assigned to beneficial activities within the bounds of the installation for which proposed Commission action is sought. These obligations are set forth in detail in Parts 19, 20 and 73 of Title 10 of the Code of Federal Regulations and in the Commission's Regulatory Guides 8.8 and 8.10.

B-4. Part 20, Appendix B specifies permissible limiting concentrations of radionuclides in air and water carriers. These limits provide guidance in the control of employee activities in order to avoid exposure to excessive radiation.

B-5. Permissible cumulative doses to employees are specified in 10 C.F.R. 20.101(a). An example of these specifications, expressed in rem/calendar quarter, is 1.25 whole-body.

B-6. Additionally to the guidance in 10 C.F.R. 20.101(a), *supra,* 10 C.F.R. 20.1(c) says that every reasonable effort *should*[29] be made to

[29] The language of the regulation uses the permissive term *should* in the ALARA statement [10 C.F.R. 20.1(c)]. The Staff asserted it considers ALARA mandatory and interprets *should* as a mandatory *shall.*
*(Continued)*

maintain the radiation exposures of the employees of a licensee to a quantity as low as is reasonably achievable (the ALARA principle), taking into account the state of technology and the economics of improvements in relation to benefits to the public health and safety.

B-7. Part 20 also addresses ancillary topics including personnel monitoring and record keeping, bioassay services, exposure reporting, area access controls and alarms. Training of personnel in the control of radiation exposure and other health risks appears in 10 C.F.R. Part 19. Additional access controls and other security matters are noted in 10 C.F.R. Part 73.

B-8. The litigation of these contentions is described by the Applicant in its summary findings as:

> Although the specific wording of the three contentions encompasses numerous issues, the League presented evidence on only a few topics. Its primary area of concern appeared to be whether Applicant has accurately assessed the potential risks from occupational exposure to radiation.[30] The remaining issues the League raised relate to the general ability of Applicant's ALARA program to keep occupational radiation doses ALARA and to specific concerns about radiation exposures and protection. These issues include: (1) whether Applicant's dosimetry program for monitoring radiation exposures to workers is sufficient to maintain doses ALARA; (2) whether Applicant's program for monitoring radiation levels inside the plant will provide accurate results; (3) whether Applicant's procedures for maintaining occupational exposure records are sufficient to maintain doses ALARA; (4) whether the size and training of Applicant's health-physics staff is sufficient; (5) whether workers at the Station, including contract or temporary workers who are not Applicant employees, are adequately trained in how to keep radiation doses ALARA; (6) whether Applicant's policies on radiation exposure to declared pregnant women will adequately protect the fetus; (7) whether the risk of possible industrial sabotage by anyone, especially a contract worker, is sufficiently small so as to maintain doses ALARA; and (8) whether the design bases of Byron Station and, more specifically, its steam generators, include features for reducing occupational radiation exposure.

B-9. The portions of these contentions respecting the ALARA principle as it relates to steam generators, was discussed in that connection, *supra.*

B-10. A total of ten witnesses gave testimony on these contentions. The Applicant presented Jacob I. Fabrikant, a physician and University of California professor of radiology and biophysics, who addressed the

Tr. 1910 (Lamastra). In any event matters such as these are subjects in Technical Specifications which, in turn, provide the Staff with an authoritative lever to compel an applicant or licensee to comply with accepted practices and conventions. Tr. 1908, 1909, 1914 (Lamastra).

[30] The effective radiation in this discussion is described as linear energy transfer (LET) radiation and is characteristic of electrons, x-rays and gamma rays. Fabrikant, ff. Tr. 1399, at 8.

subject of health effects from low-level radiation; Frank Rescek, technical services engineer for Commonwealth Edison, addressed the corporate ALARA program and its dosimetry recordkeeping procedures and training. James R. Van Laere, radiation protection manager at the Byron site, discussed the ALARA program, the health physics staff and the in-plant monitoring program. Gerald P. Lahti, who is in charge of shielding and radiological safety at Sargent & Lundy, the architect-engineer, testified about Byron plant design for reducing occupational exposure. Lawrence Conway, an engineer of Westinghouse, discussed mechanical and metallurgical design features of Byron's steam generators to reduce occupational radiation exposure. Dr. Conway's testimony formed a portion of the Board's findings on ALARA in steam generators. Section A, *supra.* Finally, Jerome L. Roulo, security administrator, addressed increased risk of sabotage from the use of temporary workers at Byron.

B-11. The Intervenors presented K.Z. Morgan, a consultant on radiation safety matters and a professor at Appalachian State University, whose testimony addressed health effects from low-level radiation and the issue of radiation safety at Byron.

B-12. For the Staff, Michael A. Lamastra and Edward F. Branagan, Jr., both health physicists, discussed the Staff's review of the Applicant's radiation protection programs and the resulting conclusions including estimates of health effects due to occupational radiation exposure. Robert F. Skelton, a plant protection analyst, testified on the security aspects of the use of temporary workers at Byron.

B-13. Basic to a portion of these contentions and for a major part of the evidence and testimony is the dependence of health effects on the character, quantity and intensity of the radiation to which employees may be exposed.[31]

B-14. That there is no incidence of cancer risk as the result of exposure to low levels (~10 rem) of radiation is not at issue. Epidemiological studies of exposed human populations and of laboratory animals do show some risks to have occurred at such low levels. Those direct measures, however, are inadequate to allow direct evaluation of that risk and, accordingly, recourse must be taken to mathematical models to extrapolate the risk. Fabrikant, ff. Tr. 1399, at 75.

B-15. In discussion within the radiation-effects community is the shape of a curve relating health effects and exposure. A linear response

[31] The Intervenors refer, in some of their proposed findings, to a "Report to Congress, Problems in Assessing the Cancer Risks of Low-Level Ionizing Radiation Exposure," dated January 2, 1981 and designated as EMD, 81-1 apparently issued by the U.S. General Administrative Office. This report was never admitted as evidence in this proceeding and cannot serve as a basis for findings.

dose relation is accepted by many of the concerned groups and individuals and is believed by them to be conservative. Others propound a "supralinear" relation which makes the effects of small doses (~10 rem) per unit exposure more severe than the effects of greater specific exposures. Other models follow a quadric relation. *Id.* at 18.

B-16. The consequences of radiation exposures to personnel are determined from a quantitative measure of the exposure and a factor called a risk estimator. The estimator is the potential lifetime risk of excess cancer incidence, to a large population, of some carcinogenic or genetic effect from a low-dose exposure of 1 rem of average quality. A value of 0.0001 or less for the estimator is based on the linear interpolation of the effect-exposure relation between the naturally occurring spontaneous incidence and the incidence observed after exposure to intermediate-to-high doses and dose rates. A value of this order is accepted as an upper limit by standards-setting and investigative groups. *Id.* at 61, 62.

B-17. The Staff determined the cancer death risk estimator for Byron to be 135 potential fatalities per million person-rem, in agreement with that of the Applicant's witness. The Staff estimator evaluation is based on the recommendations of the National Academy of Sciences Committee on the Biological Effects of Ionizing Radiation (the BEIR-I Report). It is consistent with values recommended by major radiation protection organizations.[32] Staff Ex. 2, at 5-23; Branagan, ff. Tr. 1883, at 5-7.

B-18. The Intervenors' witness offered into evidence a value of the estimator of the order of 0.001 excess cancer fatalities per person-rem for doses up to 10 rem. Morgan, ff. Tr. 1515, at 8. This value was derived from a review of some ankylosing spondylitis data, the data from the Hiroshima/Nagasaki weapons-effect measurements, and data from medical/exposure records of deceased former employees of the nuclear works at Hanford, Washington. The last were compiled and analyzed by Mancuso, *et al.,* summarized in National Research Council, The Effects on Population of Exposure to Low Levels of Ionizing Radiation: 1980, BEIR-III, National Academy Press (1980), at 455 *et seq.*

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[32] Values of the estimator reported by the Staff from other sources, all in mortalities/million person-rem, based on the linear model except as noted are:

| | | |
|---|---|---|
| FES (1972 BEIR-I) | 135 | (noted above) |
| BEIR I, with population update | 115 | |
| United Nations (1977) | 75-175 | |
| International Commission on Radiation Protection | 100-125 | |
| BEIR-III (1980) with modified linear-quadratic model | 67 | |

Branagan, *et al.,* ff. Tr. 1883, Attachment F.

B-19. The witness found the scatter in these data too great to warrant even a least-squares analysis of them. The estimator was obtained by a visual, non-computer-assisted fit. Tr. 1586-87 (Morgan).

B-20. Permissible radiation exposures to be incurred by the Byron work force were recommended by Intervenors' witness based on the 0.001 estimator. It was stated, for example, that an upper limit of 400 person-rem per year per plant be established and enforced.[33] If the average exposure were held to 300 person-rem per year, there would be, by this witness' estimate, twelve excess cancer deaths among its employees over the 40-year operating life. Morgan, ff. Tr. 1515, at 8.

B-21. The Staff, utilizing a cancer risk estimator of 1.35 potential excess cancer deaths/$10^4$ person-rem/year derived from the BEIR-I study, and an average annual work-force exposure of 440 person-rem/year determined from doses experienced at operating PWRs, arrived at 2.4 potential cancer deaths among the operating personnel during the projected life of one of the Byron units. This projection is to be compared to the twelve deaths estimated by Morgan. Branagan, ff. Tr. 1883, at 5.

B-22. In summary the Board recognizes the existing uncertainty within the current generation of scientifically qualified individuals in the interpretation of the grossly insufficient data now available on health effects of personnel exposures from radiation at rates of about 5 rem/year, characterized as low-level radiation. A basic question, of course, is the behavior of the (excess) health-effect versus dose (above natural background) relation at these low exposures. That the relation is linear, *i.e.*, damage by 10 rem is an order of magnitude greater than an exposure to 1 rem, is in controversy. In the recent past, the supralinear model has surfaced which, in effect, says that, rem for rem, a low exposure dose is more damaging than an exposure at higher rates. There are also proponents of other models which show the effects to be opposite relative to the linear concept, that is, effects per unit dose are lower at low doses. A consequence of this difference in the interpretation of the data appears in the establishment of the health effects of cumulative doses, that is, in the validity of the practice of merely summing the incremental exposures of an individual. The mode of interpretation also affects the population dose incurred through the use of temporary workers, those subjected to

[33] In a document distributed subsequent to closure of this record, the Staff reported a committal by the Applicant to an occupational radiation dose estimate of 400 person-rem/year/unit. Supplement 3 to the SER, Byron Station, NUREG-0876, November 1983, *quoting from* Amendment 40 to the Byron Final Safety Analysis Report.

a low (within specified limits) but concentrated radiation dose. This practice is alleged to have the effect of spreading low-level doses, with their relatively greater risk, over a larger segment of the population.

B-23. The Board can recognize the basic problem as a true scientific difference among knowledgeable and responsible individuals which can be resolved only by data sufficiently extensive to be statistically meaningful. A long time will be required.

B-24. In the meantime, the Board accepts the linear hypothesis, proposed by the Applicant and corroborated by the Staff, on the basis of the preponderance of supporting evidence in this proceeding, both direct and by reference. The Board is aware that such decisions are revocable and controllable at such time as new data and analyses show remedial measures to be required. Accordingly the Board rejects those portions of the Intervenors' argument on these contentions which concern health effects of radiation.

B-25. The Applicant has a corporate program to control and monitor potential radiation exposures to its employees and those of its onsite contractors. The goal of the program is to operate and maintain the Byron plant so that those exposures are well below legal limits and as low as is reasonably achievable. Rescek, ff. Tr. 1157, at 3; Van Laere, ff. Tr. 1707, at 2-11.

B-26. The policy is stated in Applicant's "Policy and Procedures for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable."[34] Applicant Ex. 3, Tr. 1159. Included in the policy is the requirement of a Nuclear Station ALARA Review Committee (Applicant Ex. 3, at 8), which serves as the executive body for ALARA reviews, decisions, audits and "post-mortems" including the development of ALARA goals. The membership of this ALARA Committee includes, in part, the Station Superintendent, as chairman, the Radiation-Chemistry Supervisor, and the Station ALARA Coordinator. *Id.* One of the Intervenors' proposed findings incorrectly claims the ALARA Coordinator not to hold membership.

B-27. Within this organization, the principal health physicist within the corporate structure (Rescek) and the Station Radiation Protection Manager (Van Laere) have unimpeded access to higher levels of supervision, including the Station Superintendent in the latter case, thereby bypassing the organizational line, on matters of radiation protection. Tr. 1213 (Rescek); Tr. 1722 (Van Laere). These statements

[34] Corporate policy, serving as guides to the promulgation of radiation protection procedures at specific nuclear plants, is described in "Radiation Protection Standards," February 28, 1982. Applicant Ex. 2, Tr. 1159.

in the testimony as referenced contradict some of Intervenors' proposed findings.

B-28. Several specific facets of the radiation monitoring program contributing to achievement of exposures as low as reasonably achievable will be discussed, *infra*.

B-29. Applicant agrees with the Intervenors that in a March 1980 appraisal the NRC cited eight areas of deficiency in the health physics program at another of the Applicant's nuclear plants. Intervenors Ex. 1. It was emphasized, however, that the Applicant took prompt remedial action to the satisfaction of subsequent NRC inspections. Tr. 1244-56 (Rescek).

B-30. Personnel exposure to individual employees will be measured by several types of detectors sensitive, collectively, to alpha, beta and gamma radiation and to neutrons. The most likely exposure to personnel will be from gamma rays. Fabrikant, ff. Tr. 1399, at 8.

B-31. Beta and gamma-ray exposures will be monitored by film badges to be worn by employees. The badges are routinely read through the biweekly services of a vendor. In the event of an emergency, 4-hour badge-reading service is available. Additionally individuals carry ionization chambers sensitive to beta and gamma radiation. The exposures detected by these devices are measured daily and the results, summed over two weeks, are compared to the film badge value. Discrepancies are investigated. Each film badge datum is recorded in the individual's personnel file. Rescek, ff. Tr. 1157, at 11; Tr. 1211 (Rescek).

B-32. The accuracy of film badge results is investigated periodically by submitting to the vendor for processing an unidentified badge purposely exposed to a known level of radiation. Rescek, ff. Tr. 1157, at 12.

B-33. Individuals assigned to variable and relatively high radiation fields are provided with instruments which emit an intermittent signal, of high pitch, at a frequency determined by the intensity of the field, thereby providing an audible indication of that variation in radiation exposure. Tr. 1751 (Van Laere).

B-34. Additionally exposures to extremities of individuals are measured by finger rings equipped with thermoluminescent dosimeters; to whole bodies through observations with portable instruments sensitive to beta and gamma radiation or neutrons. Rescek, ff. Tr. 1157, at 10.

B-35. A principal neutron detector embodies a substance called CR 39. This is a carbonate compound (possibly allyl diglycol carbonate) in which carbon and hydrogen recoils from neutron collisions produce tracks which, in turn, can be measured to obtain neutron intensity and energy information. Tr. 1274 (Rescek). Intervenors' witness was in accord with the use of CR 39. Tr. 1650 (Morgan).

B-36. More than 200 fixed radiation monitors, including air samplers, will be installed in the Byron plant to provide notice of unusual emissions. These will provide, as appropriate, element and even isotopic identification and will actuate audible signals. Van Laere, ff. Tr. 1707, at 20-21. Although Intervenors' witness opined that the number of monitors should be increased (Morgan, ff. Tr. 1515, at 21), he was unable to quantify that alleged need because he was not knowledgeable of the presently designed number. Tr. 1662 (Morgan).

B-37. Provision has been made for physical examination of individuals believed to have exceeded permissible body burdens, *i.e.*, internal exposures. These examinations center around bioassays which include whole-body (radiation) counting, radio-chemical analyses of excreta, nose swabs. The principal one of these techniques is whole-body counting.[35] Van Laere, ff. Tr. 1707, at 16-18; also Exhibit 8 attached to Van Laere's testimony; Tr. 1211-12 (Rescek). The whole-body counting will be done on site; excreta analyses will be made by an independent contractor. Van Laere, ff. Tr. 1707, at 16-17.

B-38. All female employees shall be instructed on the potential risks of radiation exposures during pregnancy. Those who know of or suspect pregnancy are limited in exposure to no more than 500 mrem during gestation without prejudice to present or future employment positions. Tr. 1194 (Rescek).

B-39. The exposure history over a few weeks preceding confirmation of pregnancy is examined to determine any unusual exposure which could influence subsequent job assignments to retain the overall exposure during pregnancy to within the limit. Tr. 1193 (Rescek).

B-40. Intervenors' witness challenged the Applicant's ability to retain the overall exposure to 500 mrem during pregnancy if sufficient strontium-90, for example, previously inhaled and/or ingested were simultaneously producing an internal exposure in excess of 500 mrem. Tr. 1643 (Morgan).

B-41. The Applicant points out that airborne strontium-90 is expected to be present in Byron in quantities very much below 10 C.F.R. Part 20 guidelines. Pre-conception internal accumulations of strontium-90 would have been detected by whole-body counting. Were an amount detected at or approaching the specific limit (0.5 rem/9 months), the

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[35] Annual whole-body counts are made of all Station personnel; individuals who frequent areas with airborne radioactivity have these counts more frequently. Tr. 1212 (Rescek). Applicant employees newly assigned to Byron and incoming contractor personnel shall receive a whole-body count. Van Laere, ff. Tr. 1707, Exhibit 8 attached to Van Laere's testimony, at 1, 2; Applicant Ex. 2, ff. Tr. 1159, at 42. Exhibit 8 attached to Van Laere's testimony is Applicant document BRP-1340-1, approved January 31, 1982, entitled "Personnel Monitoring for Internal Radioactive Contamination" and gives detailed procedures for bioassays.

employee would be precluded from areas where strontium-90 could conceivably be present. Smaller internal quantities together with previous external radiation history would collectively govern future work assignments. Tr. 1194-97 (Rescek).

B-42. It was also noted that when whole-body counting indicates an internal burden as much as 3 percent of the NRC permissible burden, an investigation of the source of the burden and a careful evaluation of it is inaugurated. The 3 percent value is established by the Applicant and is below standard values by a factor of three. The NRC permissible burden is equivalent to a bone exposure of 15 rem.[36] A burden only 3 percent as large, at equilibrium, will produce an exposure of 1.4 rem over a normal term of pregnancy. The exposure during the first nine months of a freshly deposited long-lived source will be significantly less. Tr. 1199-1202 (Rescek).

B-43. Intervenors' witness could make no specific proposal to solve this problem. He merely brought it to the attention of the Applicant. Tr. 1643 (Morgan).

B-44. Records of dosimetric data on individual employees shall be prepared and computerized within the corporate structure in accordance with the requirements of 10 C.F.R. 20.401, and as specifically directed in the instructions for the use of Form NRC-5 and as prescribed by American National Standard 13.6-1972. Lamastra, ff. Tr. 1883, at 16-17. These records are available for inspection by the individual concerned. Tr. 1280 (Rescek).

B-45. The Applicant shall obtain from each new employee who may be assigned to work in a radiation field a completed Form NRC-4 which reports all prior occupational experience in which exposure to radiation was incurred together with corresponding values of whole-body doses. In this manner, some control can be exercised of the total lifetime exposure of transient workers.

B-46. The Applicant shall verify the previous radiation-exposure history of its employees through inquiry to those earlier employers. Through this system of records built around Form NRC-4, considerable control can be enforced over the current radiation-exposure pattern. Its thoroughness depends on the accuracy and completeness of the information obtained from the new employee, both as to his radiation history and to his former employer. Tr. 1231 (Rescek). Absent any central industrial clearinghouse for this historical data, no better method is presently apparent. Tr. 1299 (Rescek).

[36] The record is incomplete in that the witness spoke merely of 15 rem. The Board presumes this is 15 rem/quarter, the time unit frequently encountered in 10 C.F.R. Part 20.

B-47. Applicant is committed to establishing and maintaining an information system consistent with Regulatory Guide 8.7, Occupational Radiation Exposure Records System. Lamastra, ff. Tr. 1883, at 15.

B-48. The Applicant shall administratively impose a limit of 50 mrem on an individual's daily dose and a weekly dose of 300 mrem. Actions in which these limits might be exceeded require issuance of a Radiation Work Permit which details not only the task but precautions necessary to minimizing further exposure. Rescek, ff. Tr. 1157, at 13; Tr. 1285 (Rescek).

B-49. The Health Physics Group at Byron is expected to be composed of forty-three individuals at the time of two-unit operation. Twenty-eight of these will be classed as radiation chemistry technicians who, by estimate, will devote 70 percent of their effort to health physics matters. At present, during construction, the Health Physics Group is partially staffed by eighteen technicians, four (college-trained) health physicists, three health physics foremen, one health physics engineering assistant — a total of twenty-six not including the Radiation Protection Manager (Applicant's Response to League's Proposed Findings says 23). This roster is six persons short of the Unit 1 requirements and seventeen short of the required staff when both Units 1 and 2 are in operation. Tr. 1717, 1759 (Van Laere). This accounting of personnel is contrary to statements made by the Intervenors in their Proposed Finding 27.

B-50. Within the present staff is considerable experience.[37] Formal training, by the Training Department, and on-the-job training will be provided to meet the requirements of Regulatory Guide 1.8, "Personnel Selection and Training," and of American National Standard 18.1, "Selection and Training of Nuclear Power Plant Personnel." Lamastra, ff. Tr. 1883, at 11.

B-51. All employees at the Byron Station, including contractor and temporary workers, are subjected to the training outlined in "Instructors Guide for Nuclear General Training" (N-GET) which includes a 1-inch-thick section on radiation, its detection and measurement, its effects, and methods of protection against it. Applicant Ex. 4, ff. Tr. 1159. Retraining is through an annual refresher course. Tr. 1189 (Rescek).[38]

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[37] Three have degrees in health physics; eighteen technicians have participated in a refueling outage at an operating power plant. Tr. 1719, 1721 (Van Laere).

[38] Subsequently, at Tr. 1243, this witness implied that only individuals who work in "radiation areas" will be trained. In the same reply, he exempted "[only] visitors." The Board concludes that all employees are subjected to N-GET.

B-52. The initial training of individual employees, the N-GET program, is given in a single day; the refresher course may take 4 to 6 hours. Tr. 1190 (Rescek); Tr. 1726 (Van Laere).

B-53. Additionally to the N-GET training program, special proficiency training will be offered to workers who will be performing complex tasks in selected high-radiation areas. Mock-ups of equipment on which workers may first practice may be beneficial in reducing the time and, hence, the exposure required. Rescek, ff. Tr. 1157, at 21.

B-54. The Intervenors note, in their finding, that no training in radiation protection will be presented to employees in contrast to the contents of Applicant Ex. 4. The Board is concerned that the information cited in that outline (Ex. 4) is presented in a one-day session.

B-55. Portions of Contentions 111 and 112 which allege the design of the Byron Station is deficient in that employees will not be subjected to radiation as low as is reasonably achievable, was addressed by Applicant's witness Conway to the extent that the deficiency is embodied in the design of steam generators. Conway, ff. Tr. 1309. As noted above, the conclusion of that testimony is that the steam generators at the Byron Station include design features intended to minimize occupational radiation exposures. Conway, ff. Tr. 1309, at 11.

B-56. Potential exposures from other radioactive sources are minimized by conventional radiation shielding techniques. These include provision of shielding and/or distance between radiation sources and employees, control of the time employees are permitted in the radiation field, and reduction or removal of the radiation source. Lahti, ff. Tr. 1830, at 4.

B-57. Although radiations of various character can be found in and around a nuclear plant, those most commonly encountered in radiation protection design requirements are neutrons and gamma rays. Alpha particles are inherent in the reactor fuel and both alpha and beta particles are decay products of fission fragments and transuranics in the fuel. The range of these in most materials is sufficiently short that adequate shielding is provided by their enclosures, the fuel clad and the pipe conducting a solution in a radioactive waste recovery operation as examples. Tr. 1831, 1843 (Lahti).

B-58. Several common materials — water, ordinary concrete, steel, lead — are effective as shields against neutrons and gamma rays. Tr. 1843 (Lahti).

B-59. The design of the Byron Station was a cooperative effort by the Westinghouse Electric Company, the supplier of the reactors, the steam generators and ancillary equipment, and the architect-engineer, Sargent & Lundy, the entity responsible for the remainder of the plants. Lahti, ff. Tr. 1830, at 3.

B-60. To the extent foreseeable the designs incorporated capabilities for inclusion of one or more of the conventional techniques to achieve minimal exposures to operating personnel. The manner of this inclusion was guided by the time, frequency and space requirements of conceived maintenance operations. For example, now removable, staggered, stacked concrete-block dry walls will suffice for operations expected infrequently, or where space limits preclude a permanent wall, to avoid exposure to workers otherwise required, in earlier designs, to construct the wall and to subsequently remove it with pneumatic tools. Tr. 1872 (Lahti).

B-61. A particular operation with solutions of radionuclides may require a tank, a valve and a pump. The last two may be more susceptible to malfunctions while the first may be a strong source of radiation. These items, in current design, may be compartmentalized by shielding adequate to allow repair of the pump without lessening the radiation source in the tank. The pump, or valve, may be repairable with long-handled tools utilizing distance as an effective shield. Alternatively the pump can be partly decontaminated by draining and back flushing with a solution bearing a solvent for radioactive deposits. Adjacent piping, after draining, may be temporarily enclosed in a lead blanket. Lahti, ff. Tr. 1830, at 4; Tr. 1854-55 (Lahti).

B-62. The above merely exemplify the many design approaches which supplement sound operating procedures to achieve occupational exposures as low as reasonably achievable. Tr. 1855-56 (Lahti).

B-63. League Contention 42, in part, cites an increased possibility of industrial sabotage arising from the presence, during operation and maintenance, of undue personnel as a consequence of the employment of temporary workers. At least by implication, such sabotage is linked to undesirable radiation exposures.

B-64. The Applicant and the Staff addressed this sabotage issue through a review of the security screening process of individual temporary workers which gives reasonable expectations of the absence of untoward actions by those individuals which might jeopardize the Byron project. There was only passing reference to Applicant's employees. Tr. 1377 (Roulo).

B-65. At a time no later than 90 days before fuel loading in the Byron reactors, each contractor employing persons requiring unescorted access to the installation shall have in place a personnel security screening plan which has been reviewed and approved by the Applicant.[39] Roulo, ff. Tr. 1356, at 2.

B-66. An employee, acceptable after the investigation, shall be issued, upon his entry into a secured area, a numbered identification badge bearing his photograph. To receive this badge the employee shall identify himself by name and Social Security number.[40] Roulo, ff. Tr. 1356, Appendix A, at A-2; Tr. 1360 (Roulo).

B-67. Grant of this security clearance is based on trustworthy employment by the present employer for a continuous period of three or more years; or a favorable result of a background search with prior employers, personal references and professionals skilled in detecting potential aberrant behavior.[41] Roulo, ff. Tr. 1356, at 2, 3.

B-68. The screening process, observation of contractor employees, and a verification of proper documentation of these security matters by contractors is audited continually by the Applicant. *Id.*; Tr. 1380, 1381 (Roulo).

B-69. The Staff has reviewed the Byron Nuclear Power Station's "Physical Security Plan," Revision 7, dated October 8, 1982 and December 22, 1982 and has found the overall Byron Station security plan, including the potential for sabotage by temporary workers, to satisfy the requirements of 10 C.F.R. 73.55(b) through (h) provided the proposed plan is effected. The Staff concludes that, although the presence of temporary workers will increase the potential for sabotage, the overall risk remains acceptably small. Skelton, *et al.*, ff. Tr. 1883, at 11, 12, 13. The Applicant asserts that the presence of temporary workers will not increase the potential for industrial sabotage at Byron. Roulo, ff. Tr. 1356, at 5. The Board considers the Staff's position more tenable.

B-70. The Applicant has committed in its Final Safety Evaluation Report (FSAR) to the design and operation of the Byron plant in a manner consistent with the ALARA principle stated in Section 12.1 of the Standard Review Plan, NUREG-0800 (1981) (Lamastra, *et al.*, ff. Tr. 1883, at 4, 14) and, further, Staff has concluded that the Applicant has established an ALARA program meeting the acceptance criteria of NUREG-0800 following the guidance of Regulatory Guide 8.8,

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[39] This testimony continually refers to "pre-employment" screening which, when taken literally, says only new employees of a contractor are so investigated. The Board believes this expression to be a minor misstatement and, in truth, each employee of contractors needing access will require investigation regardless of the length of his tenure.

[40] The specifics of this sensitive topic were aired in an *in camera* session on March 9, 1983 attended by a representative of each party. The transcript reporting that session is not in the public record. No party proposed findings on the *in camera* testimony.

[41] The testimony was clarified at Tr. 1378-79.

"Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Station Will Be as Low as Is Reasonably Achievable," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable."

B-71. Based on the evidence presented in this proceeding, the Board concludes that the Applicant has the capability to successfully determine, control and limit the radiation exposure of its employees and of others assigned to the Byron site so as to be in accord with applicable regulations. To achieve this objective the Applicant has in place policies and procedures for maintaining occupational radiation exposures as low as reasonably achievable through radiation protection standards, a growing health physics staff commensurate with need, adequate dosimetric instrumentation and other equipment, medical and recordkeeping capabilities, and an employee training program. By its commitment to the ALARA principle, Applicant is required to consider advances in the several applicable disciplines, including those as divergent as process equipment[42] and security measures, in order to keep implementation of the policy current with knowledge.

## C. The Environmental Costs of Severe Accidents

C-1. Intervenors' contentions challenge the effect on the environment, including the health and safety of the public, of severe potential accidents which have a non-zero probability of occurrence at the Byron Station. These contentions, as admitted by this Board, are stated as:

*League Contention 8:*

Neither C.E. nor the Staff has presented a meaningful assessment of the risks associated with the operation of the proposed Byron nuclear facility, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). Studies carried out by the NRC have identified accident mechanisms, considered credible, which would lead to uncontrollable accidents and release to the environment of appreciable fractions of a reactor's inventory of radioactive materials. Traditionally, these accident potentials have been downplayed or ignored on the basis of the Rasmussen Report.[43]

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[42] In Applicant's Finding 111 of the ALARA program (at 78) one finds that the commitment to the muchly discussed modifications of the Byron steam generators "mean[s] *no* radiation will occur . . ." (emphasis added). The Board finds the statement is patently untrue in the presence of natural background radiation. Further, the Board is unable to locate any bases for the statement in the accompanying citations.

[43] "Rasmussen Report" designates the Reactor Safety Study (RSS) NRC report WASH-1400 (October 1975).

However, the Lewis Committee has now called into serious question the entire methodology, as well as the findings and conclusions, of the Rasmussen Report, which led the NRC to withdraw official reliance on the Rasmussen Report, yet the Staff still regulates upon the validity of the basic conclusions therein. In addition, NRC Staff studies, which are not common public knowledge, have cast doubt upon numerous of the specific conclusions of the Rasmussen Report. For example, in one secret NRC study,[44] estimates of the "killing distance" were made, referring to the range over which lethal injuries would be received under varying conditions from the release of radioactive material in a nuclear power plant accident. Depending upon prevailing weather conditions, this "killing distance" was estimated to be up to several dozen miles from the accident-damaged reactor. Unpublished document from Brookhaven National Laboratory, USAEC. In addition, the Liquid Pathways Study, NUREG-0440 (February 1978), highlights the incomplete safety assessment currently performed by the NRC, particularly with respect to incomplete review of all credible accident sequences. A General Accounting Office report[45] pertaining to that study criticizes the NRC's failure to consider core-melt accidents in assessments of relative differences in Class 9 risks. The March 7, 1978 letter from the NRC's Mr. Case to the Commissioners (SECY-78-137) also urges the inclusion of core-melt considerations in site comparisons in the case of sites involving high population density, such as Byron and the surrounding area in which live now (or at time of proposed operation) upwards of 500,000 persons. Moreover, neither C.E. nor the NRC Staff has presented an accurate assessment of the risks posed by operation of Byron, contrary to the requirements of 10 C.F.R. § 51.20(a) and § 51.20(d). The decision to issue the Byron construction permit did not, and the presently filed analysis of C.E. and the Staff do not, consider the consequences of so-called Class 9 accidents, particularly core meltdown with breach of containment. These accidents were *deemed* to have a low probability of occurrence. The Reactor Safety Study, WASH-1400, was an attempt to demonstrate that the *actual* risk from Class 9 accidents is very low. However, the Commission has stated that it "does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accident." (NRC Statement of Risk Assessment and the Reactor Safety Study Report (WASH-1400) in Light of the Risk Assessment Review Group Report, January 18, 1979.) The withdrawal of NRC's endorsement of the Reactor Safety Study and its findings leaves no technical basis for concluding that the actual risk is low enough to justify operation of Byron. [Footnotes added.]

*League Contention 62:*

The design of Byron does not provide protection against so-called "Class 9" accidents. There is no basis for concluding that such accidents are not credible. Indeed, the staff has conceded that the accident at TMI falls within that classification. Therefore, there is no reasonable assurance that Byron can be operated without endangering the health and safety of the public. *See also* Contention 8, *supra*.

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[44] The "secret NRC study" referred to here is presumably an early report prepared by the U.S. Atomic Energy Commission entitled "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants," U.S. Atomic Energy Commission, March 1957, and often designated as WASH-740.

[45] This report of the General Accounting Office is not otherwise identified.

*DAARE/SAFE Contention 2A:*

Due to the concentration of nuclear power plants already in Northern Illinois; the Applicant's record of incidents and violations in existing plants which have emerged since the granting of a Construction License for Byron; and the credibility which must now be given to large-scale accident scenarios since TMI, Intervenors contend that the addition of Byron Station operations places an undue and unfair burden of risk from exposure to radioactive materials from accidental releases on DeKalb-Sycamore and Rockford area residents. With the addition of two more nuclear power units in operation at Byron, the potential for cumulative dose effects from discrete accident events at plants in Northern Illinois under unfavorable meteorological conditions poses an unreasonable level of risk to the health and safety of DeKalb-Sycamore and Rockford area residents.

C-2. "Severe" accidents in the context of the contentions were formerly designated as Class 9 accidents.

C-3. Litigation of these contentions is mandated by Commission regulations which require, before issuance of a license for operation at even 1 percent of design power, reasonable assurance of the absence of danger to the health and safety of the public. 10 C.F.R. 50.57(a)(3) and (6); 10 C.F.R. 50.57(c).

C-4. The documentation of information providing background for the litigation of these contentions of the possible severity of potential accidents at a nuclear power station was presented in the Final Environmental Statement (FES), prepared by the Staff as report NUREG-0848. Staff Ex. 2, Section 5.9.4, at 5-23 ff., and Appendix E. This documentation was supported by the Staff's witnesses. Hulman, *et al.*, ff. Tr. 2091, at 2.

C-5. Guidance for the preparation of this Staff document was provided by the Commission's interim policy statement on Nuclear Power Plant Accident Considerations under the National Environmental Policy Act of 1969 (NEPA), dated June 13, 1980. 45 Fed. Reg. 40,101 (1980); *see also* Hulman, *et al.*, ff. Tr. 2091, Attachment B.

C-6. Additionally to those events and accident sequences that lead to releases of some radioactive substances and which can reasonably be expected to occur, such as the design basis accidents earlier put forth by the U.S. Atomic Energy Commission in a proposed Annex to Appendix D of 10 C.F.R. Part 50, the current policy statement demands consideration of site-specific environmental impacts attributable to accident sequences that can result in reactor core melting. These are "severe" accidents. 45 Fed. Reg. 40,101 (1980). Intervenors' Proposed Finding 10 says WASH-1400 does not consider "degraded core accidents." "Degraded" is not defined, but is presumed to mean the core is not melted. WASH-1400 was a study of potential risks to the public from accidents at nuclear power stations. It was concluded early in the study that the public could be harmed only as a consequence of a melted core. Therefore "degraded-core" accidents are beyond the scope of WASH-1400.

C-7. The Applicant argued this matter through the testimony, both written and oral, of Saul Levine, a former AEC/NRC employee who, among other responsibilities, served as the project manager for the NRC activity which produced the Reactor Safety Study.

C-8. The Staff presented its case through a panel of witnesses comprised of L.G. Hulman, M.L. Wohl, Scott Newberry, and E.F. Branagan, Jr., all regular employees of the Staff with managerial responsibilities in several appropriate sectors of the Commission organization.

C-9. The Intervenors presented no witnesses of their own and made their case through cross-examination of the witnesses of the other parties.

C-10. Accordingly both the Applicant and the Staff addressed such severe accidents expected to have both a low probability of occurrence and more extensive consequences to the environment and to the plant itself. Such accidents may be characterized as involving overheated, even melted, fuel and deterioration of the capability of the reactor pressure vessel to withstand a potential and concomitant increased internal force thereby breaking still another containment boundary. Staff Ex. 2, at 5-44.

C-11. The Staff utilized the probabilistic risk assessment methodology of the Reactor Safety Study (WASH-1400) as updated. It has been amended by consideration of comments arising in peer review and from advances in knowledge which have occurred since the (1975) publication of WASH-1400.

C-12. The severe-accident analysis by the modified WASH-1400 method as applied to a number of postulated sequences of events led to probabilities of significant releases of radionuclides to the atmosphere of the order of one in $10^5$ reactor-years. The fractions of inventories, typical of operating pressurized water reactors of the Byron class in such accidents, ranged from 100 percent of the noble gases to 2 percent of the alkaline earths including the long-lived strontium-90 isotope. Applicant Ex. 2, Table 5.11, at 5-45.

C-13. The validity of the methodology propounded in the Reactor Safety Study (WASH-1400) and the confidence to be attached to results derived from that methodology have been challenged in League Contention 8, *supra*.

C-14. The Reactor Safety Study was the first comprehensive application of probabilistic risk assessment to nuclear power plants. The charter of the Study was to make quantitative predictions of the risks to the public from potential accidents at operating nuclear power plants. This was done through a detailed analysis of a pressurized water reactor and of a boiling water reactor and the extrapolation of that information to an assumed population of 100 reactors located at a "composite" site embodying significant characteristics, including population and meteorological features, of actual sites. A major result of the study was that the risks from accidents at nuclear plants will be small compared to the risks to the public arising from other events in our society. Levine, ff. Tr. 1930, at 8. In their Proposed Finding 6, the Intervenors categorically say that the Byron PWR and the WASH-1400 PWR were significantly different without elucidation of the differences and their significance. The WASH-1400 PWR is of Westinghouse design selected as typical for its generic study. In their Proposed Findings 11 and 12 the Intervenors allude to the absence of the TMI-2 event[46] from WASH-1400. One reason, of course, is timing — WASH-1400 in 1975, TMI-2 in 1979. Another is the reactor type — Westinghouse versus Babcock & Wilcox. A third reason is that TMI-2 was not a melted core.

C-15. A reading of League Contention 8, *supra,* reveals a conclusion that WASH-1400 is without value. The argument for that conclusion apparently arises from, perhaps among other sources, a peer review of the study and its product. The review was by the Independent Risk Assessment Group and its report is designated as NUREG/CR-0400, usually referred to as the Lewis Report. Hulman, *et al.,* ff. Tr. 2091, at 6 (Wohl).

C-16. The criticisms of the Safety Study voiced in the Lewis Report were not addressed to the concepts and methodology of WASH-1400. Rather they specifically included remarks on the clarity of the presentation, particularly of the mathematical formulation, an inability to establish a quantitative overall probability for core melt, and an assertion that the error bands were understated.[47] *Id.* Intervenors' Finding 14 says, in contrast to the above, that WASH-1400 was "attacked" by the Lewis Report.

C-17. During the preparation of the Reactor Safety Study, in the mid-seventies, as many as five times the then operating nuclear reactors were anticipated to be functional within a relatively short time span. It was expected that these additional plants would embody advances in

[46] The Staff and Board sense a typographical error in Proposed Finding 12 which says in part "address . . . other severe accidents such as Browns Ferry, the Browns Ferry fire . . . ." NRC Staff Reply to Rockford LWV's Proposed Findings, filed July 18, 1983, at 20.

[47] "The Executive Summary to WASH-1400 . . . does not adequately indicate the full extent of the consequences of reactor accidents; and does not sufficiently emphasize the uncertainties involved in the calculation of their probability. It has therefore limited itself to misuse in the discussion of reactor risk." NUREG/CR-0400, the Lewis Report, at ix. Disavowal of this summary by the Commission has been a cause of some confusion about the acceptance of the document as a whole.

design and would supplement the operating experience history. These two items would thereby contribute to the data base of studies like WASH-1400. For this reason the users of WASH-1400 were advised to recognize such advances. Accordingly, an artificial 5-year lifetime was arbitrarily assigned to the findings of the Study. The anticipations have, of course, not materialized. The imposed "lifetime," *per se,* was not intended to and did not invalidate the utility of WASH-1400 beyond 1980. Tr. 2071-72, 2073-74 (Levine). Intervenors' Finding 13 alleges the methodology of WASH-1400 to be invalid after October 1980.

C-18. In a constructive vein, findings of the Lewis assessment of WASH-1400 included views that

- event-tree/fault-tree methodology [basic to WASH-1400] is demonstrably sound;
- the [WASH-1400] methods provide a substantial advance over previous attempts to estimate the public risks from nuclear power plants;
- event-tree/fault-tree methodology and other aspects of the modeling have set a framework that can be used broadly to assess choices involving both technical consequences and impacts on humans;
- the event-tree/fault-tree approach with an adequate data base is the best available tool with which to quantitatively predict the probabilities of reactor accidents.

Levine, ff. Tr. 1930, at 10.

C-19. Support by the Commission of at least the basics of the Reactor Safety Study is recorded in a number of places.

> Taking due account of the reservations expressed in the Review Group Report and in its presentation to the Commission, the Commission supports the extended use of probabilistic risk assessment in regulatory decisionmaking.

NRC Statement on Risk Assessment and the Reactor Safety Study Report (WASH-1400) in Light of the Risk Assessment Review Group Report at 4 (January 18, 1979).[48] Additional Commission statements on the use of probabilistic risk analyses are noted by the Applicant's witness. Levine, ff. Tr. 1930, at 12-13. These statements are in contrast to Intervenors' Proposed Findings 15 and 16.

C-20. In a directive the Secretary of the Commission sent to the Executive Director for Operations on January 18, 1979, the Commission

[48] This statement was issued under the NRC Office of Public Affairs document 79-19. It apparently did not appear in the *Federal Register.*

stated that "[q]uantitative risk assessment techniques and results can be used in the licensing process if proper consideration is given to the results of the Review Group [Lewis Report] . . . ." At the same time the Staff was instructed to apply quantitative risk assessment techniques to estimate the relative importance of accident sequences where sufficient similarity exists to provide an adequate base; quantitative estimates in the RSS should not be used as the principal basis for any regulatory decision (such estimates can be used for relative comparison of alternate designs); the RSS consequence model shall not be used as the basis for licensing decisions on individual nuclear plants until significant refinements and tests are accomplished. These positions were in effect in late 1982. Hulman, *et al.*, ff. Tr. 2091, Attachment C (letter dated December 27, 1982 to Udall from Ahearne) at 2.

C-21. Two additional appraisals of the probabilities of consequences of reactor accidents were included in the record. One of these is reported in "Precursors to Potential Severe Core Damage Accidents: 1969-1971 A Status Report" (NUREG/CR-2497) (1982). This investigation is referred to as the "Precursor Study." It takes as points of departure summaries of nearly 20,000 licensing event reports (LERs) of occurrences at light water power reactors. The study was instigated by a recommendation appearing in the Lewis Report. After screening, 169 events were judged to be accident sequence precursors. Of these, fifty-two were selected as having a potential probability for severe core damage equal to or greater than 0.001 assuming the precursor event occurred in the manner it did. These probabilities lead to a frequency of severe core damage per reactor-year for the decade investigated. The results ranged between 0.0017 and 0.0045 per reactor-year. It is to be noted that under discussion is an analysis of events which actually occurred, including TMI-2, the fire at Browns Ferry, and the loss of instrumentation at Rancho Seco.[49] The analysis is sensitive to variations in operating requirements among the various plants. Finally, the product is an estimate of events producing core damage which may not lead to significant releases of radiation, and therefore are not included in the Reactor Safety Study, *supra,* since it evaluated events leading to reactor core melting, in accord with its charter to evaluate risks to the public. Accordingly the agreements between the RSS and the Precursor Study results for typical PWRs are not internally inconsistent. Hulman, *et al.,* ff. Tr. 2091, at 8, 9; Tr. 2282 (Hulman); Tr. 2277-81 (Levine).

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[49] These three occurrences account for 82 percent of the estimate of severe accident frequency in the Precursor Study. They were not considered in the Reactor Safety Study. Hulman, *et al.*, ff. Tr. 2091, at 9.

C-22. The Precursor Report fed generic data into a generic event-tree taking account neither of the particular plants where the infrequent precursor occurred nor of the specific failure probabilities that would be applicable to those particular plants. Accordingly this generic approach will almost certainly predict failure probabilities that are too high. Levine, ff. Tr. 1930, at 25; Tr. 2022, 2023 (Levine).

C-23. The second appraisal was by the Institute of Nuclear Power Operations (INPO). In an analysis of the Precursor Report appearing as "Review of NRC Report: Precursors to Potential Severe Core Damage Accidents," INPO 80-025 (1982), it is concluded that when the actual plant configurations, where the precursors that were analyzed occurred, are taken into account, the average (precursor) probability of core damage is reduced by more than an order of magnitude. Applicant's witness considers the INPO methodology superior to that of the Precursor Report. Levine, ff. Tr. 1930, at 25; Tr. 2023, 2028-32 (Levine).

C-24. The Staff's analysis of potentially severe accidents at Byron did not include those attributable to external and man-made actions.[50] Two reasons for this alleged omission are: design requirements to counter natural phenomena appear in 10 C.F.R. 50, Appendix A, and, in analyses, appear in design basis accident considerations. Countermeasures for sabotage are in 10 C.F.R. Part 73. Since the radiological consequences of such events will not be dissimilar from those of other "accidents," they have, in effect, been included. Further, their frequency probability is within the uncertainty bounds of the internally generated occurrences. The second reason for the absence of this type of nature-instigated events, more severe than those within the design basis, is the paucity of describing data which places them outside the state-of-the-art of probabilistic risk assessment. Staff Ex. 2, at 5-45, 5-46.

C-25. Uncertainties assignable to the probabilities of severe accidents was a subject of discussion during the hearing. *See,* for example, Levine, ff. Tr. 6956; Tr. 6957-96 (Levine); Hulman, *et al.*, ff. Tr. 2091, at 4; Tr. 2240-57 (Hulman).

C-26. Recent analyses of both the Zion and Indian Point reactors[51] included consideration of probabilities of the occurrence of severe accidents arising from both internal and external events including sabotage. In both analyses, risks from sabotage, as stated above, were considered beyond the capabilities of probabilistic assessment, and effects of successful acts would probably lie within the error bounds of the estimates

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[50] External and man-made events, in this context, include tornadoes, fires, earthquakes, sabotage, explosions, and aircraft crashes. Hulman, *et al.*, ff. Tr. 2091, at 3.

[51] Probabilistic risk analyses have been made for the Zion and Indian Point reactors. Tr. 6970 (Levine).

for internally generated accidents. Evaluation of the consequences of other externally produced events was guided by the Zion/Indian Point studies. The Staff (Hulman, *et al.*, ff. Tr. 2091, at 4) modified the Final Environmental Statement (Staff Ex. 2) by including the opinion that accident risks at Byron from both internal and external events, excluding sabotage, would be no more than 100 times greater than the risks from internal events presented in the FES. Tr. 2247-55 (Hulman). Intervenors, in Proposed Findings 20-21, claim incorrectly that no account was taken of externally originated events.

C-27. The Applicant took strong exception to this upper bound on an uncertainty factor of 100 in the direction of increased risk arising from the absence of external accident precursors in the risk analysis presented in the FES. Staff Ex. 2, Section 5.9.4.5(2), at 5-44; Tr. 2256-58 (Hulman); Levine, ff. Tr. 6956, at 1.

C-28. The probability risk assessments for the Zion and the Indian Point plants gave 10 to 1 and 30 to 1, respectively, as the ratios of external- to internal-initiated risks. Although any true relation between these ratios, respectively, and the proper value for the Byron site is coincidental, the Staff, with solely those ratios as guidance, arrived, apparently arbitrarily, at the 100 multiplier as its best estimate of the bound on the ratios for Byron. Hulman, *et al.*, ff. Tr. 2091, at 4 (Hulman, Wohl).

C-29. The propriety of applying a probability analysis determined for one site as the characterization of another has been challenged by Mr. Levine. Ff. Tr. 6956, at 3.

C-30. Additionally the treatment of severe accident probability in the Staff's Final Environmental Statement (Staff Ex. 2) is, in general, not site-specific and is sufficiently conservative to preclude the necessity of applying additional uncertainties. Levine, ff. Tr. 6956, at 4; Tr. 6991, 6992 (Levine). Particular items of conservatism built into the WASH-1400 method are: deposition of some radionuclides in the primary coolant; retention of some iodine as cesium iodide; capture of tellurium by the fuel cladding and other metals; selection of the accident sequence, Event V, which predicts the greater release of radioactivity; and a higher-than-now-expected probability of the rupture of the containment vessel over a shorter time interval. Levine, ff. Tr. 6956, at 4-7.

C-31. Applicant's witness stated that, in his judgment, no factor of uncertainty should be imposed upon the Staff's severe accident probability, $5 \times 10^{-5}$ per reactor-year, presented in its FES. Tr. 6992 (Levine); Hulman, *et al.*, ff. Tr. 2091, at 9 (Newberry).

C-32. These probabilities of accident occurrences and concomitant radionuclide emission rates together with their predicted health effects and the population distribution about the Byron site were consolidated into estimated probability distributions of specific radiation doses to individuals and, collectively, to the population. Calculated also were the probabilities of early fatalities and of latent cancer deaths.[52] A more comprehensive index of the potentials of reactor accidents was developed by the Staff as average values of environmental risks. This quantity is the sum of the products of probabilities of occurrence and their respective consequences. These results for a number of scenarios, expressed as events per reactor-year, appear in the FES. Staff Ex. 2, Table 5.13, at 5-60. Whereas the absolute values of these indices may be challenged on several grounds, including the subjective character of risk evaluation, they are useful as bases for comparisons with risks associated with normal reactor operation and with more commonly experienced events. For example, the severe accident exposure doses are comparable to the exposures expected from normal operation of a Byron reactor throughout its life.[53] Hulman, *et al.*, ff. Tr. 2091, Summary Item 2; Tr. 2291 (Hulman). The early fatality accident risk is less than that, for a comparatively sized population, from auto accidents by about four orders of magnitude; it is less than fatalities from mundane actions, such as burns or drownings, by three orders of magnitude. Staff Ex. 2, at 5-60.

C-33. The projected effects of the events are expressed more quantitatively in terms of personnel exposures and monetary costs of area rehabilitation in Table 5.12 of Staff Ex. 2, at 5-54. As an exemplificative situation, choose an impact having a probability of $5 \times 10^{-6}$ per reactor-year.[54] The predicted consequences from the accompanying atmospheric release of radionuclides are: 7000 individuals exposed to more than 25 but less than 200 rem; population exposures, expressed in million person-rem, of 1.5 within 50 miles and 12 in the entire affected area; the corresponding latent cancers are 180 and 840. The cost of lost foodstuff and of decontamination and/or property replacement is set at 430 million dollars. No early fatalities and exposures greater than 200 rem are expected. These effects are to be contrasted to an annual exposure of the public to 60 person-rem from normal operation of the Byron Station and to 130,000 person-rem from natural background radiation. Staff Ex. 2, at 5-50.

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[52] The probabilities are presented for ≥200 rem whole-body (likely requiring hospitalization), ≥25 rem whole-body (clinically detectable), and ≥300 rem thyroid (guideline for siting in 10 C.F.R. Part 100).

[53] The radiation exposure from any design basis accident is also comparable, within an order of magnitude, to the exposure expected from normal operation throughout the projected life of the plant. Staff Ex. 2, at 5-43; Tr. 2296 (Branagan); Tr. 2299 (Hulman, Wohl).

[54] This frequency is characteristic of the accident sequences selected for analysis in the Reactor Safety Study, as modified and updated, of pressurized water reactors. These several sequences, denoted as V, TMBL', PWR-3, are expected to dominate PWR risks and are detailed in Appendix E of Staff Ex. 2.

C-34. The release to the atmosphere of sufficient radionuclides to constitute a risk to the public would be the consequence of an accident of sufficient severity to melt the reactor core and violate the integrity of the containment. Tr. 2037 (Levine). The probability of a core-melt accident at a reactor, and concomitant expected early fatality is of the order of $10^{-4}$ per reactor-year. The other important potential health effect is latent cancer deaths. Typical analytical results show the probability of radiation cancer fatalities induced by severe reactor accidents is negligible compared to cancer from other causes. Levine, ff. Tr. 1930, at 22-24. The probability of two (or more) accidents of such severity occurring within a short time is correspondingly smaller. Hulman, *et al.*, ff. Tr. 2091, at 21 (Wohl).

C-35. Additionally to the matter of simultaneity, *supra,* the effect of distance between the potential sources and affected populations is a consideration. Probability risk analyses have shown that probability of early fatalities is small at distances from a source in excess of 15 miles. Considering, respectively, the potential radiation sources — Byron, Zion, Dresden, LaSalle — and the centers of population — Rockford, DeKalb, Sycamore — no center is within two of the above 15-mile-radius areas circumscribed about its nearest sources. Levine, ff. Tr. 1930, at 23, 24. As a rule the Staff does not judge the safety of one unit on the possibility that another unit may be built nearby. Tr. 2202 (Hulman).

### *Severe Accident Conclusions*

C-36. From a review of the evidentiary record as summarized in the foregoing findings, the Board concludes that the Applicant and Staff have adequately addressed and evaluated the anticipated probability and consequences of severe accidents at the Byron Station which have the potential of endangering the health and safety of the public. The Board finds that the Reactor Safety Study (Rasmussen Report, WASH-1400), *per se,* is admittedly incomplete and somewhat outdated. It has been revised to reflect both the results of recent investigations and of the peer review. With admitted shortcomings, it presents, nonetheless, the best and most applicable methodology for probabilistic risk analysis presently available, recognizing the absence of any operating experience leading to melting of the reactor fuel. Consideration of the geographical distribution of nuclear-fueled electric generating plants in northern Illinois and the spatial distribution of radiation exposure which could arise from a severe nuclear accident at Byron, the Board believes that cumulative accident effects for the public are minimal and acceptable.

C-37. Additionally the Board finds that the Applicant and the Staff have adequately reviewed and analyzed the severe accident potential in accord with an interim policy statement issued June 13, 1980 and as required by the general conditions noted in 10 C.F.R. 50.57 and 51.20. These findings are contrary to the contentions relating to severe accidents posed by the Intervenors.

## D. League Contention 1A — Quality Assurance and Quality Control

D-1. League Contention 1A, as modified by the Board, alleges:

> 1A. Intervenor contends that Edison does not have the ability or the willingness to comply with 10 C.F.R. Part 50, Appendix B, to maintain a quality assurance and quality control program, and to observe on a continuing and adequate basis the applicable quality control and quality assurance criteria and plans adopted pursuant thereto, as is evidenced by Edison's and its architect-engineers' and its contractors' past history of noncompliance at all Edison plants (whether or not now operating). In addition, Applicant's quality assurance program does not require sufficient independence of the quality assurance functions from other functions within the Company.

D-2. As litigated, the QA/QC contention asserted that the Applicant has neither the ability nor willingness to implement and maintain an adequate quality assurance program and that the quality assurance program is insufficiently independent from other company functions. Evidence was received with respect to both construction and operational quality assurance.

### *1. Applicable Law*

D-3. According to 10 C.F.R. 50.34(a)(7) the preliminary safety analysis report must contain "[a] description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility."

D-4. Appendix B to Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, sets forth the requirements for quality assurance programs for nuclear power plants. The description of the QA program must discuss how the requirements of Appendix B will be satisfied.

D-5. Appendix B lists eighteen separate quality assurance criteria which must be met in the construction and testing of a nuclear power plant. Of particular relevance to the QA contention are those that follow.

D-6. Criterion I relates to the quality assurance organization and requires a QA program for both the applicant and its contractors. The applicant may delegate QA work but remains responsible for it. QA responsibilities and authorities must be clear and in writing. Not only must the QA functions be established and executed, but they must be verified, *e.g.*, by checking, inspecting and auditing. The QA function must be independent, since Criterion I requires:

> The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided.

D-7. Criterion II requires that the QA program shall be established at the earliest practicable time and shall be documented by written policies, procedures or instructions and the program must extend for the entire plant life. This program must identify the structures, systems and components that it is to cover with the respective cognizant organization.

D-8. Criterion III requires that the QA program shall adopt measures for design control to verify and check the adequacy of design control and subsequent design changes. Criterion IV assures that regulatory requirements necessary to adequate quality are included or referenced in the documents for procurement of material, equipment and services.

D-9. Criterion V provides that activities relating to quality shall be controlled by documents, which documents are themselves to be controlled according to Criterion VI. Criterion X requires inspections which must be made by persons other than those who did the work being inspected and requires that there shall be testing which must be documented in accordance with Criterion XI. Materials and equipment must be protected (Criterion XIII). Nonconforming materials must be controlled (XV), and corrective measures for conditions adverse to quality provided for (XVI). Quality assurance activities must be provable by records (XVII), and all aspects of the QA program must be verified by an organized auditing program (XVIII).

## 2. *Commonwealth Edison Company Policies, Experience, and Corporate Structure*

D-10. The Applicant brought to the hearing those officials who are most able to explain and defend its position with respect to its commitment to safety and how its corporate and plant organizational structure has been designed to carry out that commitment. We heard from Cordell Reed, Vice President of Nuclear Operations, Robert Querio, Byron Station Superintendent, Walter Shewski, the Corporate Manager of Quality Assurance, and Michael Stanish, who is serving as Construction Quality Assurance Superintendent at Byron.

D-11. Region III of the Office of Inspection and Enforcement presented a panel of five officials who are well informed concerning Applicant's corporate organization and construction quality assurance experience. In particular, the Staff presented the testimony of Messrs. D.W. Hayes, James E. Konklin, and Cordell E. Williams, who are Region III section chiefs with responsibilities for Byron, and Mr. William Forney, who served as the Chief Resident Inspector at Byron for much of the relevant period. This panel also testified extensively on all other aspects of the quality assurance contentions, as we note in the following sections, and they stated their perceptions of Applicant's commitment to nuclear safety. We also heard from NRC's John Spraul, a quality assurance engineer.

D-12. Intervenors had no witnesses directly on the issue of corporate structure and organization, and filed no proposed findings directly on the issue. Given the strong structural implications of the contention, this is somewhat surprising, but apparently Intervenors do not dispute the objective facts concerning Applicant's avowed commitment to safety (*e.g.*, membership in nuclear safety groups) and the corporate structure pertaining to quality assurance. Intervenors, however, dispute the conclusions to be drawn from the testimony on these sub-issues. Accordingly, we have relied very heavily on Applicant's proposed factual findings and the Staff's similar proposals, frequently adopting them verbatim on the sub-issues pertaining to corporate structure and avowed policy.

*Commitment to Safety*

D-13. The Vice President for Nuclear Operations, Mr. Reed, represented to the Board that Applicant has a strong commitment to safety. Reed, ff. Tr. 2594, *passim*. The Office of Inspection and Enforcement (I&E) and the Byron NRC Senior Resident Inspector also believe that there is a real corporate commitment to safety. Region III Testimony, ff. Tr. 3586, at 10 (Hayes), at 14 (Forney). The tenor of the Region III officials throughout the hearing has been that despite noncompliances, Applicant has a corporate attitude consistent with a commitment to safety. *E.g.*, Tr. 3929-30 (Forney, Hayes, Williams, Yin, Konklin).

D-14. Applicant has taken an active role in industry groups formed to address new and ongoing safety concerns, including support for the activities of the Atomic Industrial Forum and the Edison Electric Institute. Applicant participates in *ad hoc* groups formed to address specific technical safety issues, and contributes money and the advice of experienced personnel to safety-related research conducted by the Electric Power Research Institute. Reed, ff. Tr. 2594, at 4.

D-15. Applicant has been instrumental in the formation of industry-wide groups which undertake to enhance nuclear safety, such as the Nuclear Safety Analysis Center (NSAC) and the Institute for Nuclear Power Operation (INPO). *Id.* Applicant's nuclear plants, both those operating and under construction, have been evaluated by INPO a number of times, and Applicant fully subscribes to INPO principles. Tr. 2605 (Reed).

D-16. While Mr. Reed's testimony expressing the Applicant's commitment to safety is appropriate, it is also prudent and self-serving in an operating license proceeding. We found the Staff's expressed attitude concerning the Applicant's commitment to safety to be reassuring, but summary. Applicant's participation in the nuclear safety groups such as INPO is an objective indication of a corporate commitment. However, any conclusion concerning the Applicant's attitude should rest upon the entire record concerning what Applicant actually has done about safety, and in particular, quality assurance.

*Applicant's Noncompliance Record*

D-17. Prior to 1980 the Commission designated three categories of noncompliances with its regulations and imposed fines accordingly. In order of decreasing severity, the categories were "violation," "infraction," and "deficiency." The maximum fine was $5,000 per violation with the total for any 30-day period limited to $25,000. During 1980 and 1981, the Commission used six categories of noncompliance and imposed fines based on the severity level. The categories in order of decreasing severity were designated Levels I through VI. The maximum fines varied with each severity level, the largest being $100,000 per day. Since 1982, the Commission has used five categories of noncompliance and continues to impose fines based on the levels of noncompliance. The categories decrease in severity from Level I through Level V. The largest fine remains at $100,000 per day. Region III Testimony, ff. Tr. 3586, at 7-8; Tr. 3629 (Forney). There is often an element of judgment involved in determining into what severity level an item of noncompliance falls. Tr. 3601 (Forney, Hayes).

*Civil Penalties*

D-18. Since 1974 Applicant has been fined a total of $313,000 by the NRC in connection with violations in the operation of its seven reactors and the construction of its six reactors. None were in connection with the Byron Station. Del George, ff. Tr. 2344, at 12; Tr. 2346, 2352 (Del George). We identify some of the more noteworthy events below.

D-19. In October 1975, the Commission fined Applicant $25,000 for violations at its Quad-Cities Station for an error in control rod withdrawal which caused fuel damage and for deficiencies in implementing the new station security plan. Applicant had been fined the previous year for failure to implement its security plan at its Dresden Station. Del George, ff. Tr. 2344, Exhibit 2, at 1, 2.

D-20. In May 1976, Applicant was fined $13,000 for excessive radiation exposure to an employee. The exposure had no apparent effect on his health. *Id.* at 3.

D-21. In September 1977, Applicant was fined $21,000 for inadvertently draining the pressurizer at its Zion Station while the reactor was shut down. In October 1980, the Commission imposed a compromise fine of $18,000 for inattentiveness of operators at Dresden.

D-22. In March 1981, Applicant was fined $80,000 for excess radiation exposure to two contractor employees at Dresden Station due to the failure of station personnel to survey the working environment. The exposure had no apparent effect on their health. *Id.* at 5. In July 1982, Applicant was fined $100,000 for an occurrence at Zion Station involving excessive radiation exposure to an employee who entered a high-radiation area without taking proper precautions but with no apparent effect on his health. *Id.*

D-23. Since 1980, Applicant has been denied access to low-level waste burial sites on eight occasions. On three of these occasions, the Commission also assessed a fine. Del George, ff. Tr. 2344, at 22.

D-24. The Applicant represents to the Board that there was no risk to the health and safety of the general public involved in each of the foregoing situations. Applicant also recites the corrective actions taken to prevent recurrences of the penalized violations. Del George, ff. Tr. 2344, Exhibit 2, *passim.* Applicant's claim that the public safety was never threatened by any of the occurrences was never disputed by the parties, but the Board is not able to find that such is the case because the testimony to that effect was summary, unexplained, and not self-evident or sometimes very judgmental. However, we believe that the remedial steps thoroughly described in Mr. Del George's testimony were rational and responsive. *Id.*

D-25. Region III compared nuclear units owned by Applicant with other reactors in respect to the amount of the fines imposed in two periods, 1974-78 and 1979-82. For the earlier period, Applicant's average for fines was above the national and Region III average. For the second period, 1979-82, Applicant's average was below Region III average and substantially below the national average. Region III Testimony, ff. Tr. 3586, at 12 (Forney) and Attachment B-1.

D-26. Intervenors protest that the Staff's analysis does not include \$220,000 in civil penalties proposed in three actions by the Staff against the Applicant in February 1983. Tr. 2346 (Del George). In 1983 Applicant paid a \$20,000 penalty for a violation at Dresden. Del George, ff. Tr. 2344, at 13, and Exhibit 2, at 6. In February 1983, a notice of proposed civil penalty of \$100,000 was issued, citing a steam generator bolting problem at Braidwood. The NRC believed that corrective action at Braidwood had not been taken in a timely manner. A similar problem had been identified at Byron earlier. Applicant was also considered more than 2 years late in reporting the Braidwood problem to the Commission under the provisions of 10 C.F.R. 50.55(e). Intervenors Ex. 6. Applicant disagrees with the NRC Staff's assessment regarding the timeliness of corrective action at Braidwood, and, on this basis, it intends to seek mitigation of the proposed fine. Del George, ff. Tr. 2344, at 13; Tr. 2462 (Del George).

D-27. The validity and significance of the Braidwood proposed civil penalty is still in dispute. Tr. 3927 (Hayes). Region III believes that the problem should never have surfaced at Braidwood because the same plans were used at Byron 18 months earlier and changes accommodating the bolting problem at Byron should have applied to Braidwood.[55] Region III acknowledges that the situations at the two plants were not identical, however. Tr. 3641-43 (Hayes, Forney).

D-28. Also in 1983 the Staff proposed \$100,000 in penalties for the alleged continued use of valve guides of unacceptable quality in electromatic relief valves at Dresden and Quad-Cities. Intervenors Ex. 7. Applicant has not yet decided how to respond to this action.

D-29. The \$220,000 fines assessed or proposed against Applicant in 1983 alone exceed the total fines levied against Applicant in the 1979-82 period, the later period used in Staff's comparison table. If the year 1983 were included, it is apparent that Applicant would not compare so well

[55] Intervenors Ex. 6, Region III's Notice of Violation and Proposed Imposition of Civil Penalties in the Braidwood generator-bolting case, refers to a "breakdown in the quality assurance program" at Braidwood. This would be a serious conclusion, but the notice was sent out in error and was subsequently replaced by a notice, not in evidence, softening the charge. It is now regarded as a quality assurance "problem" or "missed elements" in the quality assurance program. Tr. 3636-38 (Hayes, Forney).

with the national and Region III averages. Region III Testimony, ff. Tr. 3586, Attachment B-1. However, the book is not yet closed on the two $100,000 proposed penalties in 1983. The merits of neither the Braidwood nor the Dresden and Quad-Cities proposed fines were fully explored in our proceeding. Moreover, the ranking Region III witness respecting the Byron proceeding testified that he took into account the $100,000 proposed Braidwood penalty in his favorable assessment of Applicant's quality assurance proposal. Region III Testimony, ff. Tr. 3586, at 10 (Hayes); Tr. 3837 (Hayes). The Byron Senior Resident Inspector, Mr. Forney, who prepared the comparative civil penalty analysis testified that the proposed 1983 penalties would not change his mind about his favorable evaluation of Applicant's corporate attitude and policy. Tr. 3861, 3927 (Forney).

D-30. For no other reason than that the proposed 1983 fines haven't been imposed yet, these amounts cannot fairly be added to previous amounts in comparing Applicant's civil penalty history. In addition, 1983 information concerning other utilities is not in evidence. The amounts of civil penalties imposed by the NRC have increased in severity in the past several years. Tr. 3927 (Forney). Of greater importance, however, is the lack of grounds in the record to make any comparison among utilities and reactors based on the amounts of civil penalties. The author of Region III's comparative civil penalty analysis would not accept the proposed 1983 civil penalties as a reflection of corporate attitude and policy because, he stated, the attributes of policy and attitude are not reflected in fines. Rather, he states, they are reflected in Applicant's topical report which commits strongly to the ASME Codes, and to the regulations, and to the company's implementing quality assurance manual. Tr. 3861 (Forney). By the same reasoning, the Board does not accept in either direction, the Staff's comparative analysis showing the Applicant has a better-than-average civil penalty experience in the most recent comparison period or worse-than-average in the earlier period.

D-31. Another episode involving Applicant's other plants received widespread public and very thorough NRC attention but no monetary penalty was imposed.

D-32. In April 1977 former Pinkerton Security Agency employees made allegations concerning the security system and plant operations at the Quad-Cities Station. The NRC charged Applicant with eleven infractions and five deficiencies. A civil penalty was considered but deferred.

D-33. Applicant's corporation and two of its employees were indicted for false statements but they were acquitted at trial. The NRC then decided not to impose civil penalties, reciting Applicant's prompt corrective actions and the focus on security arrangements caused by the criminal prosecution. Del George, ff. Tr. 2344, at 19-21.

D-34. The Byron Station has been designed with industrial security as one of its design criteria. This and other measures will eliminate many of the difficulties Applicant experienced developing effective security programs at its operating stations. *Id.* at 21.

*Noncompliances Not Involving Penalties*

D-35. Generally, the number of noncompliances issued by the NRC Office of Inspection and Enforcement has increased in recent years for all nuclear sites under construction. One reason for the increase has been a greater emphasis on construction quality assurance by the Commission. Tr. 3591-92 (Hayes). Another reason is the relatively recent implementation of an NRC resident inspector program. Byron received its first resident inspector on October 5, 1981. Tr. 3592 (Hayes, Forney). There are also reasons specific to Byron why the number of noncompliances has recently increased at the Byron site. The number of inspector-hours spent on site has increased from 361 in 1978 to, conservatively, 2547 in 1982. Tr. 3601-03, 3808-10 (Hayes, Forney). Closer attention by Region III has been paid to Byron as it approaches its fuel-load date. Tr. 3605-07 (Forney, Williams). In recent years the number of noncompliances documented at the Byron site has increased. In 1978, a total of three items of noncompliance were identified at Byron by the NRC Staff: two infractions and one deficiency. Tr. 3602 (Forney). In 1982, thirty items of noncompliance were documented. Tr. 3604 (Hayes); Region III Testimony, ff. Tr. 3586, Attachment B-2.

D-36. For the period 1976-82, the average number of noncompliances at the Applicant's facilities compares favorably with other Region III plants. Noncompliances at Byron Unit 1 are approximately three times the Region III average, but according to Region III, the number of noncompliances is not indicative of a systematic failure because a large number relate to inspection of preoperational testing activities which primarily occurred in 1982. Region III Testimony, ff. Tr. 3586, at 12.

D-37. The Systematic Assessment of Licensee Performance (SALP) program is an integrated NRC Staff effort to collect available observations on an annual basis and evaluate licensee performance based on those observations. *Id.* at 13. The Applicant's operating plants and construction sites were rated as average as compared to those of other licensees both in Region III and nationwide in the SALP-1 rating period, July 1979 through June 1980. *Id.* at 13 and Attachment C. In SALP-2 the Applicant's performance is in the average range of Region III sites. SALP-2 ratings were not intended as a means to compare utility performance, however. *Id.* at 13 and Attachment D.

D-38. In the spring of 1982 the NRC Staff conducted its second very comprehensive inspection of the Byron Station by a special Construction Assessment Team. The inspection was performed by six inspectors and one supervisor, and involved 662 inspection hours on site evaluating Applicant's quality assurance program, compliance and corrective action history, corrective action system, design control, material traceability, electrical work activities, in-process inspections, weld rod control, and quality control inspector effectiveness. The inspection consisted of selective examination of procedures and representative records, observations, and interviews with personnel. Region III Testimony, ff. Tr. 3586, at 28, 29 (Forney).

D-39. The special team identified four Level IV noncompliances (more than minor significance) and five Level V noncompliances (minor significance). There were also ten unresolved or open items. Five of the noncompliance items remained open and five of the unresolved items were open at the time of the hearing. *Id.* The inspection report, 82-05 (Applicant Ex. 8), played a major role in the quality assurance phase of the proceeding.

D-40. The Region III panel testified that a simple tabulation of noncompliances without a great deal of additional explanation and information is essentially meaningless. The number of noncompliances serves for management information purposes, but must be considered in the context of the SALP evaluations and other controlling factors mentioned by the panel, such as the civil penalties, the severity levels of the noncompliances, the amount of activity at a particular site, the phase of the activity (*e.g.,* construction approaching fuel loading), the number of inspector-hours, the age of the plant, and the utility's response to the citations. Tr. 3609-15 (Hayes, Forney, Yin, Williams).

D-41. Mr. Forney, the former Senior Resident Inspector at Byron, explained that items of noncompliance must be individually analyzed and that:

> In assessing the significance of an item of noncompliance, and evaluating the adequacy of proposed corrective actions, consideration is given as to whether or not the item is: (1) a programmatic weakness rendering compliance indeterminate; (2) a programmatic weakness requiring evaluation to determine the extent of

compliance; (3) occurrences indicative of either (1) or (2); or (4) isolated occurrences.

Region III Testimony, ff. Tr. 3586, at 9-10.

D-42. Region III recognizes Applicant's noncompliance history, but believes that Applicant's quality assurance program assured timely effective corrective action. Based upon its inspection program, Region III believes that there is reasonable assurance that the Byron plant has been constructed in accordance with Commission requirements and can be operated safely. However, the Region also noted that some activities regarding the engineering and construction quality of certain as-built configurations have not been fully resolved, but predicted that those issues will be resolved. Region III Testimony, ff. Tr. 3586, at 10.

D-43. It became increasingly apparent to the Board throughout the hearing that simply counting civil penalty dollars and items of noncompliance, reading comparison tables and pondering the significance of severity levels of noncompliances, and their labels, could not resolve the quality assurance contention. We respect the Staff's use of this information as one of their management tools. We rely, in part, on the Staff's expert explanation of the significance of Applicant's very large noncompliance history as well as on the Applicant's version. But more important, and in larger part, the Board must also rely on its own item-by-item consideration, frequently prompted by Intervenors, of the many noncompliance items within the context of the construction activities at Byron. Most of these noncompliance items were not self-explanatory. For example, a single Level IV noncompliance with respect to the training and qualification of quality assurance and quality control personnel escaped our attention initially and resulted in the reopened hearing. That aspect of the hearing has had a major impact on the Board's final conclusion in this proceeding. In our findings and conclusions below, the Board reviews many of the items of noncompliance identified by the NRC Staff in discussing the performance of individual contractors at Byron and Applicant's oversight of its contractors.

*Corporate Organization; Offsite Organization*

D-44. In 1979 Applicant engaged a panel of Chicago scientists and business leaders to evaluate the effectiveness of its nuclear operations. As a result of recommendations made by this panel, Applicant reassigned responsibility for the operation and maintenance of its nuclear facilities to a single corporate vice president. The purpose of this change was to allow Applicant to better focus its efforts on safety and on the overall quality of nuclear operations, according to Louis Del George, a Commonwealth Edison licensing official. Del George, ff. Tr. 2344, at 5.

D-45. Reporting to the Vice President of Nuclear Operations is the Division Vice President of Nuclear Stations. Three functional managers report to the Division Vice President of Nuclear Stations: one for operations, one for maintenance and one for technical services. This corporate organization parallels the organization of the nuclear stations, providing corporate direction and company-wide standardization of practices and procedures at the stations. This is one way by which Applicant attempts to better utilize experience at each of its facilities to improve its operations at all of its facilities. *Id.* at 5-6.

D-46. Applicant's nuclear operations are reviewed by two independent organizations within the company: the Nuclear Safety Department and the Quality Assurance Department. *Id.* at 8. The individuals comprising the Nuclear Safety Department are very experienced senior employees. This group reviews deviation reports, licensing event reports and station operation to determine whether any long-term trends adverse to safety are occurring at any nuclear plant. This group also has the authority to perform an independent design review function in which it decides, apart from compliance with regulatory requirements, the adequacy of design of various plant structures, systems and components. *Id.* at 10.

D-47. The head of the Nuclear Safety Department reports directly to Applicant's Chairman and President, and on a day-to-day basis works with the Vice President of Nuclear Operations. A four-person onsite team from the Nuclear Safety Department will be assigned to the Byron Station when it is placed in operation. *Id.*

D-48. Applicant's quality assurance program is managed at the corporate level by the Manager of Quality Assurance who reports directly to the Applicant's Vice Chairman and is therefore separate from and independent of cost and scheduling constraints and responsibilities on the production side. Shewski, ff. Tr. 2364, at 7-8; Tr. 2580 (Shewski).

D-49. Applicant's witness Del George believes that Applicant's Quality Assurance Department audits more aspects of Applicant's operation than is the case for any other nuclear utility. Del George, ff. Tr. 2344, at 8-9.

D-50. The American Society of Mechanical Engineers (ASME) is an independent organization which monitors and evaluates Applicant's nuclear operations. ASME is the primary code-setting body for nuclear vessels, piping systems and concrete containment. Each of Applicant's nuclear operating units has an N-Stamp granted by ASME, which is required for Applicant to perform work on items subject to the ASME

Code. Applicant is one of only a handful of utilities which have obtained an N-Stamp and are technically qualified to perform their own ASME Code-related work. Applicant has obtained an ASME N-Stamp for the Byron Station. *Id.* at 11.

D-51. Intervenors, however, demur to Applicant's reference to ASME and its N-Stamp, asserting that neither the qualifications nor the significance of that designation has been proffered. Reply at 5. The referenced ASME codes are included in the Commission's own regulations, 10 C.F.R. 50.55a, where it is noted that the ASME N-symbol exceeds Commission requirements.

D-52. The Region III panel testified that Applicant's construction quality assurance program provides sufficient independence of function from other departments. In addition, on the basis of preoperational testing done to date, the Staff testified that Applicant's operational quality assurance program has demonstrated sufficient organizational freedom and independence. Region III Testimony, ff. Tr. 3586, at 15 (Forney).

D-53. The Board finds that Applicant's offsite corporate organization, as a structure, is logically designed and is adequate to implement Applicant's corporate commitment to safety and compliance with Commission regulations. The corporate-level structure is well designed to provide for sufficient independence of the quality assurance function from cost and production considerations.

*Byron Station Organization*

D-54. Byron Station organization, when fully operational, and after all start-up tests, will consist of approximately 470 employees assigned to the Byron Station to operate and maintain the plant. During initial start-up there will be additional personnel. Currently, approximately 450 persons are assigned to the Station, involved both in preparation of the plant for operation and the performance of various preoperational testing and checks. Querio, ff. Tr. 2714, at 4; Tr. 2718 (Querio).

D-55. The Byron Station Superintendent fulfills the position of plant manager as described in American National Standards Institute (ANSI) standard N18.1-1971. He is responsible for the direct management of the Station, including the planning, coordination and direction of the operation, maintenance, refueling, and technical activities. He is also responsible for the final approval of all Station procedures and reports. Querio, ff. Tr. 2714, at 5.

D-56. Intervenors level an *ad hominem* attack on Byron Station Superintendent Querio, urging the Board to find that confidence in Mr. Querio is misplaced. Intervenors' Response at 6. Intervenors incorrectly and unfairly say that Mr. Querio testified to the effect that he is "so certain that there are no flaws in either the past construction, or the future operation, of Byron that he would not consider a hypothetical question involving a possible serious accident," citing Tr. 2746-47. *Id.* at 6. We can find no support for this attribution. Actually Mr. Querio testified that the design of the plant can accommodate hidden construction problems but that he does not claim that the plant is so safe that it is impossible to have an accident. Tr. 2740.

D-57. When Byron becomes operational, its staff will be organized into four main functional groups: the operating group, the maintenance group, the administrative and support services group, and the personnel administration group. There are three assistant superintendents and a personnel administrator in charge of the four functional areas. They report directly to the Station Superintendent. Querio, ff. Tr. 2714, at 5.

*Byron Quality Control Group*

D-58. Within the Byron Station organization is a Station Quality Control Group, headed by the Supervisor of Quality Control. This group of approximately six to ten people is responsible for quality control activities at the Station such as reviewing drawings, specifications, maintenance/modification procedures, and purchase requests for fulfillment of applicable quality requirements; performing receipt inspection for ASME and safety-related incoming materials and items; inspecting fabrication and installation activities; and ensuring that nondestructive examination and other testing is performed as required. Querio, ff. Tr. 2714, at 10-11; Tr. 2535, 2537, 2543 (Shewski); Tr. 2718 (Querio). The Station Quality Control Supervisor reports to the Administration and Services Assistant Superintendent in order to function independently of the Station operating and maintenance groups. Shewski, ff. Tr. 2364, at 7; Tr. 2537 (Shewski); Tr. 3564-65 (Spraul).

*Byron Quality Assurance Groups*

D-59. The Byron Quality Control Group is to be distinguished from the Byron Quality Assurance Groups. There are presently twenty-nine quality assurance employees on site at Byron. These employees comprise two different quality assurance groups: (1) the Station (or Operating) Quality Assurance Group, and (2) the Construction Quality Assurance Group. Tr. 2536 (Shewski). Edison's Construction Quality Assurance and Station Quality Assurance Groups function independently of both the Station Operating Department and the Project Construction

Department. The Station Quality Assurance Supervisor reports off site to the Director of Quality Assurance for Maintenance and to the Director of Quality Assurance for Operating Activities. They in turn report to the Corporate Manager of Quality Assurance, and the Corporate Manager reports directly to the Vice Chairman of the company. Similarly, the Byron Construction Quality Assurance Superintendent reports to the Quality Assurance Director for Engineering and Construction, who, in turn, reports to the Corporate Manager of Quality Assurance. Shewski, ff. Tr. 2364, at 7-8.

*Byron Construction Quality Assurance Group*

D-60. Applicant's Construction Quality Assurance Group at Byron, headed by the Quality Assurance Superintendent, is composed of approximately twenty people and is responsible for ensuring that the Byron plant is constructed in conformance with Commission regulations. The Construction Quality Assurance Group fulfills this responsibility by conducting audits and inspections of work done by contractors and materials supplied by vendors. Tr. 2545, 2559 (Shewski). The incumbent Construction Quality Assurance Superintendent at Byron has been in that post since January 1981. At that time the quality assurance management on site was materially strengthened by creation of the post of "superintendent" and two additional quality assurance "supervisors."

D-61. Before 1981, a site quality assurance "supervisor," as compared to "superintendent," was in charge of the site quality assurance organization at Byron, supported by lead technicians covering structural, electrical, mechanical and documentation areas of activity. Prior to the superintendent, there were, in turn, four quality assurance supervisors since 1976. Shewski, ff. Tr. 2364, at 13-14.

D-62. One supervisor was killed in an automobile accident. The three other supervisors who preceded the superintendent were replaced according to a stated normal corporate management development and promotional sequence available to "promising management personnel." To have deprived these people of promotion would have resulted in their leaving Commonwealth Edison Company, according to Walter Shewski, Applicant's corporate-level Manager of Quality Assurance. *Id.* at 15. He testified that these changes in quality assurance personnel did not have an adverse effect on quality assurance implementation at Byron. As construction work progresses through different project phases (*e.g.*, from concrete and structural work to mechanical and electrical work to preoperational testing and start-up modes) it is beneficial to replace periodically the quality assurance person in charge with someone new who has experience matching a particular project phase. *Id.* at 16. Continuity during a supervision change tends to be sustained through the overall membership of the site quality assurance group. *Id.*

D-63. During the NRC Construction Assessment Team inspection of Byron in Spring 1982, Region III noted the frequent turnovers of supervisors and the transfer of three QA personnel to other nuclear stations and expressed concern that the constant change of personnel resulted in a "minimum experience level" in the quality assurance staff. This low level of experience was contrasted with the relatively high level of experience in key production personnel. Applicant Ex. 8, at 16, 17. The concern was noted as an open item but never became an item of noncompliance. Shewski, ff. Tr. 2364, at 15.

D-64. We infer from the Staff's inspection report that its concern was twofold. One, the construction quality assurance supervisors might have insufficient experience to meet their direct responsibilities, *i.e.*, training and supervising the QA staff. The other concern was that the relatively inexperienced QA supervisors lacked status vis-a-vis top-ranking production personnel — status obviously needed in a quality assurance program. Intervenors point to another possible problem, that the rapid promotions from QA to production were an indication that the company placed a low value on the QA function. Intervenors' Proposed Findings at 34, 36.

D-65. The turnover sub-issue is not easily resolved. The rapid promotions out of the QA function indicate that talented personnel were selected for the job in the first instance, but, as Intervenors complain, the position was not big enough or important enough to hold that talent. Mr. Shewski's testimony that the Byron QA staff changed and upgraded as the nature of the work became more complex is logical and unrefuted. The status of inexperienced QA supervisors, in comparison to the high-ranking experienced production personnel, was a problem never directly addressed in the testimony. But whatever their status, the construction QA supervisors, had the authority to stop work and upon several occasions exercised the authority. Shewski, ff. Tr. 2364, at 17.

D-66. On balance the Board believes that the rapid turnover was undesirable, but we can find no evidence that the QA function was directly affected by it. In any event, the matter was appropriately addressed by upgrading and enlarging the site QA organization when Mr. Stanish was appointed Byron QA Superintendent with two supervisors assisting him in January 1981. Whether the Byron Construction Quality Assurance Organization was effective or not will be addressed below in connection with the experiences with individual construction contractors. As an organizational structure, we find the Byron Construction Quality Assurance Group to be logically structured so as to provide for independence of that function from cost and production considerations.

*Byron Operational Quality Assurance Group*

D-67. Applicant's Station (or Operating) Quality Assurance Group, composed of approximately nine people and under the direction of a Station Quality Assurance Supervisor, is responsible for the operating quality aspects of the Byron Station. Tr. 2536 (Shewski). This group will be primarily involved in inspections, surveillances, and audits of all safety-related and ASME Code-related work performed by operating plant personnel, contractors, and other Applicant personnel. Shewski, ff. Tr. 2364, at 5. This group has two subgroups, one responsible for plant operators, *i.e.*, control room operators and equipment operators out in the plant, the other responsible for plant maintenance activities. Tr. 2541 (Shewski).

D-68. The Station Quality Assurance Supervisor reports off site to the Director of Quality Assurance for Maintenance and to the Director of Quality Assurance for Operating Activities. They in turn report to the Corporate Manager of Quality Assurance and the Corporate Manager reports directly to the Vice Chairman of Commonwealth Edison. Shewski, ff. Tr. 2364, at 7-8.

D-69. Two senior reactor operators (SRO) are required to be on site at Byron at all times. The Byron Station will have sufficient SRO personnel so that under normal circumstances at least three SROs will be at the Station. A senior SRO on the Station staff will be assigned on-call duty so that administrative-level support is available to the shift engineer on a 24-hour-a-day basis. Querio, ff. Tr. 2714, at 7; Tr. 2608-10 (Reed).

D-70. John Spraul, an NRC Staff quality assurance engineer, testified that Applicant's description of its operational quality assurance program meets the requirements of 10 C.F.R. Part 50, Appendix B. Spraul, ff. Tr. 3562, at 2-3. Based upon Mr. Spraul's testimony, the Board finds that Applicant's organization provides the Quality Assurance Department sufficient independence from cost and scheduling, sufficient authority to effectively carry out quality assurance program operations, and sufficient access to management at a level necessary to perform quality assurance functions. *Id.*; Tr. 3575, 3578 (Spraul).

D-71. The Board concludes that the various quality assurance and quality control organizations within Applicant's corporate structure are suitably designed to carry out their functions, that they possess sufficient independence from cost and scheduling consideration, and that their respective access to management is at a level necessary to perform the quality assurance function. Accordingly, we conclude that Applicant has prevailed on that aspect of the quality assurance contention which asserts that "Applicant's quality assurance program does not require sufficient independence of the quality assurance function from other functions within the Company."

### 3. *Trip Breaker Demonstration*

D-72. The Board requested an evidentiary presentation concerning Byron's automatic reactor scram (trip breaker) systems in light of the failures in February 1983 at the Salem Nuclear Generating Station. We made the request in part because of the safety considerations and in part as a spot check on Applicant's operational quality assurance. Applicant presented Mr. Querio, Byron's Station Superintendent, and Mr. Sues, Byron's Assistant Superintendent of Maintenance and Stores. The resident NRC inspectors at Byron also testified.

D-73. At Salem, Westinghouse type DB-50 trip breakers failed to automatically "trip" or shut down the nuclear reaction in the reactor. Tr. 3997, 4056-57 (Querio). A trip breaker is a device which, on signal, "trips" or opens its contacts, interrupting the power supply to the control rods, causing them to fall into the reactor, shutting down the reaction. Tr. 3993 (Querio); Tr. 4001-02, 4008-13 (Sues). This system at Salem failed because improper maintenance and lack of a preventive maintenance program allowed dirt to accumulate within the trip breaker's internal mechanism, jamming it and preventing it from functioning. Tr. 4060-63 (Querio); Tr. 4085 (Connaughton).

D-74. There were also isolated failures of Westinghouse type DS-50 trip breakers at Commonwealth Edison's Zion Station caused by improper maintenance. Tr. 4043-44 (Querio); Tr. 4045 (Sues). Since the Zion maintenance procedures were corrected, however, there have been no failures. Tr. 4045 (Sues).

D-75. Byron uses Westinghouse type DS-416 low-voltage switchgear trip breakers, classified as safety-related equipment. Tr. 3997, 4023 (Sues). Although these function similarly to the type used at Salem, they have different internal mechanisms. Tr. 3997 (Sues). Each reactor unit at Byron is controlled by two trip breakers in series. Tr. 3993 (Querio). This provides redundancy.

D-76. A Westinghouse DS-416 bypass trip breaker from the Byron Station was used as a demonstrative exhibit during the hearing. Mr.

Sues described and demonstrated its parts and operation. Tr. 4003-12 (Sues).

D-77. Prior to the Salem event, Byron developed a general preventive maintenance inspection procedure for Westinghouse switchgear which included the DS-416 trip breaker. The procedure incorporated the vendor (Westinghouse) manual instructions and recommendations for breaker maintenance. Tr. 4016-19 (Sues); Applicant Ex. 13. The Commonwealth Edison Quality Assurance manual requires such reference to vendor manuals in all maintenance procedures. *Id.*

D-78. Shortly after Commonwealth Edison received NRC notification of the failure of the automatic trip breakers at Salem, these maintenance procedures were reviewed. Tr. 4016 (Sues). A new procedure for the inspection of DS-416 trip breakers, separate from the general switchgear maintenance procedure, was developed. Tr. 4019 (Sues); *compare* Applicant Ex. 13 with Applicant Ex. 14.

D-79. Mr. Connaughton of the NRC testified that, since the trip breakers were classified as safety-related equipment, their procurement, receipt, storage, handling, preventive and corrective maintenance, and testing had to be governed by Commonwealth Edison's Quality Assurance Program. Tr. 4079 (Connaughton). Since the Salem event, the NRC has directed specific licensee testing of trip breakers. Tr. 4080 (Connaughton). The NRC's Region III office will perform inspections at Byron and other plants to verify that the applicable tests were made. *Id.* The Board is satisfied with the procedures assuring reliable operation of the Byron reactor trip breakers.

### 4. *Quality Assurance Oversight of Construction Contractors*

D-80. Criterion I of 10 C.F.R. Part 50, Appendix B, permits the Applicant to delegate to its contractors or consultants the role of establishing and executing the quality assurance program, but, of course, the Applicant remains responsible for the program. Accordingly, at Byron, Applicant's Construction Quality Assurance Group does not directly perform all the audits or inspections of work done by contractors and equipment supplied by vendors. Tr. 2370 (Shewski); Tr. 3686 (Williams). Instead, each contractor or vendor is required to have its own program of inspections and audits and to employ trained and qualified quality control inspectors. Tr. 2525 (Shewski).

D-81. All incoming equipment and materials are inspected by Applicant, or by the appropriate contractor, to ensure physical integrity and compliance with procurement document requirements. In addition, for ASME Code and safety-related items that have not been inspected at the vendor's plant, specific receipt inspection measures, such as material and dimensional checks against approved drawings and specifications, are performed to verify compliance with procurement requirements. Shewski, ff. Tr. 2364, at 6-7; Tr. 3686 (Williams). How well this receiving inspection program works is the subject of an important sub-issue in this proceeding, and particularly the allegations of Mr. Stomfay-Stitz, a former Byron contractor employee, which we address below. Paragraph D-215, *et seq.*

D-82. Applicant's Byron Construction Quality Assurance Group has conducted regular audits and surveillances of the construction work and contractor inspection activities. This quality assurance group also verifies that appropriate corrective action is taken to remedy deficiencies, whether identified by quality assurance, Station quality control, or others. Shewski, ff. Tr. 2364, at 5-6, 26.

D-83. An audit is a formal investigation of the work activities of contractors. Applicant's trained quality assurance personnel rely in part on documents generated by the particular contractor being audited. In addition, the auditors observe the operations and activities taking place at the site. Tr. 2373-74, 2376, 2569 (Shewski). In 1976, 1977 and 1978, Applicant's Byron site quality assurance group performed 37, 50, and 68 formal audits, respectively. Shewski, ff. Tr. 2364, at 26.

D-84. Intervenors challenge Applicant's auditing of its contractors' quality assurance work in part because, in Intervenors' view, the auditors improperly rely on contractors' documents which, according to Intervenors, the auditors accept uncritically as being true — even those very documents, for example, which Intervenors' witness, Mr. Stomfay-Stitz, states were untrue. Intervenors' Proposed Findings 94-95. Mr. Shewski's testimony, however, was to the contrary. The auditors are trained as auditors. They look for alterations and other indications of unreliability in documents and make external inquiries to determine whether the documents are acceptable as a basis for audit. Tr. 2376-77. However, they are not investigators. *Id.*

D-85. A surveillance is less formal and entails observing work being performed, and determining whether that work conforms to written procedures. Tr. 2371 (Shewski). In 1977 and 1978 Applicant's Byron site quality assurance group performed 486 and 550 surveillances, respectively. Applicant's project construction department also performed surveillances of the contractors' work activity. Shewski, ff. Tr. 2364, at 26.

D-86. An independent testing agency, the Pittsburgh Testing Laboratory (PTL), performs in-line acceptance inspections and over-inspections of portions of all contractors' work. *Id.*; Tr. 2381 (Shewski).

PTL has approximately fifty people at Byron. PTL reports directly to the Construction Quality Assurance Group. Tr. 2545 (Shewski). PTL's activities are not limited to inspection of contractors that have experienced items of noncompliance as a result of NRC inspections, but involve all contractors. Tr. 2567 (Shewski).

D-87. PTL's range of over-inspection is from 5 to 100 percent, depending upon the circumstances. Generally, a 100 percent over-inspection is reserved for situations of quality concern. Tr. 2567 (Shewski). Currently for example, PTL is performing an over-inspection of a reinspection of the heating, ventilation and air-conditioning system at Byron installed by Reliable Sheet Metal. Tr. 2514 (Shewski); Tr. 2664 (Stanish).

D-88. In September 1982, Applicant's corporate Manager of Quality Assurance, Mr. Shewski, instituted a "unit concept" inspection. Under this program, every week an element of the plant is selected, and every aspect of that element is reinspected by PTL. Applicant often selects a space between two floors and bound by four columns, and PTL then reinspects everything contained within that volume. Tr. 2572 (Shewski). The unit concept inspection has been used to reinspect the entire diesel generator room. Tr. 2572 (Shewski). However, some items are inaccessible to reinspection, *e.g.*, bolts that have been concreted in. Tr. 2590 (Shewski).

D-89. Applicant's quality assurance program also requires that a management audit of the program's implementation, both during construction and operation, be performed every 2 years. Applicant hires an independent organization to perform the management audit, and the results are reported directly to Applicant's corporate Vice Chairman. Tr. 2569 (Shewski).

### 5. *Contractors at Byron*

D-90. In the following discussion of the contractors at Byron we have, in one way or another, looked at most of the contractors doing safety-related work there. Despite the extended length of this portion of our decision, it is by no means a systematic or complete review of the quality of the construction at the Byron Station. Information selected for specific adjudication derived from two basic sources — the NRC Staff inspections and the allegations of workers sponsored by Intervenors.

D-91. The NRC Staff actually inspects only 1 to 2 percent of the activity at a construction site. Tr. 3685 (Hayes). It depends mostly upon a review of the QA programs of the Applicant and the contractors, and a review, on a sampling basis, of the records documenting the QA/QC inspections and audits by the Applicant and the contractors. Region III Testimony, ff. Tr. 3586, at 5. The NRC inspection program is not designed, considering its resources, to identify every problem of material and workmanship (*id.*) and, in practice, the program finds only a small number of them. Tr. 3691 (Forney). In addition, only those aspects of the Staff's inspections that happened to be selected by the Intervenors or the Board were subject to thorough and specific consideration in the hearing.

D-92. Worker allegations considered directly by the Board are necessarily random. However, the Staff's inspection procedure is to thoroughly inspect all worker allegations, and the results of the inspections are in turn reflected in the Staff's position on the adequacy of the Applicant's and contractors' quality assurance programs.

D-93. In sum, the specific factual situations considered by the Board in the quality assurance litigation were initially identified on a sampling basis by the NRC Office of Inspection and Enforcement, or on a random basis as a result of worker allegations.

*Systems Control Corporation (SCC)*

D-94. Systems Control Corporation is a supplier of safety-related electrical and control equipment at Byron including cable trays and supports, instrument racks and main and local control boards. According to a Region III Inspection Report (80-04):

> A Commonwealth Edison audit of Systems Control Corporation on May 19-20, 1977 pointed out major deficiencies in SCC's implementation of their QA Program. The major findings of that audit included:
> 1. No documented evidence of any receiving, in-process, or final inspections.
> 2. No indoctrination or training program for new or existing employees, and no evidence of training for inspection, test or audit personnel.
> 3. No evidence of procedure qualifications for welding, material coatings, or NDE.
> 4. No evidence that welders had been qualified to AWS D1.1 criteria.
> 5. No evidence that any NDT personnel or procedures had been qualified to ASNT-TC-1A [sic].
> 6. No evidence of review and acceptance of suppliers' QA Programs.

Intervenors Ex. 8, at 29-30.

D-95. In addition, the 1977 audit revealed that Systems Control had failed to perform one of their own audits scheduled for January

1977. Tr. 2505-06 (Shewski). As a result Applicant issued a stop-work order in May 1977 and made a "50.55(e)" report to the NRC.[56]

D-96. The stop-work order was soon lifted, however — on June 10, 1977. Tr. 2507 (Shewski). Problems with SCC persisted, and in March 1978 Applicant's inspection of SCC's main control boards identified three nonconformances on one of the boards. Intervenors Ex. 8, at 25. In September 1978, more, but unspecified, problems with SCC were found. In June 1979, Applicant's QA surveillance noted lack of inspections by SCC and "questionable" SCC welds. A second formal stop-work order was issued against SCC. Tr. 2507 (Shewski); Intervenors Ex. 8.

D-97. The NRC became involved again with SCC in February 1980 after receiving allegations that local instrument panels fabricated by SCC had nonconforming welds. Region III Testimony, ff. Tr. 3586, at 29-30. An investigation was initiated at the Byron site and at SCC's plant. *Id.* A number of items of noncompliance were identified. Tr. 3843 (Williams).

D-98. After a former employee alleged that SCC was improperly implementing its QA/QC program, Region III conducted another investigation of SCC in March 1980 which produced Inspection Report 80-04. Intervenors Ex. 8. Region III's findings were serious, and several major allegations were substantiated.

D-99. Systems Control's QA/QC manager was not qualified. The American National Standards Institute (ANSI) standard required a graduate of either a 4-year accredited engineering or science college plus 5 years experience in quality assurance, 2 in nuclear and 3 in equivalent to nuclear. Alternatively the QA/QC manager could be a high school graduate with 8 years equivalent experience and 2 years in nuclear. The incumbent QA/QC manager had 3 years of college business administration and, purportedly, 11 years of quality assurance experience. Of those 11 years, 6 years were as a furniture manufacturer's quality control inspector, and he had never been involved in the nuclear industry before his employer, SCC, accepted Class 1 nuclear safety-related projects. In fact, the record does not demonstrate any prior experience equivalent to nuclear work. Intervenors Ex. 8, at 11.

D-100. Inspection Report 80-04 also found that, beginning in 1977 through July 1979, the semi-annual SCC internal audit reports of its Quality Control, Engineering, Production, Receiving and Purchasing

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[56] Section 50.55(e) provides in pertinent part:

> [T]he holder of the permit shall notify the Commission of each deficiency found in design and construction, which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant, and which represents:
>
> (i) A significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B to this part; or....

Departments *were falsified.* The unqualified QA/QC manager was deeply involved. *Id.* at 6-9.

D-101. The unqualified QA/QC manager, who was involved in false audit reports, as noted above, was also the project engineer for a relevant safety-related job and reported to SCC's Chief Engineer, rather than to the Executive Manager, contrary to SCC's organizational chart. This dual role effectively eliminated separate review and approval of engineering documents by the cognizant QA official. *Id.*

D-102. The NRC inspectors were mindful that Applicant had earlier become aware of similar major shortcomings in SCC's quality assurance program in the May 1977 audit and cited Applicant for failure to take timely and effective corrective action. The inspectors found further:

> During this [March 1980] investigation, the RIII inspectors found that, in spite of the three-year history of deficiencies in the SCC QA Program and in equipment fabricated by SCC, Byron Station Construction Department personnel waived, without QA concurrence, final inspection of twenty safety-related local instrument panels at the SCC plant during the period from December 1979 to February 1980. The twenty local instrument panels were then receipt inspected at the Byron Site by CECo Station Construction Department, with no significant deficiencies noted, placed in the Unit 1 containment, and were later found, on reinspection in place, to require extensive repairs.

*Id.* at 30.

D-103. At the time of the hearing, the problems with SCC remained an open item. As many as 40 to 60 percent of the welds on the local instrument panels were unacceptable. Tr. 3847 (Hayes). The Department of Justice was inquiring into the issue of falsified records at SCC and it is still an open question as to whether the falsified records extended to the qualifications of other personnel at Systems Control. Tr. 3853 (Hayes).

D-104. On the other hand the defective welds may not represent a direct safety problem. In February 1980, Applicant assigned personnel of the Pittsburgh Testing Laboratory (PTL), the independent testing agency at Byron, to the SCC plant to do a 100 percent reinspection of all items. All items were required to pass inspection by PTL before being shipped either to Byron or Braidwood. Tr. 2579 (Shewski). Panels already shipped and received at Byron were reinspected and repaired. Tr. 2509, 2579 (Shewski); Tr. 3898-99 (Hayes, Williams). As a result of the modification and repairs, however, Westinghouse must perform another seismic analysis, and Region III awaits the Westinghouse report. Tr. 3898-99 (Hayes).

D-105. Applicant discontinued new purchases from SCC in January 1978. As a result of Region III's findings, Systems Control has been barred from procurement activity on safety-related purchases indefinitely. Intervenors Ex. 8, Attachment A, at 3.

D-106. Applicant urges the Board to find that its response to the problems with SCC was very responsible, citing the testimony of Messrs. Hayes and Williams of Region III. Proposed Finding 529. It is true that Messrs. Hayes and Williams believe that Applicant acted responsibly, but their testimony falls short of unrestrained acclaim. Mr. Hayes stated:

> Q. Did the problems with Systems Control cause you to identify any corporate policies or attitudes on the part of Commonwealth Edison?
>
> A. No, I thought they were very responsible. You might fault them for not immediately taking corrective action, but I think we both knew that the problem was not going to go away. The equipment is quite large, and it is hard to hide, so they knew and we knew that the problems were there, and they knew that we were going to insist that it be corrected before that plant operates.

Tr. 3836.

D-107. Mr. Williams noted that corrective actions were initiated and taken by the Applicant and that they met their responsibility under 10 C.F.R. 50.55(e) to report the difficulties to the NRC early in the development of the problem. With one or two exceptions, Applicant, in Mr. Williams' view, acted completely responsibly. The exception that he recalled, and referred to as a perturbation, was when Applicant waived inspections of Systems Control products. Tr. 3852-53 (Williams); *see* Paragraph D-102, *supra*.

D-108. The Board can find, as urged by Applicant, that it acted responsibly in reporting its troubles with Systems Control. The Staff's testimony to that effect is clear and unrefuted. Section 50.55(e)(2) requires such reports within 24 hours. However, the situation with respect to Systems Control is very bad. Region III did not applaud Applicant's oversight of Systems Control's quality assurance program. Applicant has not produced any explanation of how the situation deteriorated as it did. Applicant can take no credit for discovering the latest of SCC's deficiencies — those involving SCC's false audit reports and the unqualified QA/QC manager. The findings after inspection by Region III in Inspection Report 80-04 were the result of allegations by a former employee. There is no assurance that the problems with Systems Control would have come to light in normal course in this hearing because the matter is being pursued by the Department of Justice. A copy of the inspection report was inadvertently supplied to Intervenors. Region III Testimony, ff. Tr. 3586, at 32; Tr. 2501-02 (Whicher-Young).

D-109. The Board concludes that the Systems Control quality assurance program broke down, was unreliable and fraudulent, and that Applicant defaulted in its responsibility to be assured of the adequacy of Systems Control's quality assurance program as required by Criterion I of Appendix B to Part 50.

*Reliable Sheet Metal*

D-110. Reliable Sheet Metal is the Byron contractor responsible for installation and inspection of safety-related heating, ventilating and air-conditioning (HVAC) systems and components.

D-111. On September 17, 1982, Applicant ordered Reliable Sheet Metal to stop work on all new installation of safety-related HVAC systems and attachments to safety-related structures. The stop-work order was issued because of inadequate and incomplete inspections, inadequate procedures, lack of documented evidence that some material purchased by Reliable met procurement requirements, and a number of open audit deficiencies. Shewski, ff. Tr. 2364, at 19; Tr. 2513-17 (Shewski).

D-112. The stop-work order against Reliable remained in effect at the time of the hearing, and it was to continue until Reliable's QA/QC program becomes entirely acceptable to Applicant's quality assurance organization. Tr. 2580 (Shewski).

D-113. Reliable's QA/QC organization has been reorganized, expanded and retrained. There is a backfit inspection program in progress and Applicant's audit schedule for Reliable will be accelerated after work resumes. Shewski, ff. Tr. 2364, at 19-20. In addition there will be a 100 percent independent over-inspection of Reliable's work by Pittsburgh Testing Laboratories. Tr. 2514 (Shewski); Tr. 2664 (Stanish).

D-114. Intervenors assert that the flaws in Reliable's quality assurance program existed from a 1978 violation, discussed below, until the stop-work order in September 1982. Reply Findings at 13. The background of the stop-work order is that Applicant had problems with the Zack Company, a HVAC contractor at LaSalle during 1981 and 1982. This led Applicant to audit HVAC work at Byron and at another site which in turn led to the stop-work order against Reliable. Shewski, ff. Tr. 2364, at 22. There is no direct evidence either way relevant to Intervenors' assertion that Reliable's work at Byron was flawed during the period 1978 until the stop-work order. Also, Intervenors' related claim that Applicant took no action as a result of its experience with the Zack Company at LaSalle is refuted by the events leading to the stop-work order against Reliable.

D-115. The evidentiary record with respect to the Reliable stop-work order is insufficient to support any major conclusion with respect to Commonwealth Edison's quality assurance program. The audit and stop-work order in September 1982 can, on one hand, indicate Applicant's diligence. But on the other hand, perhaps the Applicant should not have allowed the situation to become so serious as to require a stop-work order, reinspection and 100 percent over-inspection.

D-116. Intervenors' apparent concern is that the root cause for the problems with Reliable have been continuing since at least 1978. Reply at 13. Region III cited Applicant in 1978, in part, because Reliable failed to prescribe, in an Applicant-approved and documented instruction, the experience required for quality assurance and inspection personnel with reference to American National Standards Institute (ANSI) standard N45.2.6-1973, Section 3. Intervenors Ex. 3, Attachment, at 1.

D-117. As corrective action, Applicant assured Region III that Reliable had rewritten its QA personnel procedures to require compliance with the 1973 ANSI standards, and that Applicant had approved the revision. *Id.*, Attachment, at 2.

D-118. However, during the special NRC Construction Assessment Team inspection in the Spring of 1982, the team found Reliable did not require its inspection personnel to be trained and certified to ANSI N45.2.6-*1978*. The certification record for the Reliable QA/QC supervisor did not contain a satisfactory basis for his certification and the record did not reflect his level of capability. As a consequence of this finding against Reliable and six other contractors doing safety-related work at Byron, Region III has insisted upon a very extensive inspector recertification and reinspection program at Byron, which we discuss in greater detail below. Were it not for the fact that Reliable had already been subject to a 100 percent over-inspection by Pittsburgh Testing Laboratories, Region III would have insisted that Reliable be a part of the enforcement reinspection. Ff. Tr. 7801, at 6.

D-119. Applicant's failure to assure that Reliable's inspection personnel complied with appropriate training and certification standards reflects poorly on Applicant's record of quality assurance performance. However, the Board does not find that Reliable had continuously failed to meet the appropriate ANSI QA personnel and inspection standards from 1978 until the 1982 citation. The 1978 ANSI standards superseded the 1973 standards and it was not until March 1981 that Applicant committed itself and its contractors to the 1978 ANSI standards. Tr. 7819-23 (Forney).

*Hunter Corporation — Mr. Smith's Allegations*

D-120. The Hunter Corporation is the contractor responsible for the installation and inspection of piping and piping supports at Byron. Intervenors' case regarding Hunter Corporation depended initially and principally upon the allegations of Michael Smith, a former Hunter employee. Findings by Region III as a result of the inspections by Mr. Yin, an NRC engineer, are also relevant.

D-121. Mr. Smith worked at Byron from November 1978 until January 1980. Smith, ff. Tr. 3243, at 1. He was hired to perform surveillance inspections. His 2-month training included written tests on Hunter's site implementation procedures (SIP) and quality assurance manual. Mr. Smith also received on-the-job training as a surveillance inspector. *Id.* at 13, 14. After working as a surveillance inspector for 3 months, he became an auditor in Hunter's QA program. *Id.* at 14-15.

D-122. Before addressing the specifics of Mr. Smith's allegations, a few observations about Mr. Smith's testimony, credibility, and the general nature of the Intervenors' case respecting the Hunter Corporation would be helpful. As is their right, Applicant and Staff request the Board to find that Mr. Smith was fired by Hunter in January 1980 for 20 percent absenteeism and that Hunter would not rehire him. This is in fact the case. Tr. 3244 (Smith). The invited inference, of course, is that Mr. Smith was therefore motivated to bring inaccurate charges against Hunter. However, the Board could not discern any tone of revenge or bitterness in his account of the relevant events. He freely acknowledged that he was fired and seems to concede that it was for good cause. *Id.* Neither he nor Intervenors suggest that it was a "whistleblower" firing. An inference could also be drawn, if the Board were so inclined, that Mr. Smith, now separated from Hunter Corporation, with no chance of rehire, is free from any perceived economic incentive to withhold information critical of Hunter. On balance we see no credibility implications in the circumstances of his firing, with the possible exception that a 20 percent absentee rate might have affected the continuity of his work, thus his perceptions.

D-123. The Board, however, is troubled by the large number of important inaccuracies in Mr. Smith's original allegations compared with his testimony at the hearing. Mr. Smith swore in an affidavit dated September 21, 1982 to many inadequacies in the Hunter QA program. The affidavit was originally attached to Mr. Smith's direct written testimony before the latter was received in evidence on April 5, 1983. Ff. Tr. 3243. Commendably, Intervenors' counsel presented this testimony with several substantive corrections to the statements in the affidavits. Mr. Smith explained that subsequent document review had refreshed his memory. Smith, ff. Tr. 3243, at 4-6. Subsequent cross-examination of Mr. Smith forced him to modify other important allegations in his affidavit and even in his direct testimony. *E.g.,* Tr. 3257, 3259, 3267-68, 3269, 3273, 3276-77.

D-124. As a result, the Board declined to accept Mr. Smith's September 21, 1982 affidavit into evidence until it had been marked to demonstrate that eight of the paragraphs had been modified by subsequent testimony and one paragraph deleted. It was received as a separate exhibit, Intervenors Ex. 21.

D-125. Intervenors' proposed findings on Mr. Smith's allegations are rather sparse and summary given the number and scope of his charges, and the proposed findings do not fully track his written testimony and modified affidavit. Proposed Findings 41-47. We do not, however, require Intervenors to vouch for all of the allegations made by Mr. Smith and the other worker witnesses, and we accept Intervenors' proposed findings, not Mr. Smith's affidavits and testimony, as their statement of the case.

D-126. *First Allegation: Hunter production workers were under pressure to work quickly, and as a result were doing shoddy work.* Proposed Finding 41. In support of this allegation Intervenors offer only Mr. Smith's written testimony. Ff. Tr. 3243, at 21-22. He stated that two pipefitters and many production workers told him this — apparently with respect to pipe supports. *Id.* No further evidence was developed directly on the allegation, except that Mr. Smith conceded on cross-examination that he could not identify nor is he aware of any supports with bad welding or any out of the prescribed tolerance. Tr. 3275. The allegation was insufficiently specific to require a more precise response by Applicant. The Board could find no corroboration of the charge of general shoddiness in analyzing Mr. Smith's specific complaints or in reviewing the relevant Region III inspection reports. We find that the allegation is not substantiated.

D-127. *Second Allegation: As confirmed by a Region III inspection, Hunter lacked a prescribed program for pipe hanger inspection.* Proposed Finding 42. The issue, as litigated by the parties, involved the location of pipe supports, not the quality of their welding. Mr. Smith alleged that in preparing a checklist for an audit in 1979 of hanger process control (referred to throughout his testimony as "Audit 059-3"), he became sufficiently aware, for the first time, of Sargent & Lundy's (Byron's architect-engineer) manual "M916" which sets adjustment tolerances when piping components cannot be installed in accordance with construction drawings and provides for documentation of "as-built" data when the M916 tolerances were used. Smith, ff. Tr. 3243, at 31, Exhibit

G. Mr. Smith stated that the auditors just happened to find a reference to M916 in an interoffice memorandum, HC-QA-#23, in the very back of the Hunter Site Implementation Policy (SIP) manual on hanger control. Manual M916 was a crucial document, but Mr. Smith had not been trained in it. Nor, as Audit 059-3 revealed, had the welding quality control inspectors. *Id.* at 33, and Exhibit E. Moreover, "as-built" documentation of the use of M916 was lacking. *Id. See also, id.,* Exhibit C (Audit 059-3 Report), and Exhibit D (follow-up audit).

D-128. As it happened, Mr. Yin, a mechanical engineer with Region III, routinely inspected the hanger support welding process in March 1980 and made findings concerning the failure of location control documentation similar to Mr. Smith's observations. Mr. Yin regards Mr. Smith's allegations to be substantiated. Region III Testimony, ff. Tr. 3586, at 25 (Yin). Mr. Yin regarded the Hunter interoffice memo, HC-QA-23, which referenced M916, to be "not really a procedure" but "informal instruction to request the foreman to document the installation locations." Tr. 3677 (Yin). However, Mr. Yin believes that Applicant's corrective action, discussed below, was effective. Region III Testimony, ff. Tr. 3586, at 25.

D-129. The evidence revealed that the supervisors of the hanger installers, responsible for the correct installation of the hangers, had been provided with and were aware of the M916 procedures. Tr. 3354-55. But the central issue is not whether the hangers happened to be installed accurately, but whether the quality assurance program produced such assurance.

D-130. During the audit, Mr. Smith reviewed documentation for five component supports. Tr. 3268 (Smith). With regard to item 10, support locations, the auditors found that M916 was properly referenced for two of the five supports, incompletely or erroneously referenced for two of the supports, and not referenced at all for one support. Tr. 3372-73 (Smith). Thus, use of M916 was documented on four of the five component supports reviewed, albeit improperly on two of them. Smith, ff. Tr. 3243, Exhibit C. On cross-examination, Mr. Smith agreed, in part, with Applicant's assertion that M916 procedures were being followed, although in instances it was not being properly documented. Tr. 3374 (Smith).

D-131. With regard to checklist item 11, acceptance of "as-built" data, Audit 059-3 found that of the five component supports reviewed, as-built data had been properly accepted by the QC inspectors in only one instance. As corrective actions the audits recommended that the cognizant craft production supervisors must provide proper "as-built" data and that acceptance or rejection of "as-built" data be properly accomplished, either through training of inspectors to M916 or through removal of responsibility for acceptance or rejection from the inspectors. Smith, ff. Tr. 3243, Exhibit C. A follow-up audit to Audit 059-3 indicated that the items relevant to the M916 procedure had been resolved. *Id.*, Exhibit D.

D-132. The Board finds that the essence of Mr. Smith's second allegation is correct; that although there was a procedure for the installation and inspection of pipe hangers, it was too informal, and insufficiently documented and apparently inadequate. We also find that adequate corrective action was taken with respect to the use of M916 for location tolerances and the acceptance of "as-built" data when M916 was used, primarily through the training of quality assurance welding inspectors to M916 and proper documentation of "as-built" data. On balance, because of the prompt and positive corrective action, we do not regard the episode to be a serious reflection on the Hunter QA program.

D-133. *Third Allegation: On at least ten occasions, support weld inspections were documented when in fact they had never been performed.* Proposed Finding 43. With respect to this allegation Mr. Smith testified that:

> Frequently in our surveillances, we would uncover documents that were never signed by inspectors. When that occurred, we were told, by Mr. Somsag, to go out and check all records that inspectors make daily to see if in fact they had inspected an item. If they had not, we were told to go to the Quality Control inspection superintendent, Frank McGhee, and have him initial that particular inspection process. Mr. McGhee would place another inspector's initials in the places where the inspector should have initialed, yet there was no record that any inspector did in fact perform that inspection procedure.
>
> Mr. McGhee would not inspect the item; he would simply place an inspector's initials on the sign-off sheet as if that inspector had actually performed the inspection, and date it as if it had been inspected on a date in the past when it should have been inspected.

Ff. Tr. 3243, at 15.

D-134. Mr. Michael Zeise was the lead auditor at Hunter during Mr. Smith's tenure. His testimony was stipulated by Applicant and Intervenors. Board Ex. 4. He is aware of Mr. Smith's allegations concerning Mr. McGhee and states that he, Zeise, was aware of three instances when Mr. McGhee initialed an inspection report with the initials of another inspector when there is no conclusive record to indicate that the inspector had inspected the weld. In each instance, it was a weld on piping to the river screen house; these were Class 3 welds due to the distance from the reactors. Mr. Zeise believes that subsequent tests would have indicated any deficiencies in the welds and he would have learned of any such deficiency. While he believes that it is possible that there could be instances other than the three known to himself, but known to Mr. Smith, he, Zeise, believes that would be unlikely because of Zeise's position and because Mr. McGhee soon left Hunter. *Id.*

D-135. Mr. McGhee is retired and no longer employed at Hunter. Tr. 3954 (Somsag). He did not testify. Mr. Somsag, Hunter's QA/QC supervisor testified that, to his knowledge the false sign-off by Mr. McGhee never occurred; that if an inspection could not be performed it would be brought to his attention, and that Mr. McGhee, an honest man, would not falsify information.

D-136. Mr. Zeise's stipulated testimony is the best evidence. Mr. Somsag's want of knowledge does not overcome Mr. Zeise's specific knowledge. Mr. Smith's record of exaggerations and his failure to specify the instances of false sign-offs (Tr. 3429) weaken the force of his testimony. Nevertheless, Mr. Smith's third allegation is, in part, substantiated. However, having accepted Mr. Zeise's account as the best evidence, we must also accept his statement that the inspection reports involved did not pertain to safety-related work, that it was a limited practice by one individual. It is not an important reflection on Hunter's QA program.

D-137. *Fourth Allegation: Even when audits were actually performed, Mr. Smith was sometimes instructed not to include in his final reports problems that he had discovered, on the purported excuse that the problem would be caught later on.* Proposed Finding 44. Mr. Smith made this allegation in the context of finding documentation for supports which could not be located. At the time there was a program for handling and reporting such problems known as the "hanger field problem" system. When reporting the missing supports to Mr. Somsag, the auditors would be instructed not to document the missing supports because it would be handled by the "hanger field problem" system and that the missing supports would be identified during the final "walk-down" of the respective system. Sometimes there would be supports but no documents. In short the auditors were told to forget missing supports and documents because the problem would be caught later on, a practice referred to as "tabling" which, according to Mr. Smith, occurred at least once or twice a week. Ff. Tr. 3243, at 22-23; Tr. 3447.

D-138. To demonstrate the scope of the problem, Mr. Smith referred to the five supports audited in Audit 059-3, stating that only one of them had complete and correct documentation. He was told not to include this information in that audit because it was beyond the audit's scope. Tr. 2447-48 (Smith). Mr. Smith's basic concern was that he had no evidence that the missing supports or support documentation had been placed into the "hanger field problem" system; that he would have to take Mr. Somsag's word that the problem would be caught. Ff. Tr. 3243, at 22-23.

D-139. Applicant's response to the "tabling" allegation does not meet the thrust of the charge head on. First, Applicant developed on cross-examination that there might be a good reason for a pipe support being missing in that it could have been temporarily removed to facilitate construction with the expectation that it would be reinstalled. Tr. 3383 (Smith). This observation is, at best, irrelevant. It seems to the Board that a temporarily removed support presents at least the same problem as one not installed in the first instance — perhaps worse if existing documentation shows the missing support in place.

D-140. Second, Applicant established that, not only Mr. Somsag tabled matters, but Mr. Smith and Mr. Zeise also tabled matters on their own. Tr. 3383-84. Aside from demonstrating a patent inconsistency in Mr. Smith's testimony compared with an earlier deposition (*id.*), this circumstance does little to help Applicant. Mr. Smith testified that he would table a matter on his own only when assured that QA management was aware of it. Tr. 3385. In any event, even if Messrs. Smith and Zeise engaged in the same tabling practice criticized by Mr. Smith, it tends to exacerbate the problem, not justify it. Mr. Somsag had, after all, more authority and, presumably, more knowledge to exercise such discretion.

D-141. Mr. Yin of Region III inspected for an allegation by Mr. Smith that one support was found without any documentation. Mr. Yin did not substantiate the allegation. However, Mr. Yin expressed the view that the allegation by Mr. Smith (exaggerated as we later learned) that there was 100 percent noncompliance with proper design locations of supports checked by Mr. Yin could be factual because the QC inspection program had not then been formally established. Region III Testimony, ff. Tr. 3586, at 25-26 (Yin).

D-142. While the Board is not fully confident in the complete accuracy of Mr. Smith's allegations, and his credibility was damaged on this very issue, we conclude that the essence of the "tabling" allegation has been substantiated. Applicant does not deny that there was in fact a tabling practice (Proposed Findings 560-64), but suggests that, based on Mr. Smith's testimony as to his practice, the practice was followed only when the quality assurance personnel had the matter under their control (Proposed Finding 564). Nor does Applicant support the statement attributed to Mr. Somsag that missing supports and missing documents were beyond the scope of Audit 059-3 which as we noted above, related, as pertinent, to support location and "as-built" documentation. Missing supports and documents clearly relate to the audit.

D-143. Mr. Somsag is the only witness who could have explained how he handled tabled matters and what assurance he had that the "hanger field problem" system reliably resolved the matters. Although Applicant knew of Mr. Smith's "tabling" allegations as early as the prehearing deposition, and although Mr. Somsag returned to the hearing in rebuttal, Mr. Smith's "tabling" allegation was not addressed. Tr. 3950, *et seq.*

D-144. The quality assurance auditors should have had a formal documented method to assure that their discovery of missing supports and documentation was properly addressed.

D-145. Subsequently in March 1983, during Applicant's audit of very comprehensive reinspection programs at Hunter and other Byron contractors, which we discuss below, Applicant found that Hunter was not taking appropriate steps to identify, document, segregate, disposition, and notify affected organizations of nonconforming items. Specifically, Hunter was found to have failed to issue nonconforming reports for nonconforming conditions. Field problem sheets, rather than discrepancy reports, were used with respect to component supports and mechanical joints. Intervenors Ex. 29, at A1.

D-146. *Fifth Allegation: Mr. Smith was instructed not to perform a thorough audit of the Authorized Nuclear Inspector (ANI) and, under no circumstances, to tell him that anything he did was wrong.* Proposed Finding 45. Mr. Smith testified that the lead auditor, Mr. Zeise, told him not to dig very deeply into the Authorized Nuclear Inspector's (ANI) work pursuant to Mr. Somsag's instructions. Accordingly Zeise and Smith made their audit of him as superficial as possible. Ff. Tr. 3243, at 16-17. The ANI is trained and certified by the National Board of Boiler and Pressure Vessel Inspectors and served as an independent inspector at Byron. Tr. 2905 (Somsag).

D-147. The basic trouble with Mr. Smith's allegation concerning the ANI is that he doesn't know what an ANI is or how to audit one. He could not describe accurately the ANI's duties. Tr. 3201-05, 3211-12. The prescribed way to audit an ANI is not to check on what he does, and certainly not to tell him what to do, but to determine whether he has access to all of the documents he needs for his work. Tr. 2911-12 (Somsag). In fact, Mr. Smith's own testimony reveals that he did not understand the attachment to his testimony (Exhibit B, ff. Tr. 3243) which instructs the auditor to determine whether the Authorized Nuclear Inspector has had certain documents presented to him with no other instructions. Mr. Smith's testimony indicates that he made a superficial determination that the ANI "has done his work" when such a determination was beyond his jurisdiction. Ff. Tr. 3243, at 16. If Mr. Smith was instructed not to tell the ANI that anything he did was wrong, it was probably an appropriate instruction if it was perceived as necessary to set the bounds of the audit. Mr. Smith's fifth allegation has no substance.

D-148. *Sixth Allegation: Mr. Smith was instructed to "stay out of sight" when NRC personnel were on site, and to answer only "yes" or "no" if they were to ask him any questions.* Proposed Finding 46. Mr. Smith stated that he was given such instructions both by Mr. Somsag and Somsag's assistant every time NRC personnel were on site and he believes that all Hunter QA people were so advised. Ff. Tr. 3243, at 40-41.

D-149. Even if corroborated, however, the allegation has little direct significance with respect to Mr. Smith's other allegations because he had very little knowledge of the NRC's function. Tr. 3245. Even after he was discharged by Hunter, thus free of any restraint, he did not go to the NRC with his allegations. Tr. 3245-46.

D-150. Mr. Somsag flatly denies the charge. Tr. 2906-07.

D-151. NRC Section Chief D.W. Hayes investigated this allegation. He testified that he spoke with individuals at Byron and was unable to find any policy that would preclude personnel at Hunter Corporation or at Commonwealth Edison Company from talking to an NRC inspector. Mr. Hayes concluded that it was not the policy of these entities to prevent any of their employees from talking to an NRC inspector, and he added "[i]n my inspection activities at Byron, I have never run into a case where I could not talk to anyone I wanted to." Tr. 3798-99 (Hayes); *see also* Tr. 3897.

D-152. Mr. Hayes also testified that beginning in 1977, the NRC took actions to meet with workers at nuclear power plants to inform them of the role of the NRC, and NRC inspectors wore hats that were labelled "NRC Inspector" on both sides. Tr. 3894. When Mr. Forney, the Senior Resident Inspector at Byron, arrived on site in October 1981, NRC Form 3 (informing workers of their right to contact the NRC) was posted on bulletin boards, but Mr. Forney increased the number of postings. Tr. 3662-63 (Forney); Tr. 3896 (Hayes).

D-153. Mr. Smith's sixth allegation is not substantiated.

D-154. *Seventh Allegation: Mr. Smith's audit reports were often changed by Mr. Somsag, both substantially and stylistically, in a manner which lessened the impact of the audits by giving those audited more leeway and by deleting many critical passages.* Proposed Finding 47. In support of this allegation Mr. Smith discussed examples of changes imposed by Mr. Somsag found in documents produced by the NRC and Applicant, and attached to Mr. Smith's testimony as Exhibit C, the Audit 059-3; Exhibit D, the follow-up to that audit; Exhibit E, a draft of the follow-up audit; and Exhibit F, Audit 058-2.

D-155. With respect to the first example, as we discussed in preceding paragraphs, one of the corrective actions for the problem of incorrectly inspecting for M916 adjustment tolerances was to train quality control welding instructors to that procedure. Mr. Smith testified that his original finding would have required this training immediately following the audit, an opinion he had noted on a draft audit report. Mr. Somsag, however, changed that recommendation in the final report to the effect that training to the M916 tolerances had been committed to by the QA supervisor, Mr. Somsag himself, and would commence at Mr. Somsag's discretion. Ff. Tr. 3243.

D-156. We do not find that the change imposed by Mr. Somsag is improper in the sense that it changed any audit findings. Since it was Mr. Somsag, not Mr. Smith, who committed to training the welding inspectors, the change was in the direction of accuracy and it was Mr. Somsag's prerogative to make the change. As it turned out the training was instituted within a month, and Mr. Smith agrees that the training was timely. Tr. 3395. Apparently Mr. Smith was annoyed that his boss, Mr. Somsag, would not commit to a specific time for the training. We find no significance adverse to Hunter in the first example of Mr. Somsag's changes.

D-157. The next three examples of changes imposed by Mr. Somsag on Mr. Smith's initial drafts are set out in context in his direct testimony and involve Audit 058-2, which was an audit of the Hunter QA organization itself. Mr. Smith states:

> In the handwritten portion of Exhibit F the changes are evident. . . . On page F-65, Mr. Somsag deleted two sentences which were very critical of the Hunter QA program. One of them reads:
>
> > "In any event, these undoubtedly indicate a lack of indoctrination and training of personnel performing activities affecting quality as necessary to assure that a suitable proficiency is achieved and maintained."
>
> In the auditor's note, also on page F-65, Mr. Somsag deleted another sentence:
>
> > "This could account for the fact of recording training to an obsolete S.I.P, as mentioned above, but cannot be accepted as an excuse for this type of unprofessional act."
>
> In both these sentences, I was indicating deficiencies in the QA organization, of which Mr. Somsag and I both were a part.
>
> On F-66, in my recommendation and commitment for follow-up to a finding, I recommended that:
>
> > "the Project Engineer confirm that personnel in all divisions of the Engineering Department are being trained and documented in a uniform manner to facili-

> tate the minimum training requirements provided in Section 1 of the Hunter Corporation Quality Assurance Manual."

> He replaced my recommendation with one that the Engineering Department "take steps to indicate that training is being performed in good faith."

Smith, ff. Tr. 3243, at 26-27.

D-158. Mr. Smith agrees that at least the first of the deletions was a conclusion leaving the factual basis undisturbed. Tr. 3433. His major criticism is not that facts were dropped from the audit report but that Mr. Somsag reduced the impact of the audit report by dropping purposely critical comments. Ff. Tr. 3243, at 28.

D-159. The Board has no basis upon which to determine whether Mr. Somsag's deletions and modifications in Audit 058-2 were the best things to do. Perhaps Mr. Smith's sterner language was the better response to the audit findings, perhaps not. The controlling point is that Mr. Somsag, not Mr. Smith, was the boss. As the audit form indicates, Mr. Somsag was required to approve the report. *E.g.*, ff. Tr. 3243, Exhibit F, at F-1. He had more experience and it was at a higher level than Mr. Smith's experience. Mr. Somsag was in a better position to determine whether the abrasive language proposed by Mr. Smith would in fact have the impact intended by Mr. Smith, or perhaps produce counter results — aside from whether the critical comments were fair. Without even analyzing Mr. Somsag's reasons for the changes, the Board concludes that there are insufficient bases to find that Mr. Somsag improperly made the foregoing changes to the Audit 058-2 report.

D-160. In still another change in Audit 058-2, Mr. Smith believed that Mr. Somsag replaced deleted language with a recommendation that a "site management committee" be established to participate in training. But, Mr. Smith had never heard of such a committee for QA/QC training nor has he since ever found any indication of such a committee. Ff. Tr. 3243, at 27-28. Mr. Smith stated he wasn't looking for a committee, he was looking for a "commitment." *Id.* at 29. But on cross-examination, Mr. Smith had to concede that he had misread Mr. Somsag's substitution, and in fact Mr. Somsag had recommended a "site management commitment," exactly what Mr. Smith had recommended. Tr. 3420-22.

D-161. Mr. Yin of Region III testified that he examined a number of Hunter Corporation audits in which Mr. Smith participated, as well as audits prepared by other auditors. Mr. Yin concluded that the changes in Mr. Smith's audits were editorial in nature only; the audits continued to include the findings made by the auditors. Mr. Yin concluded also that the paragraphs deleted by Mr. Somsag involved the personal concerns of

Mr. Smith, and that the changes made by Mr. Somsag did not have safety impact. Region III Testimony, ff. Tr. 3586, at 26-27, Attachment G, at 7-9.

D-162. The Board finds that changes in audits prepared by Mr. Smith, and identified by him at the hearing, were not extensive and did not result in the deletion of any audit findings. They were editorial in nature. The evidence does not establish that the changes made by Mr. Somsag were improper.

D-163. However, before leaving the allegation that Mr. Somsag altered audit reports improperly, the Board notes its concern about another aspect of the controversy. Mr. Smith testified that if he had the rough draft to Audit 059-3, he could better support his allegation and that it was Hunter policy to retain such draft notes. Ff. Tr. 3243, at 25. Mr. Somsag originally testified that it was not Hunter's policy to retain the rough drafts. However, he recanted that testimony later, and acknowledged that Hunter policy did require that the auditors' drafts be retained. Although Mr. Somsag searched for the Audit 059-3 draft, he could not locate it. Somsag, ff. Tr. 2883, at 17-18; Tr. 2891-95. Our concern is not that the rough draft could not be found, because there was a sufficient testing of Mr. Smith's allegations on documents and drafts which were produced. Our concern is that Mr. Somsag, the chief Hunter QA official at Byron, was mistaken about a rather simple and important aspect about the Hunter QA auditing procedures. The matter cannot be resolved, however. Our conclusions with respect to this allegation are founded on the preponderance of the evidence. The allegation is not substantiated.

D-164. *NRC Inspections of Hunter* constitute another aspect of Intervenors' quality assurance case. First, Intervenors make a fleeting reference to Region III Inspection Reports 80-05 and 81-09. Proposed Findings 48 and 49. A Region III inspection conducted in March 1980 (Inspection Report 80-05) identified piping suspension systems which were not QC-inspected in concurrence with installation activities. As a result, an Applicant's reinspection program was soon initiated based on the revised Hunter procedures which included more detailed process control checklists and expanded QC inspection criteria. However, during a Region III follow-up inspection conducted in July 1981 (Inspection Report 81-09), a number of snubbers in Unit 1 containment were again found without timely QC inspections. These snubbers were subsequently inspected and an Applicant's review was initiated to identify all other supports and restraints that had not been inspected using the current procedures. Subsequently the Applicant stated that during December 1981, the reinspection of supports was not progressing in accordance with the schedule, and that Hunter had been instructed to step up their review of QC inspection records and to document any support assemblies that were without current inspections in nonconformance reports. To January 1982, approximately 8500 supports were reviewed per the revised inspection procedural requirements. Fifty-five supports did not have inspections completed. However, they were being redesigned and were documented in a nonconformance report. These fifty-five supports were revised and inspected in September 1982. The Region III inspector reviewed the pertinent documentation and considered the Applicant's QC hanger reinspection effort adequate. Region III Testimony, ff. Tr. 3586, Attachment G, at 10.

D-165. As the inspector, Mr. Yin, later explained, the failure to follow the revised procedure to assure timely QC inspections was caused by a foreman in only one particular area misinterpreting the requirement. Tr. 3797. We find that, with respect to Inspection Reports 80-05 and 81-09, the Applicant required Hunter to take adequate and effective corrective action in response to Region III findings. Whether the problem should have arisen in the first instance was not addressed adequately by the evidence — the issue was timeliness. Tr. 2662, 2708 (Stanish).

D-166. Region III Report 80-24 in January 1981 found that Hunter workers bent anchor bolts, without documenting the action, in order to accommodate a safety-related pump diesel motor in Unit 2. Tr. 2653 (Stanish). This Level IV noncompliance was neither discovered nor reported by Edison, and was discovered by Region III just before the bolts were to be grouted over. Tr. 2655 (Stanish). The NRC inspectors then went over the same piece of equipment in Unit 1 and found that the bolts had been similarly bent and grouted without being inspected. Tr. 2655, 2657 (Stanish). It was necessary to replace the bolts in each unit. Tr. 2657 (Stanish).

D-167. Applicant concedes that the foregoing facts, as proposed by Intervenors (Proposed Finding 50), have been established, but responds that the situation was not as bad as Intervenors imply. Mr. Stanish's testimony explained that Applicant did not discover the bent diesel anchor bolts at Unit 1 before they were discovered by the NRC Staff because inspection of this particular equipment had not yet been performed by the contractor. Tr. 2657. In addition, while Intervenors imply that grouting anchor bolts renders them inaccessible for purposes of inspection, the record is to the contrary. Mr. Stanish testified that even after grouting, the tops of the bolts remain visible, and if bent they have a noticeable out-of-plumbness appearance. Tr. 2655. In fact, the bent anchor bolts at Unit 1 were discovered after they had been grouted. *Id.*

D-168. The Board finds that even though the defective bolting can be discovered and corrected after grouting, the incident is nevertheless an adverse reflection on Hunter's quality assurance program. An inspection to be timely should have been scheduled and made before grouting. Tr. 2709 (Stanish). However it is not a matter of great consequence.

Conclusions — Hunter Corporation

D-169. We found most of Mr. Smith's allegations against the Hunter Corporation to be unsubstantiated. The allegation that there was an inadequate and insufficiently documented procedure for the inspection for the location of pipe hangers, the allegation that Mr. McGhee has signed inspection reports without evidence that the nonsafety-related inspection had been conducted, and the finding that anchor bolts for the safety-related pump diesel motor had been grouted over before inspection, we found to be substantiated but individually of no great significance to the Hunter quality assurance program. Collectively, these incidents suggest sloppiness in Hunter's QA program not easily quantified. The allegation concerning the "tabling" practice (not reporting nonconformances pending final "walk-down"), we regard as a serious matter which could have important consequences. We were particularly concerned that Hunter continues to fail to take appropriate steps to issue documentation on nonconforming conditions. Intervenors Ex. 29, at A1.

D-170. Hunter is one of the eight contractors performing safety-related work at Byron found by the NRC special Construction Assessment Team in the Spring of 1982 to be deficient in its standards for certifying the qualifications of QA/QC personnel. As a consequence, Hunter and other contractors are subject to a very extensive inspector recertification and reinspection program involving a large sampling of the Hunter inspectors. Region III Testimony, ff. Tr. 7801, at 6; *see* Reinspection Program, Paragraph D-365, *infra*. In view of Hunter's experience at Byron, we conclude that the reinspection program, if effective, is essential to a verification of the adequacy of Hunter's QA program.

*Blount Brothers Corporation*

D-171. Blount Brothers Corporation is a general contractor at Byron primarily responsible for concrete work, post-tensioning and containment structural steel. Issues concerning Blount were raised by two former employees. Daniel Gallagher makes allegations concerning the quality of concrete production. Mr. Stomfay-Stitz alleges quality assurance problems with particular emphasis upon the receipt and storage of construction materials used by Blount.

Mr. Gallagher's Allegations

D-172. Mr. Gallagher was employed as a concrete batch plant operator for Blount at Byron. He worked for Blount at Byron from August 1975 to November 1977, and from February 1978 to June 1979. Gallagher, ff. Tr. 3459, at 1. He was hired as an apprentice batch plant operator. Tr. 3460 (Gallagher). Blount sent him to a training school which involved the operation of the Erie-Strayer batch plant, one of the two batch plants on the Byron site. Gallagher, ff. Tr. 3459, at 2-4. After working as an apprentice, Mr. Gallagher became a batch plant operator. Tr. 3460 (Gallagher).

D-173. Crucial to the resolution of Mr. Gallagher's allegations was the testimony of Mr. Pope, who testified at the instance of Applicant. Mr. Pope was employed by Blount Brothers as a batch plant operator throughout the entire time Mr. Gallagher was employed at Byron. Mr. Pope has been a batch plant operator for 16 years, having worked at Commonwealth Edison's Zion Station before coming to Byron. Mr. Pope and Mr. Gallagher arrived at Byron at the same time, in 1975, and Mr. Pope still worked for Blount at Byron at the time of the hearing. Pope, ff. Tr. 2833, at 1. As batch plant operators, their responsibilities include the operation and maintenance of the batch plant at the site. As the more experienced batch plant operator, Mr. Pope taught Mr. Gallagher how to mix or "batch" concrete out of the plants. Mr. Gallagher described Mr. Pope as a "conscientious" operator who "would never mix a bad batch," and who taught him "to be a good, conscientious worker who always made a quality product." Even after leaving Byron Mr. Gallagher would seek Mr. Pope's advice about the machines Mr. Gallagher was running on other jobs. Gallagher, ff. Tr. 3459, at 16-17. Both Messrs. Pope and Gallagher have been and are members of Operating Engineers Local 150. *Id.* at 1; Pope, ff. Tr. 2833, at 2.

D-174. The Board observed both Mr. Gallagher and Mr. Pope to be candid witnesses. Below, the resolution of Mr. Gallagher's allegations in the direction of Mr. Pope's testimony is primarily because of Mr. Pope's greater experience, better information and corroboration. As we discuss in greater detail below, however, we were concerned about the very high level of inaccuracies in Mr. Gallagher's testimony.

D-175. Applicant also sponsored the testimony of two employees of Pittsburgh Testing Laboratory, Marvin Tallent, Jr., and Joseph Johnson. As we have frequently noted, Pittsburgh Testing Laboratory (PTL) is the independent testing firm at Byron which, among other functions, performs tests on concrete and its component materials. Tr. 3961-62 (Tallent and Johnson).

D-176. In addition, Region III also investigated and testified concerning Mr. Gallagher's allegations. The investigation report itself was appended to the Staff's prepared testimony as Attachment F. Region III Testimony, ff. Tr. 3586, at 23-25 and Attachment F.

D-177. A batch plant is the facility where the ingredients used in concrete — cement, water, admixtures and aggregate — are measured and mixed. At the time Mr. Gallagher worked for Blount there were two batch plants at the Byron site, the Erie-Strayer and Ross plants. Pope, ff. Tr. 2833, at 3. Only the Ross plant remains at the site. Tr. 2867-68 (Pope). The Erie-Strayer plant was a central mix plant which produced a "wet batch"; that is, the concrete was mixed at the plant and simply had to be transported to the placement site. The Ross plant, in contrast, is older, smaller, and is a "dry batch" plant in that the ingredients are poured unmixed into trucks which then do the actual mixing. The Erie-Strayer plant was computerized; ingredients were weighed and batches mixed automatically. At the Ross plant, on the other hand, the batch plant operator manually controls the weighing and mixing of the ingredients. Pope, ff. Tr. 2833, at 3-5.

D-178. *First Allegation: Blount was under tremendous pressure from Edison to increase production. As a consequence, the more primitive Ross plant, which could not make competent concrete, was used.* Proposed Findings 51-54. Mr. Gallagher alleged that the Ross plant was designed to be used as a backup to the Erie-Strayer and, in no event was it capable of producing Category I (safety-related) concrete. Gallagher, ff. Tr. 3459, at 4-5, 7, 8, 24. At the outset, the Board sees an inconsistency in Mr. Gallagher's own testimony on this point. At one point he acknowledges that the Ross plant was intended as a backup to the Erie-Strayer plant because, where the construction design called for a continuous pour, there must be a backup in case the main plant broke down. *Id.* at 8. The inconsistency, as we see it, is, if Ross was designed as a backup, it must then have the same capability as the plant it backs up. Mr. Gallagher does not suggest that continuous pours are used only in nonsafety-related construction. Containment buildings, for example, are continuously poured and are, of course, safety-related structures.

D-179. Moreover, Mr. Pope flatly disagrees with Mr. Gallagher on that issue. He believes that the Ross plant is as accurate as the Erie-Strayer plant in measuring the ingredients. It is capable of producing safety-related concrete and in fact was later used for that purpose. Tr. 2863-67.

D-180. There is only one clear and specific reason why Mr. Gallagher believed that the Ross plant could not produce safety-related concrete. As we noted, the "dry-mix" Ross plant delivers the measured ingredients to the transporting trucks but depends on the trucks to mix them. Mr. Gallagher believes that the trucks rented by Blount for use at Byron were not mixing machines and were capable only of transport. Gallagher, ff. Tr. 3459, at 5. On the day he was fired for refusing to operate the Ross plant, for example, the temperature was so high that ice would be needed to control the added heat from the chemical reaction after the batch was loaded onto the trucks. With non-mixing transport trucks the ice might arrive at the construction site unmixed and "ice balls" would remain.[57] Tr. 3508-10 (Gallagher).

D-181. However, Region III inspection concluded that no safety-related concrete batched in the Ross plant was transported in trucks without tested "ASTM C-94" uniform mixing capability. The trucks actually used were identified. Region III Testimony, ff. Tr. 3586, at 23 and Attachment F; Tr. 3884 (Hayes). The report noted that rented trucks which had not been properly tested were on site but no such trucks were used at the Ross plant. Tr. 3886-87 (Hayes). As we noted above, non-qualified trucks would be adequate for the "wet-batch" Erie-Strayer plant.

D-182. The Board is aware of no reason why a batch plant such as the Ross plant at Byron would be inherently incapable of producing safety-related concrete. The Ross plant was used infrequently for that purpose, however. Region III Testimony, ff. Tr. 3586, Attachment F; Tr. 3887 (Hayes). Apparently because it was seldom used for safety-related work during Mr. Gallagher's tenure, he concluded that it was not capable of that work. We conclude that Mr. Gallagher was incorrect in that conclusion and we find that his first allegation is not true.

D-183. *Second Allegation: There was no maintenance program for the Ross[58] batch plant and a Blount employee lied to the NRC stating that there was a program.* Proposed Finding 55. Mr. Gallagher testified that when NRC inspectors inquired about the maintenance of the Erie-Strayer plant, Mr. Andre, a Blount employee, falsely told them that Blount had

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[57] Mr. Gallagher testified that he was fired by Blount because he refused to produce concrete out of the Ross plant for the cooling towers which he incorrectly believed to be safety-related structures. Ff. Tr. 3459, at 24-27. Mr. Pope who worked closely with Mr. Gallagher believed that Mr. Gallagher was sincere in his refusal — albeit wrong. Tr. 2863 (Pope). Applicant would have us infer that Mr. Gallagher refused to operate the Ross plant because it was hot, dusty and noisy compared to the isolated air-conditioned Erie-Strayer control room. With Mr. Pope's corroboration and our own observation of Mr. Gallagher, we take him at his word that his concern was safety.

[58] Contrary to Intervenors' Proposed Finding, Mr. Gallagher testified concerning the maintenance at the Erie-Strayer plant, not the Ross plant.

a schedule for the maintenance of the plant. Gallagher, ff. Tr. 3459, at 14-15. Here is another apparent inconsistency in Mr. Gallagher's testimony because in the very same context he stated:

> Q. Why did you and Mr. Pope maintain the plant?
>
> A. It was our feeling, as conscientious workers, that we wanted to keep the plant running in top form so that we would consistently make good concrete. If there was a failure that we noticed and we could not fix it right ourselves, we'd notify Blount that we had to get it repaired before we continue production. This way we could assure continuous concrete pours and continuous operation.
>
> Q. Were you and Don Pope the only Blount people who worked with the Erie-Strayer plant?
>
> A. Yes. If there was any maintenance to be done, it was done by me and Don Pope. If there was something we could not fix ourselves, Blount, at our request, would call in mechanics from Local 150.

*Id.* at 15.

D-184. Apparently the essence of the allegation is that maintenance wasn't *scheduled,* not that it wasn't performed. Mr. Gallagher's own testimony demonstrated that the operators themselves had the responsibility to maintain the plant. *Id.* at 2. He also stated that maintenance was in fact properly performed. Tr. 3486. The Board does not understand the significance of this allegation. The Staff's explanation is that Mr. Gallagher confused Mr. Andre's discussion with the NRC about calibration of the plant with plant maintenance. This is probably correct because Region III inspectors testified that the NRC monitors calibration of the batch plants, but does not exercise jurisdiction over routine equipment maintenance, which is left to the contractor. After interviewing Mr. Gallagher in November 1982, the Staff concluded that Mr. Gallagher's allegation was based on his misunderstanding of the scope of NRC inquiries with regard to maintenance. Region III Testimony, ff. Tr. 3586, at 24-25, and Attachment F.

D-185. Mr. Gallagher does not assert that the plant was not calibrated according to schedule. In fact he testified that it was calibrated on a schedule. Ff. Tr. 3459, at 6-7.

D-186. Again, we take Mr. Gallagher at his word and find that the Erie-Strayer plant was well maintained by Mr. Gallagher and Mr. Pope. The second allegation is not substantiated and is probably incorrect.

D-187. *Third Allegation: There was a recurring problem with aggregate containing an excessive amount of fines throughout Mr. Gallagher's employment with Blount until the aggregate pile was finally condemned in 1979.* Proposed Finding 56. Mr. Gallagher testified that from 1975 until the concrete aggregate pile was condemned in 1978 (actually 1979) he would observe muddy aggregate when it was brought in wet and dusty aggregate when it was brought in dry. This indicated to him that the aggregate contained too great a fraction of fine particles or "fines." In 1976, several years before the aggregate pile was condemned, an engineer from Israel was, according to Mr. Gallagher, amazed because the pile was rather dirty and not suited for safety-related concrete. Although Mr. Pope complained many times about the aggregate, Edison did nothing to remedy the problem. Consequently, about 100,000 yards of concrete was made with nonconforming aggregate according to Mr. Gallagher. Ff. Tr. 3459, at 11-14.

D-188. The NRC inspectors regarded the allegation to be substantiated in part. Excessive fines were identified in December 1975 and the aggregate was nevertheless used in safety-related structures based on an engineering evaluation by Sargent & Lundy. Region III Testimony, ff. Tr. 3586, at 23-25. This approval was founded on the chemistry of the aggregate and the predominance of limestone material. Tr. 3887-89 (Forney, Hayes). In March 1979 the aggregate pile was condemned, and the nonconforming portion segregated. Region III Testimony, ff. Tr. 3586, at 24. The nonconforming aggregate had failed a sieve test. Mihovilovich, ff. Tr. 2750, at 12. Mr. Gallagher does not assert that the nonconforming aggregate was used in making concrete after it was condemned.

D-189. Mr. Pope also testified about aggregate fines. As we noted above, Mr. Gallagher's allegation was predicated largely on the muddy and dusty appearance of the aggregate and upon Mr. Pope's concerns. As to the appearance of the aggregate, Mr. Pope states that even conforming wet or dry aggregate is, respectively, always muddy or dusty. Tr. 2871. Nevertheless, Mr. Pope acknowledged that he and Mr. Gallagher looked at the aggregate pile in 1975 and, as a result, Mr. Pope reported to quality assurance that he believed the pile had excessive fines. Pope, ff. Tr. 2833, at 17-18; Tr. 2871-72 (Pope). He requested cylinder test results.

D-190. A cylinder test or break test is where a cylinder is filled with concrete during the pour and then compression-fractured at set intervals after the pour. It is the ultimate test of the strength of the concrete. Pope, ff. Tr. 2833, at 12. As to the fines and aggregate, the cylinder test can reveal whether the aggregate is adequate by the way in which the cylinder of concrete breaks. If it breaks through the aggregate rock, the entire mixture is necessarily as strong as the aggregate and it is a good test. *Id.* at 18. Intervenors challenge Mr. Pope's expertise to explain the reliability of the cylinder fracture tests. Reply Finding at 24. The Board notes, however, that the use of the cylinder test, as described by Mr. Pope, is so fundamental to concrete testing that it is within his area of knowledge. Also, it is predicated on basic engineering concepts which we officially notice.

D-191. Mr. Pope testified that he was satisfied with the cylinder test results. Pope, ff. Tr. 2833, at 18. Mr. Pope also denies that the Israeli engineer commented on the quality of the aggregate. Since the conversation reported by Mr. Gallagher was between Mr. Pope and the Israeli, we accept Mr. Pope's memory of the event as the more accurate. Perhaps the Israeli was not understood correctly by Mr. Gallagher. Also, there were in fact excessive but chemically acceptable fines in 1975 and the comment, even if made, could have been in that context.

D-192. Messrs. Tallent and Johnson are employees of Pittsburgh Testing Laboratories (PTL). The Board and the parties questioned them extensively concerning PTL's concrete testing program and methods. Tr. 3960-87.

D-193. They testified that during the time that PTL had been at Byron, from September 1977 to date, less than one-half of one percent of the 5,500 concrete test cylinders have failed the compression testing to which they are subjected. Mr. Tallent noted that the cause of a cylinder test failure is not necessarily a problem with the concrete; cylinder failures can be attributable to improper testing techniques, such as molding or maintenance of the cylinders. Mr. Tallent further testified that, at a site such as Byron, where concrete is subjected to a variety of tests, and not merely cylinder tests, the percentage of cylinder failures attributable to improper testing is likely to be greater than at a site which does not have the control factor of other types of testing. Tr. 3968, 3978-79, 3982-84 (Tallent and Johnson).

D-194. A 200-mesh sieve test was performed by PTL on the aggregate daily in order to determine the percentage of fines in the aggregate. If a sample of aggregate demonstrated an excessive percentage of fines, two additional samples would be taken from the area of the aggregate pile where the initial sample was taken. If one of these two additional samples also failed the sieve test, that portion of the aggregate pile was condemned and was not used in the batching of concrete. Mr. Tallent testified that between September 1977 and June 1, 1979, the date Mr. Gallagher last worked at the site, the aggregate failed a sieve test only on three occasions, each in March 1979. The failed tests in March 1979 ultimately led to condemnation of the aggregate pile. Tr. 3962-65, 3980-81 (Tallent and Johnson).

D-195. In the final analysis, Mr. Gallagher's reasoning is that the aggregate had excessive fines *in* 1975 *and in* March 1979, *ergo,* there were excessive fines *from* 1975 *to* March 1979; that the reason he knows that the aggregate had excessive fines was that it looked dirty and because Mr. Pope said that it looked dirty. Confronted with Mr. Pope's testimony, the evidence concerning the cylinder break tests and the daily sieve tests, Mr. Gallagher persisted in his view that logical thinking still requires the conclusion that nonconforming aggregate was used from 1975 through March 1979 and as a result 100,000 yards of concrete containing nonconforming aggregate was used. Tr. 3476-84.

D-196. The strong preponderance of the evidence is that excessive fines appeared in the concrete aggregate on only two occasions, 1975 and 1979 and that, in each instance, the correct action was taken.

D-197. *Fourth Allegation: Edison and Blount had too few QA employees to adequately supervise the placement of concrete which at any given time might be taking place in as many as five different locations. As a result, production workers were able to add water to the concrete in excess of specifications without either recording it or it being discovered by QA personnel.* Proposed Findings 57-58. Concrete should leave the batch plants with the correct amount of ingredients including water. Water is one of the essential ingredients but a certain amount of water may be added to the mix to achieve the proper "slump" or placeability of the concrete. "Slump" derives its name from that property of wet, newly batched concrete which allows it to sink, settle, or slump when released from its cone-shaped test container. A slump test measures this phenomenon from a standardized test cone. Confusion can arise in discussing slump because a reference to, say, a "high slump" would probably mean that there was a large amount of slump demonstrated by the slump test when the concrete sample slumped to a lower level from its initial height. We use the term "high slump" to indicate a more fluid mixture with higher water content. A low slump, of course, is the opposite.

D-198. Workers could be motivated to add water to seek a higher slump in concrete so that it would be easier to place and to finish — for example, to vibrate the mixture into spaces among reinforcing bars. The amount of water must be carefully controlled because adding excess water weakens concrete below specifications. On the other hand the mixing and transport of concrete takes into consideration that some water could be lost before placement, for example, evaporation due to high chemical reaction temperatures. Therefore water may be added at the placement site to replace lost water. There is also a margin, or "trim" in the amount of water which may be added and remain within specifications. *See generally,* Gallagher, ff. Tr. 3459, at 17-22; Pope, ff. Tr. 2833, at 2, 21-22. Water added at the placement site was to be noted on the batch ticket, copies of which were returned to the batching plant, and to Blount, PTL, and to Applicant. Gallagher, ff. Tr. 3459, at 21; Tr. 3986 (Tallent and Johnson). Sometimes the truck drivers would radio back to the batching plant requesting that more water be added to the mix on the next batch. Gallagher, ff. Tr. 3459, at 21.

D-199. Contrary to Intervenors' version of Mr. Gallagher's allegation, in their Proposed Findings 57 and 58, a close reading of his prepared testimony indicates that he is not charging the Blount production workers with adding *excess* water to the concrete. The charge is that they added some water without recording it, especially when no one was around to check. Water was sometimes added for placeability when it was not necessary to increase the slump. *Id.* at 20-21. This point was pursued again on cross-examination, and Mr. Gallagher specifically testified:

> Q. . . . So let me return to my earlier question: that is, you talk about water being added at the site and not recorded. Are you claiming, Mr. Gallagher, that an excessive amount of water was ever added and not recorded or is your claim simply that water was added but not recorded.
>
> A. Water was added and not recorded.

Tr. 3492.

D-200. Nevertheless, the parties approached Mr. Gallagher's allegation with the concern that excessive water might have been added in the instances he alluded to.

D-201. The NRC's investigation of Mr. Gallagher's allegation is not dispositive because it depends upon the records of concrete testing. Intervenors now assert that testing samples could have been taken before water was added to the trucks.[59]

D-202. The Applicant meets the allegation with several points. Messrs. Pope, Tallent and Johnson testified that, contrary to Intervenors' allegations, PTL personnel were present at all pours. Tr. 3978 (Tallent and Johnson); Tr. 2879 (Pope). Changes in the mixing rates of the truck when water is added could be observed and heard and the appearance of the concrete would change. Tr. 3977-78 (Tallent and Johnson).

D-203. The Board has no reason not to accept the testimony of Messrs. Johnson and Tallent to the effect that there were sufficient PTL inspectors at placement sites to catch the addition of any excessive water. Whether the proper notations were placed on the batch tickets

[59] Messrs. Tallent and Johnson testified also that excess water added at the placement site would be detected by the slump test. Tr. 3966. But the slump test would not catch a concrete mix which was improperly diluted after the slump specimen was taken, as Mr. Gallagher noted. Tr. 3536.

cannot be resolved except by uncritically accepting Mr. Gallagher's testimony which was based upon unspecified conversations with truck drivers. Since he neither makes nor repeats any allegations that excess water was added, we do not believe that a safety problem is present. In any event, Mr. Johnson's testimony was reassuring. He stated that it was a very rare occurrence when water was added at the placement site. Out of 20 pouring days a month, maybe on 1 day they would have added water. Tr. 3966-67.

D-204. The Board finds that Intervenors' allegation that excess water was added to the concrete mix at the placement site is not substantiated — in fact, it is not even supported by Mr. Gallagher. The allegation that any water was added at the placement site without the proper documentation remains unresolved and unsubstantiated.

D-205. *Fifth Allegation: Even if QA/QC personnel did observe the unauthorized addition of excess water into the concrete, they did not have sufficient authority to overrule Blount's production supervisor who would order the workers to add the water anyway.* Proposed Finding 59. This allegation is based solely on a single statement in Mr. Gallagher's prepared testimony:

> Also, I know QA/QC workers didn't make much money, and if the QA/QC person disagreed with adding more water, it was easy for the Blount production supervisor to say "Put the water in there anyway."

Ff. Tr. 3459, at 22.

D-206. Intervenors cite no other support nor can we find any. The allegation is unsubstantiated.

D-207. *Sixth Allegation: For a period of at least one month, there was a problem with oil leakage from a faulty cement storage silo blower into the dry cement mixer and subsequently into the concrete mixture.* Proposed Finding 60. Intervenors also allege that although the problem was complained of repeatedly, "there is absolutely no evidence in the record that it was actually fixed." *Id.* The fact that there was an oil leak from the cement blower is acknowledged by Applicant. The second allegation, that there was no record evidence that it was ever repaired is simply a flat misstatement of the record. Mr. Pope testified that "Blount Brothers immediately took that blower off the line and sent it out and had it completely rebuilt." Ff. Tr. 2833, at 24.

D-208. The NRC inspectors made a thorough inspection and analysis of this allegation. They considered the type of blowers involved, the method and amount of lubrication and the cement ratios. They concluded that any gross leakage from the manually filled oil cups used to lubricate the blowers would cause excessive bearing heat and failure in a short time unless the oil was continuously added. Given the amount of cement and the small amount of oil involved, the unit contamination would be necessarily extremely low. Strength tests did not reveal any concrete below design values. The inspectors also noted documentation of two instances of repair work on the blowers, both for overhaul. Mr. Pope's recollection that the blower unit was rebuilt was corroborated by the NRC inspection. Region III Testimony, ff. Tr. 3586, Attachment F, at 8-9.

D-209. Mr. Gallagher was not very well informed on this matter and his allegation is incorrect.

D-210. *Seventh Allegation: It was made apparent to Mr. Gallagher that he was not supposed to talk to NRC investigators and, as a result, he "often sat quietly as the Blount QC people stretched the truth on a variety of quality control practices." (Gallagher, ff. Tr. 3459, Exhibit A, at 6).* Proposed Finding 61. The foregoing is an exact quote of Intervenors' seventh and final allegation based upon Mr. Gallagher's statements. It is the only reference to this particular perception of Mr. Gallagher in any of the Intervenors' proposed findings. The cited record support for the allegation in its entirety is:

> Also, in general, it was apparent to me that the Blount QC staff did not want the batch plant operators and the other workers talking to the NRC investigators. When NRC engineers spoke with Blount QC staff persons in my presence, I knew from the cold glares directed at me that I was not supposed to talk about safety topics being discussed, despite the fact that they often concerned matters about which I was quite familiar. I often sat quietly as the Blount QC people stretched the truth on a variety of quality control practices.

Gallagher, ff. Tr. 3459, Exhibit A, at 6.

D-211. It is not possible to investigate or to adjudicate this type of naked allegation. It has not been substantiated in this proceeding.

D-212. Out of seven allegations made by Mr. Gallagher — or more accurately, the allegations by Intervenors based upon Mr. Gallagher's statements — none was substantiated. Some of them were capable of reliable and objective factual resolution. For example, the Ross plant was capable of making safety-related concrete and Mr. Gallagher was simply wrong about the transport trucks. The cement blower was promptly fixed. His visual perception of the quality of the concrete aggregate was belied by daily sieve testing. In other instances his testimony was inconsistent, such as his statement that there was no regularly scheduled maintenance on the Erie-Strayer batch plant when his other testimony indicated that he and Mr. Pope continuously maintained the plant and freely called in experts when needed. His testimony that the

Ross plant was designed as a backup to the Erie-Strayer plant but not capable of producing safety-related concrete is internally inconsistent.

D-213. The Board is perplexed by the very high level of inaccuracies in his allegations. He seemed to be sincere in demeanor. He is clearly intelligent and articulate. He did not seem to be vengeful and he explained rather well generally how concrete is made. The Board cannot explain why he was so inaccurate, but we are satisfied that he was. The allegations have had a full airing. The Region III inspection report on his charges (ff. Tr. 3586, Attachment F) demonstrates a thorough professional inquiry which initially accepted at face value the validity of his allegations but could not substantiate them, with the possible exception that excessive fines in aggregate were identified in 1975. The Applicant brought to the hearing the appropriate people to explain and to be cross-examined on the allegations.

D-214. Mr. Gallagher has not demonstrated inadequacies in Blount's QA program or in Applicant's management of that program.

Mr. Stomfay-Stitz' Allegations

D-215. Mr. Peter Stomfay-Stitz worked at the Byron site from June 1978 through April 1979 for Blount Brothers. At age 18 it was his first full-time job after graduating from high school. Within 5 months he became a Quality Assurance/Quality Control Materials Controller trainee. Stomfay-Stitz, ff. Tr. 2939, at 1-3. He was certified as a Materials Receiving Controller in January 1979. Tr. 2950 (Stomfay-Stitz).

D-216. A Blount QA/QC Materials Controller is responsible for ensuring that materials and accompanying documents conformed to specifications. The Materials Controller documents his inspection of received materials on a checklist form referred to as Receiving and Inspection (R&I) Report. The Materials Controller is also responsible for ensuring that material storage areas meet certain requirements. Stomfay-Stitz, ff. Tr. 2939, at 7-8.

D-217. Mr. Richard Barnhart was responsible for training Mr. Stomfay-Stitz for the position of Materials Controller and was one of Mr. Stomfay-Stitz' supervisors. Mr. Barnhart currently is a project engineer for Blount. He has been employed by Blount in a variety of positions at the Byron site since July 1976. Mr. Barnhart immediately preceded Mr. Stomfay-Stitz in the position of Materials Controller at the Byron site. In addition to training Mr. Stomfay-Stitz for that position, Mr. Barnhart also instructed Mr. Stomfay-Stitz in "bolting-in," which involved reviewing bolted connections in structural steel which relates to one of Mr. Stomfay-Stitz' allegations. Mr. Stomfay-Stitz' training consisted of study, on-the-job training and testing. In addition Mr. Stomfay-Stitz testified that he performed some inspections under the supervision of Mr. Barnhart, who also testified. Stomfay-Stitz, ff. Tr. 2939, at 4-5.

D-218. Mr. John Mihovilovich has been the lead structural engineer for Commonwealth Edison at Byron since 1975 and he also testified. Mr. Mihovilovich holds an engineering degree and has been an engineer with the company for 30 years. Mr. Mihovilovich's responsibilities include ensuring that various structural contractors, including Blount Brothers, fulfill their contractual obligations. Mihovilovich, ff. Tr. 2750, at 1.

D-219. Pittsburgh Testing Laboratory (PTL) employees Marvin Tallent, Jr., and Joseph Johnson, who testified regarding Mr. Gallagher's allegations also addressed Mr. Stomfay-Stitz' charges.

D-220. Region III conducted an investigation into some of Mr. Stomfay-Stitz' allegations and testified. Region III Testimony, ff. Tr. 3586, Attachment H, at 20-28.

D-221. This phase of the proceeding has been difficult for the parties and the Board. Mr. Stomfay-Stitz was not a very good witness in his oral testimony, particularly during the first 2 days of it. Tr. 2931, *et seq.* It is understandable that his memory and perception of the events at Byron, some 4 years earlier, were uncertain, especially considering his lack of experience and rather short tenure as a Materials Controller. However, he was also uncertain about the events in the hearing room, frequently asking for questions to be reread. His answers were often unresponsive and some appeared to be evasive. *Id.* As he later explained, at the request of Intervenors' counsel, he had been very nervous during the first 2 days of testimony. The involvement of the Board, the magnitude of the hearing and the presence of the press and public were disconcerting. Tr. 3227. He conceded that his ability to understand and completely answer questions had been impaired. Tr. 3238. The Board observed Mr. Stomfay-Stitz' nervous stress. But, in addition, we believed that there was a strong element of wariness in his demeanor and testimony which may have contributed to some of his incomplete or evasive answers.

D-222. Mr. Stomfay-Stitz' testimony most often tended, of course, to support his written testimony, affidavit, and the position of the Intervenors. Often, however, his testimony on cross-examination tended to support the Applicant. Either way, we believe it would be imprudent to rely heavily on his factual testimony and his opinions, which are totally without expert support. We do not, however, question Mr. Stomfay-Stitz' sincerity or his genuine concern about safety at Byron. The Board sensed that he had been shocked at what was apparently his first experience with the relatively rough and tumble environment he perceived at the large and varied construction project.

D-223. Before moving to Intervenors' specific allegation regarding Blount's materials control and storage practices, we address a rather creative, albeit anomalous, litigation technique employed by Intervenors on this issue. Intervenors allude to Mr. Stomfay-Stitz' demeanor on the witness stand and concede that it raises questions about the accuracy of his recollections. This factor, however, we are told is not to be taken as a reflection on Intervenors' case. Instead, we are urged to find that "it reflects most seriously on the general level of competency of the QA/QC personnel at Edison and its contractors." Intervenors' Proposed Finding 66. In somewhat the same vein, Intervenors next propose to the Board that we find that Mr. Stomfay-Stitz himself (under orders) departed from stated Blount procedures and that this resulted in a completely unreliable and misleading set of QA/QC documentation records. Proposed Finding 67.

D-224. In sum, Intervenors contend that Blount hired incompetent personnel in QA/QC who produced unreliable and misleading QA documentation and that the proof of this, Exhibit A as it were, is Mr. Stomfay-Stitz himself — upon whom Intervenors totally rely for their case. His incredibility is offered as proof of his credibility.

D-225. *First Allegation: Within the broad allegation that Mr. Stomfay-Stitz departed from procedures and produced unreliable, misleading records, Intervenors allege that*

> miscellaneous steel items from Mid-City Architectural Iron that arrived without proper documentation were supposed to be either rejected or accepted and quarantined. However, Mr. Barnhart, in order to generate less paperwork, instructed Mr. Stomfay-Stitz to accept the items, call the manufacturer to request the documentation and, upon receiving the documentation, to date and fill out the inspection report as if the item and documentation had originally arrived intact.

Proposed Finding 67. Mr. Barnhart, addressing the allegation, testified:

> A. . . . The situation surrounding this allegation is such that Mid-City would deliver embed frames, miscellaneous steel items, to the job site weekly or biweekly. In a few instances, the documentation was found to not be complete. That is, documentation comprised of CMTRs, certificate of compliance, required documentation.
>
> I initiated, or my practice is, as you would say, or was, rather than filling out a quarantine tag or tags for steel affected by the missing documentation, I would simply have it off-loaded in a quarantined area, segregated area of the embed yard. I would notify engineering that we had a problem with these embeds and that they were to be unloaded in a segregated area. Before the truck was

unloaded, I would contact Mid-City, verify that the documentation was in hand — that is, confirm that the documentation could be supplied, establish that fact. I would ask them to put it in the mail, and I would have it the next day, which was generally the case.

The use of the tags was a bypass by me, was a manner of being expedient or a little more practical. I'm not saying it is correct. I believe it is wrong. But those are the facts.

Tr. 2808-09.

D-226. Mr. Stomfay-Stitz alleged that the embeds were not segregated and he feared that they had worked their way into the plant by the time the paper work was completed. However, he knew of no particular instance where that occurred. Mr. Stomfay-Stitz also stated that it was solely a traceability problem and the embeds themselves were checked for all physical problems. Tr. 3151-52 (Stomfay-Stitz). The Board accepts Mr. Barnhart's version of the practice as being the most logical and because Mr. Barnhart was better informed.

D-227. We find that the allegation is partially true to the extent that it was a departure from stated procedures. It is not true that the practice produced unreliable paper work or that it posed a risk that the embed material was defective.

D-228. *Second Allegation: Intervenors state:*

Second, Mr. Stomfay-Stitz was instructed by Mr. Rick Donica, Blount's QA/QC Control Manager at Byron, to accept concrete blocks that were wet and dirty from Eller and Wylie, a supplier of Category I materials, without recording the fact that these blocks were not properly protected. (Stomfay-Stitz Prepared Testimony at 14-15.)

Proposed Finding 68. This is an unfair and deceptive allegation, which is, by the way, Intervenors' allegation, not Mr. Stomfay-Stitz'. The allegation implies that nonconforming concrete blocks, received from the Category I supplier, were used in Category I construction. As Mr. Stomfay-Stitz' own testimony, cited by Intervenors, reveals, Mr. Donica actually exceeded acceptance requirements and would disqualify the wet and dirty blocks as Category I material. It was segregated to be used as Category II material. Thus there was no need to note the condition of the block. Mr. Stomfay-Stitz' concern was not whether documentation was correct but whether the blocks were actually used only in Category II construction, which is the thrust of Intervenors' Seventh Allegation, discussed below.

D-229. The second allegation is deceptive and irrelevant.

D-230. *Third Allegation:*

> . . . Mr. Stomfay-Stitz was also responsible for inspecting tendons that arrived on site. Upon being informed that the tendons would be reinspected by the manufacturer, the quality and thoroughness of Mr. Stomfay-Stitz' receiving inspections slacked off considerably, with the knowledge and acquiescence of Mr. Barnhart and Mr. Donica.

Proposed Finding 69. Mr. Stomfay-Stitz stood by his prepared testimony on cross-examination and explained further:

> Q. You say that's the way you were told to do it. In your testimony you refer to the acquiescence of Mr. Donica and Mr. Barnhart. Did they acquiesce or did they actually tell you to slack off?
>
> A. No, they did not.
>
> Q. They did not what?
>
> A. Tell me to slack off, but it was obvious.
>
> Q. You simply inferred that they knew you were slacking off?
>
> A. I'm sure they did.
>
> Q. Did you ever talk about it with them?
>
> A. I don't recall.
>
> Q. Are you aware of any specific instances in which material which did not conform to receiving specifications slipped through and was accepted because you slacked off?
>
> A. No, I did not [sic].

Tr. 3011.

D-231. Moreover, Mr. Stomfay-Stitz testified that he continued to check every item on the receiving and inspection checklist, and the respective documents do not reflect the fact that he slacked off. Tr. 3010-11.

D-232. This allegation of unspoken acquiescence is not substantiated.

D-233. *Fourth Allegation: Continuing the general allegation that Mr. Stomfay-Stitz departed from procedures, thus producing misleading and unreliable paper work, Intervenors allege:*

> Fourth, and most importantly, Mr. Stomfay-Stitz was ordered to fill out receiving and inspection reports for items which he did not see, and which he had no opportunity to inspect. (Stomfay-Stitz, Prepared Testimony, at 38-39; Barnhart, Tr. 2805-07).

Proposed Finding 70. Mr. Stomfay-Stitz testified that the situation described in his fourth allegation would arise when he was out in the plant on other inspections but Messrs. Donica or Barnhart, who are both authorized to do receiving inspections, would receive the materials. They would then direct Mr. Stomfay-Stitz to write up the report as if he had inspected the shipment personally. Stomfay-Stitz, ff. Tr. 2939, at 38-39.

D-234. Applicant's position on this allegation is somewhat inconsistent. On cross-examination of Mr. Stomfay-Stitz, Applicant's counsel tried to establish that Messrs. Donica and Barnhart had actually inspected the materials, and Mr. Stomfay-Stitz would be correct in signing the receipt and inspection forms on the belief that the materials had been inspected correctly. Any doubts could be resolved by Mr. Stomfay-Stitz' going to the materials after receipt and doing the inspection. Tr. 3053-56.

D-235. Mr. Barnhart had previously testified, however, that when Mr. Stomfay-Stitz was absent at the time material arrived on site he would be told when he returned that "the documentation had been looked at and then we had taken a look at the materials." Mr. Stomfay-Stitz was then told to go to the materials and look for himself, then do the documentation. Tr. 2805-06.

D-236. The inconsistency, as it appears to us, is that under counsel's version on cross-examination, the actual inspection would take place on receipt with Mr. Stomfay-Stitz resolving doubts by an additional inspection. In Mr. Barnhart's version, the actual inspection would always be done, as pertinent, by Mr. Stomfay-Stitz.

D-237. In either event, however, Mr. Stomfay-Stitz testified that sometimes the material, perhaps fungibles such as block and sand, would already be unloaded and indistinguishable. Tr. 3056.

D-238. This allegation cannot be resolved by documentation because the very essence of the charge is that the documents incorrectly reflect Mr. Stomfay-Stitz' own inspection. While we are reluctant to make a finding based upon Mr. Stomfay-Stitz' unsubstantiated testimony, in this instance, the allegation cannot be lightly dismissed. His version seems more logical. Mr. Barnhart's version is incomplete. There is no dispute that either of two senior inspectors had seen the material and documentation upon arrival. Mr. Barnhart did not allude to a holding area for incoming materials pending receipt inspection — something akin to a temporary quarantine area. We believe that there was none or it would have come up in the testimony. To off-load pending inspection, then reload for ultimate storage or use would be an inefficient practice. Therefore the materials yet to be inspected by Mr. Stomfay-Stitz were necessarily somewhere in a storage area pending use or were being used.

We have no evidence to illustrate that all such materials were still capable of segregation and inspection. Mr. Stomfay-Stitz' reference to his inability to inspect fungible materials after unloading rings true.

D-239. The allegation is probably true. However, it is also probably true that the senior inspectors would not have permitted the practice with respect to materials where more detailed inspection, compared to their own overall observation, was required. Tr. 3058 (Stomfay-Stitz).

D-240. *Fifth Allegation: The fifth allegation is the first in a series pertaining to Blount's allegedly deficient QA program for overseeing the storage of safety-related items. Intervenors state that the Blount procedure required ten daily surveillances of tendon storage barns, but Mr. Barnhart instructed Mr. Stomfay-Stitz to perform the surveillances on a weekly basis.* Proposed Finding 70. A tendon, also known as a "post-tensioning" tendon, is a cable installed in containment buildings to provide additional strength. The buttonhead, discussed in a following allegation, is a small steel anchoring knob at the end of the cable. The tendons arrived on site rolled around wooden beams and sealed in plastic. Upon arrival at the site the bundles were opened and the cables were inspected for dust, rust, or nicked or bent wire. Tendons were then stored in warehouses on site; each tendon was greased and placed under two sheets of protective plastic. Stomfay-Stitz, ff. Tr. 2939, at 16-17, 21; Tr. 3029 (Stomfay-Stitz).

D-241. The work procedure did in fact call for daily inspections of tendons. However, the work procedure was qualified by the architect-engineer to provide that a weekly inspection would be adequate if the storage barn was ventilated by fans. Tr. 2809-10 (Barnhart); Tr. 2787-88 (Mihovilovich).

D-242. However, the evidentiary record cited to the Board by the parties and otherwise reviewed by it, does not permit a reliable finding as to whether the barn had ventilating fans. Mr. Mihovilovich stated that it had fans as far as he knew, but that he has no way of really knowing. *Id.* Mr. Barnhart apparently was aware of the qualification permitting weekly inspections only because of Mr. Mihovilovich's mention of it and acknowledged that the work procedure called for daily procedures. Without further explanation, Mr. Barnhart acknowledged that he had nevertheless instructed Mr. Stomfay-Stitz to inspect weekly. Tr. 2809-10. Were it not for the wet and muddy conditions of the tendon storage barns discussed in connection with the next allegation, the most reasonable inference would be that the weekly surveillances were correctly predicated on the architect-engineer's qualification because that seems to be the unspoken premise of Mr. Barnhart's testimony. However, we infer from the entire record that the allegation is correct. Mr. Stomfay-Stitz was improperly instructed to survey the tendon barns weekly instead of daily. We do not, however, arrive at this conclusion with a great deal of confidence. The Applicant should have addressed the allegation more completely.

D-243. *Sixth Allegation:*

> . . . in his surveillances of tendon storage barns, Mr. Stomfay-Stitz found that conditions were unacceptable because of the presence of mud and water and deficiencies in security, aisle spacing and accessibility.

Proposed Finding 72. Mr. Stomfay-Stitz alleges further that Mr. Donica ordered him to report that the storage conditions were acceptable in order to save paper work. Stomfay-Stitz, ff. Tr. 2939, at 18. Intervenors also allege that Mr. Stomfay-Stitz falsely and knowingly noted acceptable storage conditions in fear of losing his job, citing Tr. 3032. The cited testimony, although suggesting that conclusion, is not quite as definite as Intervenors assert. Mr. Stomfay-Stitz testified rather illogically:

> Q. And [Donica and Barnhart] disagreed with your opinion, didn't they?
>
> A. It was the general consensus that they were going to be all moved around and jostled around and back and forth to fuel handling, or wherever they were going to be performed, future inspections, so that it would have been a waste of time and effort.
>
> Q. Did you agree with that consensus?
>
> A. At the time I did for fear of my job.
>
> JUDGE SMITH: Wait a minute. At the time you did, and then what was the balance of your answer?
>
> THE WITNESS: In fear of my job.
>
> JUDGE SMITH: You believed that in fear of your job?
>
> THE WITNESS: Sure.

*Id.*

D-244. Moreover, Mr. Stomfay-Stitz conceded that he was not threatened with firing in so many words; that his supervisors disagreed with the factual premise of his observations in that they believed the tendons were adequately spaced and accessible; and that the disagreement was a professional one. Tr. 3033-34 (Stomfay-Stitz).

D-245. After Mr. Stomfay-Stitz left Blount's employ, rust was discovered on some of the tendons. Intervenors point to this circumstance as evidence that the storage conditions were faulty. The rust occurred because protective grease had been rubbed off next to posts where inspection could not detect it. Many had to be replaced. Mihovilovich, ff. Tr. 2750, at 8-9. Contrary to Intervenors' citation, Region III did not attribute the rust to storage conditions. Although the storage barn was wet and muddy, the Region III inspector, Mr. Konklin, attributes the rust to the wearing through of the protective covering and grease and stated that rust would have occurred from normal humidity. Tr. 3734-35.

D-246. Contrary to Applicant's assertion (Proposed Finding 655), however, the Board cannot find record support for the assertion that the NRC had regularly examined storage conditions and found them to be satisfactory.[60]

D-247. Although Region III's inspection report concluded that Mr. Stomfay-Stitz' allegations concerning tendon storage conditions was not substantiated, the conclusion was based in large part on Mr. Stomfay-Stitz' own inspection report which he now disavows. Region III Testimony, ff. Tr. 3586, Attachment E, at 11-12.

D-248. Mr. Konklin's testimony that rust would have occurred in any event in natural humidity is logical. In this instance, however, the fact that the barn was wet and muddy is not disputed and this condition suggests that the building was not ventilated with fans and that humidity was not normal. In the Board's view it was incumbent upon the Applicant to come forward with a better evidentiary showing on the allegations regarding tendon storage. Accordingly, we find that the allegation is substantiated.

D-249. *Seventh Allegation: Category I (for safety-related use) and Category II (not for safety-related use) concrete blocks were not stored properly at Byron. Therefore neither Applicant nor Blount can provide assurance that only Category I blocks have been used for safety-related constructions.* Proposed Finding 73. The basis for this allegation is that, when blocks were limited to Category II use, they were segregated and marked by yellow tape, but that sometimes they were segregated near Category I construction activity. Mr. Stomfay-Stitz was concerned that, because it was more accessible, construction workers, either ignoring or failing to understand the segregation symbol, yellow tape, would use Category II

[60] Applicant's reference in Exhibit 5 to Mr. Mihovilovich's testimony, ff. Tr. 2750, is too general to be helpful. Support may be there, but after devoting considerable time looking for it, we gave up. This was a recurring problem in Applicant's proposed findings and exhibits. As Mr. Mihovilovich's testimony demonstrates, multiple exhibits were attached back to back, some of which were rather lengthy. It is difficult to determine where one ends and another begins because the pages are not numbered for that purpose. Also, frequently Applicant's counsel's references to such exhibits are without exhibit page numbers. This has caused a significant amount of time to be wasted by the Board.

block on Category I construction. Stomfay-Stitz, ff. Tr. 2939, at 15-16. Applicant points out that this allegation rests on Mr. Stomfay-Stitz' inherent distrust of onsite construction workers. Tr. 3020-22 (Stomfay-Stitz).

D-250. The Board concludes that this allegation lacks substance. Not only was Category II block designated by yellow tape, it was also identifiable because of its dirty or wet appearance. Stomfay-Stitz, ff. Tr. 2939, at 15. Although apprentice construction workers might not understand the significance of yellow tape, or the difference between safety-related and nonsafety-related, we assume that safety-related work had competent and experienced supervision. This allegation provides insight into what the Board believes to be an important basis for Mr. Stomfay-Stitz' allegations concerning QA/QC at Byron. He tended to be pessimistic without stated reasons. His concern that construction workers could be relied upon to do the wrong thing pervaded his testimony. This essential distrust might be a welcome attribute in quality assurance workers such as Mr. Stomfay-Stitz, but the Board is mindful that Mr. Stomfay-Stitz did not know much about constructing nuclear power plants when he worked at Byron. Another example of his inherent distrust of construction workers can be seen in his Thirteenth Allegation concerning the possible use of nonconforming concrete aggregate, discussed below.

D-251. *Eighth Allegation: Inspections of slotted and fixed-bolt connections were performed improperly, without a schedule and apparently at random when time permitted. Instead of documenting a missing bolt, Mr. Stomfay-Stitz was instructed simply to have a worker replace it on the spot. In many cases he would not inspect bolts that required inspections.* Proposed Finding 74. Mr. Barnhart explained that Mr. Stomfay-Stitz was not inspecting for acceptance purposes. His examinations of the bolting (only slotted connections according to Mr. Barnhart) were for the purpose of spot-checking ongoing work to identify problems then arising in the work. Barnhart, ff. Tr. 2797, at 4. If so, it would not be inappropriate to point out missing bolts to the workers without additional documentation.

D-252. This allegation is related to another allegation, discussed below as Intervenors' Fifteenth Allegation pertaining to the adequacy of Mr. Stomfay-Stitz' training. Region III, investigating the qualifications of the "bolting-in" inspectors, could not determine from documentation whether the examinations performed by Messrs. Barnhart and Stomfay-Stitz were final acceptance inspections or whether they were simply surveillances of ongoing work. At that time Mr. Barnhart himself was not certified to inspect structural steel bolting. Tr. 3725-28 (Hayes); *see also* Paragraph D-284, *infra*.

D-253. Mr. Hayes who inspected this allegation was uncharacteristically ambiguous on this point. He stated first that neither Mr. Barnhart nor Mr. Stomfay-Stitz were doing bolting-in inspections in 1979 (Tr. 3725) but later stated that he could not tell from the records whether Mr. Stomfay-Stitz' examinations entitled "surveillance inspections" were acceptance inspections or for discovery (Tr. 3727).

D-254. The Board accepts the testimony of Mr. Barnhart that the "bolting-in" inspections were a part of Mr. Stomfay-Stitz' training and were not intended to be acceptance inspections, and were designed to spot problems in ongoing work. Mr. Hayes' testimony is not inconsistent.

D-255. However, surveying for missing bolts during ongoing work seems to be a task more related to construction than to quality assurance, particularly in view of the fact that the procedure also called for seeing to the placement of the bolts without documentation. This point was not pursued. It may be a practical approach but it suggests a vagueness in the distinction, or a blending as it were, of production and quality assurance, and it suggests weak independence in quality assurance.

D-256. Intervenors also imply that this allegation is substantiated because, at the time of the hearing, all of Blount's structural steel bolting was being reinspected. Proposed Finding 74. Intervenors misstate the record. Mr. Shewski did not testify that *all* structural steel bolting was being reinspected. Tr. 2382. Mr. Barnhart testified that slotted bolting inspected by Mr. Stomfay-Stitz was being replaced and all slotted bolting was being reinspected because of design changes. Barnhart, ff. Tr. 2797, at 5; Tr. 2814.

D-257. This allegation is not substantiated. The evidence as stated, however, indicates a lack of independence of the quality assurance function at Blount.

D-258. *Ninth Allegation: The inspection for cracks in (tendon) buttonheads was conducted in a careless unprofessional manner, and indeed Edison did not produce any documentation of that inspection.* Proposed Finding 75. Region III found that tendons arrived at Byron with buttonhead cracks; that they had left the manufacturer, INRYCO, that way, which indicated a weakness in INRYCO's shop. Tr. 3740 (Hayes).

D-259. Mr. Stomfay-Stitz testified that, when Blount attempted to reinspect for cracked buttonheads, a complete inspection could not be performed because some of the buttonheads were not accessible for inspection. Ff. Tr. 2939, at 22-23.

D-260. Applicant disputes the allegation as it relates to the quality of Blount's inspection. Mr. Mihovilovich testified that the problem was first discovered at LaSalle and as a result Blount was asked to perform an *informal* review of tendon buttonheads at Byron. A similar problem was found to exist at Byron. Actual inspection was then performed by the manufacturer. Ff. Tr. 2750, at 10-11; Exhibit 7. Applicant is correct that Mr. Mihovilovich's testimony indicates that the informal review by Blount would look to the general extent of the problem and not to the condition of each buttonhead. To this extent Mr. Stomfay-Stitz' allegation is without foundation.

D-261. However, Mr. Mihovilovich's testimony does not explain why the buttonhead cracks were not found on the initial acceptance inspection at the time of original delivery to Byron. We make no finding, however, because the record is unclear. Because of the extensive preservative coating and the wrapping around the tendons, the fact that they were rolled around cores, and the fact that the containment was probably not then ready for post-tensioning, final acceptance inspection may not yet have been performed. On the other hand Mr. Stomfay-Stitz' testimony, as we noted in connection with his allegation, indicated that some type of gross acceptance inspection was being performed pending INRYCO's reinspection. Ff. Tr. 2939, at 16-17, 20-21. No party raised this issue; but the doubt is somewhat unsettling.

D-262. Applicant takes strong exception to Intervenors' statement that records generated from the inspection of the buttonheads "have been destroyed or otherwise rendered unavailable." Intervenors' Proposed Finding 75. Intervenors offer no support for this serious implication. It is therefore irresponsible.

D-263. The Board concludes that the informal review of the buttonheads at Byron for cracks was not an acceptance inspection and, considering its purpose, it was not careless or unprofessional.

D-264. *Tenth Allegation: Mr. Stomfay-Stitz inspected for the location of certain structural beams to determine whether they match the design drawing locations. But when finding that a beam was missing he would call the architect-engineer, Sargent & Lundy, who would simply delete the beam from the design and generate paper work to cover the decision that the beam was not needed.* Proposed Finding 76. According to Mr. Stomfay-Stitz, Sargent & Lundy "would either say the beam was either not needed or that it would be changed to a Category II."[61] Then he makes the most grave accusation — an item would be changed from Category I to II

[61] This *non sequitur*, or perhaps incomplete thought, demonstrates the difficulty the Board has had in its attempts to make a careful analysis of Mr. Stomfay-Stitz' claims. The concept "not needed" should not be associated with the concept "change to Category II" by the disjunctive "or" in the context of the allegation. Presumably a missing structural beam is either needed or it is not.

(safety-related to nonsafety-related) because it was too difficult or expensive to replace it. Stomfay-Stitz, ff. Tr. 2939, at 41.

D-265. Mr. Stomfay-Stitz' testimony on cross-examination was totally without value except to discredit his written allegation. He could recall only one such occasion clearly. It was within Containment 2 but he did not know where. Tr. 3191-93. He didn't know whether the beam was missing inadvertently or consciously. He didn't know what type of analysis Sargent & Lundy did, if any. Tr. 3199. His sole support for the allegation on recollection of the incident is that, from his end of the process, he perceived the change too easy to make. *E.g.,* Tr. 3195 (Stomfay-Stitz).

D-266. The NRC Staff investigated the allegation and testified:

> Q. Did you go back and investigate this allegation?
>
> A. At the time, as you will recall, Mr. Stomfay-Stitz was unable to give us very much information. No specific information was contained in his affidavit. After discussing or interviewing him on — I believe on January the 29th, was it, whatever date it was, we went back and, during the time frame that he was employed by Blount Brothers, and in the capacity of inspecting structural steel, we looked at all the field change requests. We looked at all the telephone memos. We looked at all the correspondence files within Blount Brothers. We went down to Sargent & Lundy and did the same thing down there. We asked the Commonwealth Edison Company and Sargent & Lundy to compile a list of any of the people that had anything to do with structural steel at that time, and they did provide us such a list. We balanced that list against the field change notices or field change requests and the ECNs, and things that had been issued in that time frame, to make sure that they didn't miss one of those names, and we didn't find any of the conflicts. All of the names that they had given us, there was no different names that appeared on any of those records. We could not find any record of any design change that related to any telephone call by Mr. Stomfay-Stitz in regard to a missing structural steel member.
>
> We spent many hours trying to run this thing down.

Tr. 3742-43 (Hayes).

D-267. The Board concludes that the allegation has no foundation. It was irresponsibly made, and Intervenors were irresponsible for pursuing it in their proposed findings in view of the testimony at the hearing. Mr. Stomfay-Stitz' statement that such changes were made because of difficulty or expense is beyond his knowledge, as Intervenors well know. The allegation is one of those that seriously eroded Mr. Stomfay-Stitz' credibility and diminished Intervenors' credibility on the QA/QC issue. Moreover it wasted hours of Region III time which could have been devoted to safety.

D-268. *Eleventh Allegation: Mr. Stomfay-Stitz stated that the procedure to identify and separate bad aggregate was woefully inadequate.* Proposed Finding 77. Mr. Stomfay-Stitz was disturbed by what he perceived to be Pittsburgh Testing Laboratory's inability to exactly pinpoint the place from where failing aggregate gradation test samples were taken. He had no confidence in the PTL inspectors. Stomfay-Stitz, ff. Tr. 2939, at 23-28. We will not dwell long on this particular allegation because it is clear that Mr. Stomfay-Stitz knew little about aggregate testing.

D-269. Messrs. Tallent and Johnson acknowledged that it was not possible to pinpoint the exact spot from which a failed sample was taken, but their explanation was that five different samples were taken from the face of the aggregate pile being worked. The reliability of PTL aggregate testing depended upon the large number of localized tests. Tr. 3974-75 (Tallent and Johnson).

D-270. *Twelfth Allegation: In response to the problem where aggregate failed the sieve test (too many fines), Sargent & Lundy merely changed the specifications to increase the allowable amount of the fines.* Proposed Finding 77. Intervenors cite the testimony of Region III's Mr. Hayes for this allegation, Tr. 3774-75. The allegation, apparently based on the 1975 episode, lacks credence. It is not even a half truth. Mr. Hayes did not testify that Sargent & Lundy changed the specifications, but they determined, on an engineering evaluation, that the chemical composition of the fines that exceeded normal specification were not detrimental. *Id.* As we noted in connection with Mr. Gallagher's allegation, the excessive fines were predominately composed of limestone and the handling was found acceptable by Region III. Tr. 3887-89 (Forney, Hayes); *see also* Region III Testimony, ff. Tr. 3586, Attachment F, at 4-5.

D-271. *Thirteenth Allegation: Although Mr. Stomfay-Stitz had the responsibility to isolate the source of the failed aggregate samples on the pile, he was never given any guidelines to follow on the size of the area. No precautions were taken to assure that isolated aggregate was not used or that the isolated aggregate corresponded to the tested sample. The isolated aggregate was actually used in construction or was covered by new aggregate.* Proposed Finding 78. Intervenors cast this allegation as if it were a continuing situation. In fact, as we noted before, there has been only one instance in which the coarse aggregate pile was condemned, March 1979. In the weeks preceding the condemnation, a portion had been segregated because it had failed a gradation test. The allegation apparently refers to those weeks. Stomfay-Stitz, ff. Tr. 2939, at 23-25.

D-272. Mr. Stomfay-Stitz had no responsibility for testing aggregate and did not know anything about testing it. His sole responsibility was to isolate the nonconforming aggregate from the aggregate being used. He did not even inspect the aggregate. Stomfay-Stitz, ff. Tr. 2939, at 23-24. He attached documents, contemporary working memoranda, to his testimony to demonstrate his concern. *Id.*, Exhibit H.

D-273. His first complaint, that he was never given guidelines concerning the size of the area to be isolated, was never resolved in the hearing. But, during the time that a portion of the aggregate pile was nonconforming, his instructions were apparently quite conservative because, as his own memoranda indicate, the entire eastern face of the coarse aggregate pile was placed on hold marked with signs and yellow safety tape. *Id.*, Exhibit H.

D-274. Mr. Stomfay-Stitz also reported at the time that the front-end loader operators had been instructed to use only the south face of the pile which at the time had been found to be acceptable. Moreover, he reported that he would periodically check to see that the aggregate was being taken from the south side and that the failing aggregate would be used for backfill only. His reports of subsequent checks of the aggregate pile indicated that he had in fact made periodic (daily) checks, had found the yellow safety tape and signs in place and observed the front-end loader operators correctly using only aggregate from the south face. *Id.*

D-275. By March 29, 1979, PTL failed the entire pile on gradation testing and it was placed on hold. Acceptable coarse aggregate was then brought on site to batch Category I concrete and Mr. Stomfay-Stitz observed operators using the acceptable stockpile. No Category I construction was in progress during the time the pile was segregated. *Id.*

D-276. On cross-examination, Mr. Stomfay-Stitz, after he was admonished not to be evasive, confirmed his own belief in the accuracy of the memoranda he had prepared at the time (Exhibit H). Tr. 3123-44. It turned out that the only basis underlying his concern and the foundation of this allegation is that he did not have confidence that the construction workers would observe the segregation signs and yellow tape when he was not there watching them (*e.g.*, Tr. 3122, 3124, 3141), in addition to his basic distrust of PTL's aggregate testing procedures.

D-277. The allegation has no substance.

D-278. *Fourteenth Allegation: Intervenors make a series of allegations relating to Blount's organization and personnel. The first of these alleges that, contrary to the formal organization chart, Blount's QA/QC office at Byron was not independent but was actually controlled by production.* Proposed Finding 80.[62] Intervenors pick up on Mr. Stomfay-Stitz' allegation that decisions regarding hiring, overtime and pay increases for QA/QC personnel were made by the production managers who repeatedly denied him overtime and pay increases. In particular, Mr. Stomfay-Stitz stated that his supervisor, Mr. Donica, would go to the offices of the production managers to request overtime and pay increases for Mr. Stomfay-Stitz, thus indicating production control over quality assurance. The Board was dissatisfied with Applicant's presentation on this issue. The allegation was explicit and made known before the hearing.

D-279. While we agree with both Applicant and Region III that, if Mr. Donica did, in fact, consult with production personnel on overtime, it could be an appropriate discussion because obviously there must be coordination so that QA personnel would be present when production personnel were. However, this would not address Mr. Stomfay-Stitz' other allegation, that overtime would be needed because QA/QC personnel were overworked, which is the subject of his Sixteenth Allegation below.

D-280. The only support for the allegation concerning the pay increase and overtime rests on Mr. Stomfay-Stitz' own testimony concerning what Mr. Donica told him. In view of the hearsay nature of Mr. Donica's statement and the general vagueness of Mr. Stomfay-Stitz' testimony, we are reluctant to conclude that production managers did control QA wages. Yet it should have been possible to present exact evidence on this allegation, as we indicated during the hearing, but none was presented. *See* Board's comments at Tr. 2939, 3753-56. Although the NRC Staff investigated the general allegation, it did not inquire into the statement that production controlled QA's wages. Despite the general unreliability of Mr. Stomfay-Stitz' overall testimony, we believe that this allegation has a ring of truth and we believe him. He was paid $4.00 per hour by Blount which, considering general construction worker wages in 1979 and the importance of the job, seems to us to be rather paltry. It is a manifestation of a low regard for the QA function by Blount. When Mr. Stomfay-Stitz voluntarily left Blount's employ in the Spring of 1979 for another contractor at Byron, he was hired at $11.00 an hour.

D-281. We find that Mr. Stomfay-Stitz' allegation that Blount production managers at Byron controlled his wages and overtime is probably

[62] Although Intervenors' Proposed Finding 79 is set out as a separate allegation, it is a series of accusations and implications that were embodied in other allegations. Thus we skip to the allegation of Proposed Finding 80.

true, and is probably true with respect to other QA personnel. This was an inappropriate interference with the independence of the QA function.

D-282. *Fifteenth Allegation: Blount provided training for QA personnel that was inadequate in terms of its length and quality.* Proposed Finding 81. This allegation subsumes a series of related charges. The first concerns the asserted unreliability of Blount's document *vis-a-vis* Mr. Stomfay-Stitz' testimony with respect to the amount of training he received. It depends solely upon Mr. Stomfay-Stitz' memory and his testimony concerning his confusion, without any analysis of why the respective Blount documents would be unreliable. That charge cannot be substantiated. Another charge which cannot be resolved by the record is that Mr. Stomfay-Stitz, as a trainee, was infrequently (5 percent of the time) accompanied by Mr. Barnhart when the former performed inspections during on-the-job training.

D-283. Yet another charge has Mr. Barnhart conceding that he instructed Mr. Stomfay-Stitz to perform his receiving, storage, and bolting inspections in a manner flatly in conflict with company procedures. The cited testimony by Mr. Barnhart (Tr. 2808-09) refers not at all to bolting. His testimony regarding improper procedures was a reference to the practical approach used by him in handling late-arriving documentation for materials arriving earlier at Byron and is a restatement of an earlier allegation resolved by the Board above in connection with embeds from Mid-City.

D-284. There is, however, one charge in this broad allegation which requires a thorough examination. Intervenors allege that Mr. Barnhart, who trained Mr. Stomfay-Stitz for "bolting-in" inspections was not himself certified to perform those inspections. This allegation is substantiated.

D-285. Mr. Barnhart testified that he instructed Mr. Stomfay-Stitz "in the inspection process known as 'bolting in'." Barnhart, ff. Tr. 2797, at 3. At the time of the events referred to by Mr. Stomfay-Stitz, however, Mr. Barnhart was not certified to do that inspection and this was discovered by Region III inspectors in the process of investigating Mr. Stomfay-Stitz' allegations. Region III Testimony, ff. Tr. 3586, Attachment H; Tr. 3725-28.

D-286. Applicant urges the Board to distinguish between Mr. Barnhart's status as "certified" compared to his qualifications to do the limited work involved in surveying the bolting as discussed with respect to Intervenors' Eighth Allegation, *supra.* As the Board discusses in greater detail below, there is a difference between meeting the formal ANSI certification requirements for QA inspectors and being functionally qualified to perform the inspections. Inspecting for slotted connections was simple. Such connections have a slotted rather than a round hole for the bolt so that the structural members can move. The inspector need only determine that each slotted hole had a bolt and a nut, that the nut was finger tight and that the bolt was burred to prevent the nut from loosening. Barnhart, ff. Tr. 2797, at 3.

D-287. The allegation that Mr. Barnhart, not being certified to the task, should not have been training Mr. Stomfay-Stitz in formal acceptance inspections for structural connections is substantiated. It is not a very important matter, however, because Mr. Barnhart was qualified to train Mr. Stomfay-Stitz to do this simple job. Moreover the issue never really materialized because Mr. Stomfay-Stitz was never called upon to perform formal acceptance bolting-in inspections.

D-288. The allegation concerning the adequacy of Mr. Stomfay-Stitz' training possibly captures the basic tension between Mr. Stomfay-Stitz and his employer, and perhaps explains the underlying reasons for his general concern. Intervenors propose that "his testimony (*passim*) that he was often confused as to how to go about the performance of his duties" is evidence itself that he was poorly trained. Proposed Finding 81. Yet, as the evidence unfolded as to many of Mr. Stomfay-Stitz' allegations, it became evident that the tasks assigned to him as a QA materials controller, and later as a trainee for inspecting structural steel connections, were not difficult to perform and they did not require a high level of training. He served as a materials controller apprentice for five months, which seems long enough to master that job. We cannot find therefore that he received inadequate training for his designated duties.

D-289. But Mr. Stomfay-Stitz apparently was not content to simply perform his duties and to trust others to perform theirs. He perceived that where opportunities existed for others to fail, they would fail. But trained and experienced in only a narrow aspect of plant construction, he lacked the knowledge and information to understand the significance of his observations. If there was a failure by Blount in training him, it was probably in not imparting an adequate understanding as to how his job related to other jobs.

D-290. Mr. Stomfay-Stitz' allegations that his training as a QA materials controller was inadequate is without substantiation.

D-291. *Sixteenth Allegation: As a result of lack of QA/QC independence, QA/QC personnel were severely overworked.* Proposed Finding 82. This allegation depends largely upon Mr. Stomfay-Stitz' perception of his duties and was not directly addressed by either Applicant or Staff witnesses. We do not fault this lack of response, however. Although the allegation was made known early in the proceeding as a part of Mr.

Stomfay-Stitz' affidavit, it is a very subjective judgment on his part, not easily refuted.

D-292. However, the Board received the overall impression that Mr. Stomfay-Stitz was very busy while employed at Blount — perhaps too busy. For example, it is not disputed that it was necessary for Messrs. Donica or Barnhart to initially receive shipments within Mr. Stomfay-Stitz' responsibility to receive because he would be elsewhere in the plant — doing bolting-in surveillances, or checking the aggregate pile.

D-293. While we observed with respect to the training allegation that the job involved simple tasks, it was nevertheless important and required responsible personnel. In that light, we noted with concern that Mr. Stomfay-Stitz was paid a rather meager $4.00 an hour as a journeyman materials controller and left for higher wages, $11.00, after a rather short tenure. Mr. Stomfay-Stitz and Mr. Barnhart had to share Mr. Herbing's duties, when Herbing suddenly left. Stomfay-Stitz, ff. Tr. 2939, at 34. Because of our lack of confidence in Mr. Stomfay-Stitz' account of the events at Byron we cannot, and do not find, by a preponderance of the reliable, probative and substantial evidence, that the QA personnel at Blount were overworked. But there are some troubling indications that they were overworked, and we leave the issue with the nagging concern that the allegation is true.

D-294. *Seventeenth Allegation: Blount did nothing to encourage workers to come forward with evidence of wrongdoing.* Proposed Finding 83. Intervenors cite the testimony of Mr. Stomfay-Stitz and Mr. Gallagher (Tr. 3540) as support for this allegation. Mr. Gallagher made a similar allegation with respect to reporting wrongdoing to the NRC, and the Board found that it was not possible to investigate or adjudicate that type of naked allegation. [*See* Seventh Allegation.]

D-295. In the testimony attributed to Mr. Stomfay-Stitz, he did *not* state, as Intervenors assert, that he was discouraged from coming forward with information. He stated that he did his job the way he was told to do it because he was afraid of being fired if he did not — perhaps the same concept, but perhaps not. Ff. Tr. 2939, at 43. In any event Blount had no control over Mr. Stomfay-Stitz after he left their employ in April 1979 and it was not until 3 years later that he came forward with his version of the events at Blount.

D-296. Mr. Hayes of Region III testified that he interviewed QA inspectors at Blount and he believed that they had sufficient authority to identify QA problems and to come forward to management with their findings. Tr. 3744, 3756.

D-297. The allegation cannot be supported with Mr. Stomfay-Stitz' and Mr. Gallagher's testimony alone. The best evidence, but not totally reliable, is the inquiry by Region III which uncovered no information concerning workers being discouraged in coming forward with evidence of wrongdoing. The allegation is not substantiated.

Conclusions — Blount Brothers

D-298. We have concluded above that Mr. Gallagher's allegations have not demonstrated inadequacies in Blount's QA program or in Applicant's management of that program. Mr. Stomfay-Stitz' allegations are not so easily resolved, however. He was not a convincing witness but some of his allegations are substantiated when the entire respective record is evaluated. Now the Board must analyze the significance of these findings.

D-299. There was probably too much laxity in the inspection procedure involving the receipt of fungibles such as concrete block and sand when Mr. Stomfay-Stitz was busy with other duties. But we also find that experienced inspectors informally attended to the receipt-inspections in his absence and we conclude that the matter is of little consequence.

D-300. Partly as a consequence of Applicant's default in failing to make a full evidentiary presentation on the allegation concerning storage of post-tensioning tendons, we found above that Mr. Stomfay-Stitz was improperly told to inspect the tendons weekly instead of daily as prescribed. We also found that the tendons were stored in a wet and muddy place with high humidity. In fact some of the tendons rusted and had to be replaced. While the tendon storage situation did not constitute a safety matter because of later inspections and corrective actions, the storage conditions represented a careless quality control procedure, a finding we weigh against Blount's general QA program.

D-301. Three of our findings indicate a weakness and lack of independence in the Blount quality assurance function. We found that Mr. Stomfay-Stitz was required to help the production workers find slotted connections missed by the workers. This was an inappropriate assignment for a quality assurance employee as it suggests a blending of the production and quality assurance functions.

D-302. Not only was Mr. Stomfay-Stitz required to do production work, but his boss at the time, Mr. Barnhart, supervised him in that activity. Thus the overlapping of production and quality assurance was not limited to Mr. Stomfay-Stitz who performed those tasks as a trainee. In addition, Mr. Barnhart was not even certified to do the bolting-in inspections during the time he was training Mr. Stomfay-Stitz in that procedure. We also found that the onsite production managers controlled

Mr. Stomfay-Stitz' wages, which we believe to be relatively low. This in our view was a further demonstration of insufficient quality assurance independence and a low regard for that function at Blount.

D-303. On one hand, the Board is reluctant to conclude that the Blount QA organization was weak and lacked independence, because it depends in part upon Mr. Stomfay-Stitz' allegations and in part upon negative inferences drawn from Applicant's failure to present evidence. Yet, there were multiple signs of this weakness. Mr. Stomfay-Stitz' assignment of checking the slotted connections was not disputed nor was Mr. Barnhart's role in that effort. The allegation that production managers controlled the inspectors' low wages is convincing and the negative inference that this was the case is a reasonable one. We find therefore that, at least during Mr. Stomfay-Stitz' tenure, Blount had a weak quality assurance organization lacking independence.

D-304. What is to be made of this finding? It is impossible to assign a quantitative value to it. Our finding is limited to the QA organization as a structure and not to the quality of its performance. The Board has been afforded only a brief snapshot view of a small sampling of events surrounding Blount during 1979 as a result of Mr. Stomfay-Stitz' allegation. We were impressed by the very thorough investigation conducted by Region III into Blount's activities as a result of the worker allegations and we weigh heavily Region III's view that there were no inherent QA problems at Blount during the relevant period. Region III Testimony, ff. Tr. 3586, Attachments E and F. We note also that, in the Byron Inspection Chronology, Blount infrequently appears as a problem.[63] The Staff's Systematic Assessment of License Performance, second report (SALP-2) rates the work traceable to Blount at Byron (containment) as better than average for Region III. Region III Testimony, ff. Tr. 3586, at 12, Attachment D. Finally we note that Blount is one of the contractors whose work is subject to the very intensive reinspection program as a result of the special Construction Assessment Team inspection, 82-05.

D-305. Balancing all of these factors, the Board concludes that despite our finding of a 1979 structural weakness in Blount's quality assurance organization, there is insufficient basis for the Board to conclude that the Blount quality assurance program was inadequate.

[63] Region III Testimony, ff. Tr. 3586, Attachment A. *But see id.* at 21 where a notation of poor housekeeping was made and later closed, and *id.* at 49 where a Level IV violation was noticed with respect to concrete and civil/structural procedures.

*Hatfield Electric Company*

D-306. The Hatfield Electric Company is the electrical contractor at Byron. It is a fairly small company and the Byron job is essentially its only project. Stanish, ff. Tr. 2619, at 6. Our findings with respect to Hatfield are predicated primarily upon its quality assurance noncompliance history as revealed by earlier NRC inspections, the allegations of John Hughes, a quality assurance inspector previously employed at Hatfield, NRC inspections of other, unidentified workers' allegations, and the NRC's special Construction Assessment Team's inspection of Byron during the Spring of 1982 (the "82-05" inspection).

Hatfield's General Noncompliance History

D-307. In August 1978, Region III issued a Notice of Violation to Applicant based, in part, on Hatfield's failure to delineate in an Applicant-approved procedure how Hatfield intended to comply with American National Standards Institute (ANSI) standard 45.26-1973 which refers to the qualifications, levels of capability and physical capabilities of quality control inspectors. Intervenors Ex. 3 (Inspection Report 78-07), Appendix A; Tr. 3645 (Hayes). This noncompliance was closed out with the expectation that Applicant's audits would assure compliance with the ANSI standard. Tr. 3648 (Konklin).

D-308. In June 1979 a former Hatfield employee made general allegations concerning Hatfield quality assurance practices. As a result of its ensuing inspection, Region III issued a Notice of Violation finding noncompliances where Hatfield incorrectly installed cable connectors and concrete expansion bolts. The faulty installation had been correctly identified but not in an established document control system. Intervenors Ex. 4 (Inspection Report 79-18), Appendix A. Region III regarded the noncompliance with respect to concrete expansion bolts to be a programmatic weakness, but the matter was finally resolved to its satisfaction. Tr. 3650 (Hayes). This is the first one reported in a series of six such episodes indicating a continuing weakness in Hatfield's ability to maintain a reliable document control system. It is a matter of considerable concern to the Board, as we discuss in more detail in the following paragraphs.

D-309. Region III inspected Hatfield's activities at Byron again in December 1980 and in its Inspection Report 80-25 (Intervenors Ex. 5) made several very serious findings of noncompliance against Applicant as a result of seven different violations of the Appendix B QA criteria, including such significant deficiencies as:

(1) failure to apply QA program requirements to the design, construction, purchase, and installation of a safety-related component, (2) failure to adequately translate design documents into drawings, instructions and procedures, (3) failure to identify and correct deviations and nonconformances, (4) failure to establish procedures, and (5) failure to follow procedures.

Intervenors Ex. 5, at cover page. Region III stated that it was in particular very concerned about the first item of noncompliance and that it had strongly considered classifying the cited deficiencies in the electrical and instrumentation quality assurance program as a Severity Level III violation. *Id.* This severity level at the time was very serious because such a violation could compromise the safety of the plant. Tr. 3655 (Williams).

D-310. The first item of noncompliance found that cable entrance frames for seismic category 1 safety-related equipment were designed without engineering approval, built without an approved QA program and purchased and installed without QA approval. Intervenors Ex. 5, Appendix A, at 1. The second violation involved impermissible bundling of safety-related cables with nonsafety-related cables. *Id.* at 2. Other violations included a misinstallation of a cable, nonconforming welds, deviation from cable tray filling specifications (above side rails), and failures to implement documented instruction, procedures and other document controls. *Id.* at 2-3.

D-311. Region III, however, decided that the violations would issue at Severity Level IV because the matter did not indicate a "breakdown of the program." But an Immediate Action Letter issued because it was a matter that could not wait for normal action. *Id.* at 2; Tr. 3695-96 (Williams). As a result of the 80-25 inspection findings, Applicant stopped Hatfield's work at Byron from January until April 1981. Tr. 2578 (Shewski). Applicant presents the stop-work order as an example of its strong control over its contractors. Proposed Findings 516 through 522. Shewski, ff. Tr. 2364, at 17-18. Stopping work was the appropriate action, but Applicant fails to mention that Hatfield's work was stopped when the NRC participated in Applicant's deliberative process and participated in arriving at the stop-work conclusion. Had Applicant not stopped Hatfield's work, Region III would have stopped it. Tr. 3918 (Williams).

D-312. Subsequently, corrective actions acceptable to Region III were implemented. Shewski, ff. Tr. 2364, at 18-19.

D-313. The special Construction Assessment Team inspection of Byron during Spring 1982, the 82-05 inspection, was very extensive. Hatfield, more than any other contractor, brought troubles upon Applicant as a result of additional noncompliances. *See generally* Applicant

Ex. 8 and Stanish, ff. Tr. 2619, *passim*. To the Board the most significant of these noncompliances involved several episodes in the pattern of maintaining unreliable and inadequate documentation of nonconforming conditions.

D-314. As we have noted throughout this decision, a system of maintaining documentation of nonconforming conditions is essential to the reliable tracking and trending of nonconforming conditions. The need for reliable reports on deficiencies and nonconforming conditions pervades the QA criteria of Appendix B. *See* Tr. 2646-49 (Stanish).

D-315. In the first of this 82-05 series, Region III found:

> On March 30, 1982, it was identified that Hatfield Electric Company was utilizing a Discrepancy Report System, which was not referenced or controlled by a procedure, to track and correct discrepancies and nonconforming conditions discovered during inspections of safety-related equipment.

Applicant Ex. 8, at 3. Applicant's explanation of this noncompliance is that the documents in question ("trouble letters") were used by quality control personnel to notify production personnel, not to document corrections, but that appropriate Hatfield procedures had not explained the use of the document. Stanish, ff. Tr. 2619, at 8-9. This explanation is not very reassuring.

D-316. The next noncompliance involving inadequate quality assurance documentation was discovered on April 7, 1982 by the Special Construction Assessment Team who found

> that three (3) nonconformance reports [NCRs] (98, 99, and 100) had been voided by the Hatfield Electric Company rather than closed, with reference to corrective action taken to resolve the nonconformance. The subject NCRs were voided because an FCR [Field Change Request] was or would be issued to accept the items as installed. At the time the NCRs were voided, there was no assurance that all the FCRs would be approved. By voiding the NCRs, the tracking system to verify that the proposed disposition was accepted, was negated and the NCRs were removed from the trend analysis system.

Applicant Ex. 8, at 4.

D-317. Applicant explains this item of noncompliance by stating the Staff's concern was that tracking would be lost if the Field Change Requests were rejected. This was not the case, however, according to Mr. Stanish who stated that adequate tracking existed with the FCRs. However, a revision of the procedures was made to provide for correct trending. Stanish, ff. Tr. 2619, at 13.

D-318. Also on April 7, 1982, the special Construction Assessment Team found:

> that the Hatfield Electric Company had improperly closed NCR [nonconformance report] 168, in that after CECo engineering dispositioned the subject NCR to replace the item, the Hatfield Electric Company closed the NCR without accomplishing the approved disposition. At the present time, there is a nonconforming cable installed, and the tracking system to replace the cable, has been negated.

Applicant Ex. 8, at 5. Mr. Stanish explained that the nonconformance report only *appeared* to have been closed before the cable was replaced because the report indicated that the action taken was that the cable did not require replacement. A subsequent review revealed that the cable in question was removed and a new cable had been installed. As a result, the required corrective action had been satisfactorily completed and the information on the NCR was incorrect. Stanish, ff. Tr. 2619, at 14.

D-319. In addition to documentation problems, the special team in its 82-05 inspection found that Hatfield Electric did not tag torque wrenches which were past due for calibration. Mr. Stanish testified that they were immediately tagged (Applicant Ex. 8, at 5) and sent out for recalibration. The appropriate personnel were instructed in the importance of identifying tools past due for calibration. He believed that the discrepancy did not affect quality because wrenches were not used after the calibration due date. Stanish, ff. Tr. 2619, at 15.

D-320. Finally, during the 82-05 inspection, the special team noted:

> that Hatfield Electric Company procedures did not contain an electrical cable rework procedure nor the requirements to calculate electrical cable sidewall pressures prior to pulling cable.

Mr. Stanish explained this situation by stating that when calculating the maximum cable pull tension, Hatfield did not determine the maximum pressure on the cable sidewalls. He said that "[w]here extremes of limitations of pulling radius were used and actual tension required to pull a cable were near maximum, it is possible in isolated cases to violate the maximum allowable sidewall pressure . . . ." The method of calculating maximum pulling tension was revised and implemented. Cable-pull reports for cables already installed are being reviewed against the current criteria and any needed corrective action will be taken with the advice of the cable manufacturer. All cables, regardless of when installed, will meet the current criteria.

D-321. Another aspect of the cable-pull noncompliance was that the Hatfield procedure did not address precautions to take when "reworking" cable pulls. The NRC inspector felt precautions should be documented to have cable-pulling activities require the same care in reworking cables as in initial installation — which was the actual practice, but undocumented. Stanish, ff. Tr. 2619, at 10-11.

John Hughes' Allegations

D-322. Mr. John Hughes was an employee of Pittsburgh Testing Laboratory (PTL) assigned to and under the control of Hatfield Electric Company as a quality control inspector from October 1, 1982 to January 7, 1983. On April 27, 1983 Intervenors moved late to allow the testimony of Mr. Hughes with respect to his allegations against Hatfield and attached a short affidavit to the motion. The quality assurance phase of the hearing had already passed. Because of the serious implications of Mr. Hughes' allegations, the Board by order of May 12 ordered that Mr. Hughes' deposition be taken in a session to be presided over by the Board. The purpose of the deposition was to determine whether the evidentiary record should be reopened on the QA contention with the expectation that, if so, the deposition itself could constitute a portion of that record where appropriate. The session was convened and the deposition was taken over a full day as if it were an evidentiary hearing, *i.e.*, with direct and cross-examination and exhibits. Tr. 7012-7231.

D-323. The parties filed briefs arguing the significance of Mr. Hughes' allegations much the same as though they were proposed findings on an evidentiary record. Intervenors requested the Board to accept Mr. Hughes' testimony at the deposition into the evidentiary record and to consider it in ruling on the QA contention.[64] At that point Intervenors rested their QA case. They sought no further hearing opportunities except for a later motion, subsequently denied, to reopen on other QA matters.[65]

D-324. On June 21, 1983 the Board issued its order ruling on Intervenors' motion and on the same date issued another order reopening the evidentiary record on certain aspects of Mr. Hughes' allegations and on other matters relating to Hatfield.[66]

D-325. The other matters surfaced because, in evaluating Mr. Hughes' allegations and deposition, the Board focused more sharply on other evidence related to Hatfield that had been presented during the QA phase of the hearing. In our view the record was incomplete. One of the other matters related to the allegations of three unidentified persons concerning certain Hatfield QC practices then and now under inspection and investigation by Region III and the Office of Investigations. Another matter pertained to a short reference by Region III in its testimony on

[64] Joint Intervenors' Brief in Support of Motion to Admit Testimony of John Hughes, June 7, 1983.
[65] Intervenors' Motion to Supplement QA/QC Record Regarding Preoperational Testing, June 29, 1983.
[66] Memorandum and Order Ruling on Intervenors' Motion to Admit Testimony of John Hughes, June 21, 1983 (unpublished) ("Hughes Order"), and Memorandum and Order Reopening Evidentiary Record, June 21, 1983 (unpublished) ("Reopening Order").

the 1982 inspection findings that QA/QC supervisors and inspectors employed by several contractors were not adequately qualified or trained. Accordingly the Board reopened the evidentiary record for further inquiry into the allegations by Mr. Hughes and the two other matters. *See* June 21 Reopening Order. Subsequently on July 7, we supplemented the order to clarify that the reopened proceeding would be limited to Hatfield Electric Company.

D-326. In our June 21 Hughes Order, we ruled that most of Mr. Hughes' testimony and some exhibits would not be a part of the evidentiary record. However, two of his allegations were litigated and their resolution depends in part on how accurate the Board perceives Mr. Hughes to be generally. Accordingly we revisit our June 21 Hughes Order in that context.

D-327. First we noted that his memory was uncertain on some matters important to his perceptions. Hughes Order at 5. Of more importance, however, his initial allegations did not comport with his later deposition testimony in very important aspects. Several examples from his affidavit are evident.

D-328. *Initial Allegation:*

> My training at Byron was by Hatfield and consisted of reading procedures and being tested.

*Implication:* Insufficient training and fraudulent certification of training.

*Fact:* His respective testimony recalled at least 2 hours of classroom training and some on-the-job experience. A Region III inspection confirmed even more formal training. Nevertheless, because of his persistence that he began to work as an inspector before there was sufficient time to train him, and for other reasons, training and certification thereof was one of the issues heard on the reopened hearing.

D-329. *Initial Allegation:*

> I failed my first exam and was retested about ½ hour later and was given the answers for the questions which I missed the first time before taking the test for the second time.

*Implication:* Mr. Hughes could not pass the test on his own and the test was fraudulent.

*Fact:* Mr. Hughes took six tests, passed five, and failed one by only two points. A senior inspector reviewed the "pertinent procedures" with Mr. Hughes before he took the second exam, compared to simply providing the answers. The tests were rather simple. Mr. Hughes testified later that he didn't use or need the provided answers because there were only a few questions involved and that he "had researched them over in [his] own mind. . . ."

D-330. Accordingly the Board ruled that there was no significant safety issue to reopen with respect to the testing allegation as it narrowly pertained to Mr. Hughes' inspection skills. But, because he testified, with corroboration from Mr. Souders, that, soon after failing the first test, the failed test was given to him with correct answers to use during the retesting, and because he alleged that this was a regular practice, and because of a pending investigation about this allegation by the Office of Investigations, the Board included the possibility of fraudulent testing at Hatfield Electric as one of the two Hughes' issues to be heard during the reopened hearing.

D-331. *Initial Allegation:*

> I was ask[ed] to sign off [on] Documentation for inspection I did not perform.
>
> I did this until Dec. 1 then I refused to do this any more.
>
> My supervisor ask[ed] me if I did not like their program. I replied it is not your program it is the way you are going about it. After refusing to sign documentation, I was given other tasks and then laid off after about a month.

*Implication:* Fraudulent documentation on nonexistent quality assurance inspections. Hughes, an indignant inspector, refused to be involved and was fired on that account.

*Fact:* The printed inspection forms required Mr. Hughes to sign at a place designated "Inspection Completed By," but he said that the form was misleading because, as a Level II inspector, he did not physically inspect the work. He reviewed against specifications the information produced by Level I inspectors. Mr. Hughes would have been satisfied with the form if the sign-off line had indicated "results evaluated by." There is no evidence that the form was intended to mislead or that it did mislead. No one was lead to believe that Mr. Hughes had physically inspected the work when in fact he hadn't. Mr. Hughes was not discharged for refusing to sign fraudulent inspection documents — the documents weren't fraudulent. Contemporaneous records report that Mr. Hughes was laid off for repeated instances of lack of productivity, poor corporate attitude and inattentiveness to the tasks at hand.

D-332. Mr. Hughes' allegation that he was asked to document inspections he did not perform and the implication that he was fired for refusing to do so was, in part, a distorted exaggeration, and in all other parts, untrue.

D-333. Mr. Hughes also made allegations about bad welding which are probably true but were already the subject of an NRC-imposed inspection and remedial program as we discuss below. He also made an allegation concerning improper welding on a cable tray while cables rested in the tray, which was not substantiated. But the allegation was probably made reasonably and in good faith.

D-334. In sum, the Board reopened the hearing partly to inquire into Mr. Hughes' allegations, with serious doubts about the accuracy of his memory and with low confidence in his candor. However, based on the entire record as it existed at the time, we could not discount these important aspects — training, certification and testing.

MR. HUGHES' TRAINING AND CERTIFICATION

D-335. Although Mr. Hughes' memory of many aspects of his original allegations was poor, he seemed to be positive in his memory that he was certified as a Level II QC inspector and began working in that capacity within two weeks of his assignment to Hatfield on October 1, 1982. Tr. 7059-60; Tr. 7208-09, 7216. If so, Mr. Hughes' testimony that his training was perfunctory would be believable. He would scarcely have had time for the training reflected in the Hatfield summary records.

D-336. Mr. Hughes was to be trained as a cable pan and cable pan hanger inspector at Level II. Originally he had been designated to qualify in three separate inspection procedures but because of insufficient on-the-job training he was certified to perform only one inspection procedure, 9A — cable pan hanger installation.

D-337. Because of previous experience at other nuclear plants he was eligible, after suitable training, to be a Level II rather than a Level I inspector. According to Mr. Koca, the Hatfield witness, Mr. Hughes was qualified by that experience alone to meet NRC and ANSI (American National Standards Institute) standards as an inspector, but Hatfield had a general policy requiring its inspectors to have a high school diploma or an equivalency (GED) certification in addition to the pertinent experience and training, to be certified as an inspector. An exception to this policy is possible, and there has been one exception, but the policy was imposed on Mr. Hughes who at the time of his employment on October 1, 1982 had neither a diploma nor a GED certificate. *See generally* Koca, ff. Tr. 7418.

D-338. Mr. Hughes acknowledged that Hatfield required him to receive his certificate of GED high school equivalency before he was allowed to begin working as a certified inspector. Tr. 7200-02. The GED certificate, received in evidence as Applicant Ex. 38, indicates that Mr. Hughes passed his GED tests on October 28, 1982. The certificate itself was dated October 29. These facts would suggest that Mr. Hughes' memory that he began work within two weeks of the date he was hired, about October 1, was incorrect. But he testified that prior to receiving the official certificate, he had received a card from the GED examiner indicating that he had passed the test. Tr. 7201. This testimony, and the fact that the certificate stated: "Date Reported *October 12, 1982,*" suggested to the Board that the examination dates on the certificate could be incorrect and that the GED examiner may have reported Mr. Hughes' passing test results on October 12. It was mainly for this reason that the Board requested an evidentiary showing on the training and certification background of Mr. Hughes' employment.

D-339. An inspection by Region III into the timing of Mr. Hughes' training and certification established conclusively that Mr. Hughes did not receive his GED high school equivalency certification until October 29, 1982, as stated on the certificate. Region III reported:

> Mrs. Darlene Lee of the Office of Education was contacted in regard to Mr. Hughes' GED. In response to questioning, Mrs. Lee stated that an applicant is required to successfully complete examinations in six general areas to receive a GED certificate in the State of Illinois. This includes a test in the U.S. and Illinois Constitutions. Mrs. Lee also stated that Mr. Hughes failed a GED test given by the Armed Forces Institute in November 1967, but was given credit for three of the examinations and successfully completed the other three required tests in Rockford, Illinois (two on October 13, 1982 and the third on the evening of October 28, 1982). Mrs. Lee stated that no official verification or document would have been given by the Rockford Regional Office of Education prior to Mr. Hughes completing all State of Illinois requirements for a GED certificate.

Region III Testimony, ff. Tr. 7801, at 16-17.

D-340. Furthermore, Region III inspectors produced documentation establishing that Mr. Hughes received formal, precertification training on various aspects of Hatfield inspection procedures from October 6 through 29, 1982. Attachment H to Region III Testimony is an indoctrination checklist initialed by Mr. Hughes himself which establishes this training. Attachment G to the testimony established that Mr. Hughes signed off on a notification that he had received formal classroom training on October 28 in four separate training sessions.

D-341. On November 1, 1982 the QA/QC manager at Hatfield certified that Mr. Hughes was qualified as a Level II inspector on Procedure 9A and that certification was received in evidence. Region III Testimony, ff. Tr. 7801, Attachment to Attachment F. Mr. Hayes of Region III testified that the Staff reviewed approximately 1800 inspection reports covering the entire period Mr. Hughes was employed by Hatfield. Based upon these reports, and based upon discussions by the Staff with Hatfield inspectors who worked with Mr. Hughes, the Staff concluded that Mr. Hughes' allegation that he performed inspections during his first two weeks on the job, or at any time prior to his certification on November 1, 1982, other than the inspections which comprised his on-the-job training, was not substantiated. Region III Testimony, ff. Tr. 7801, at 17.

D-342. Mr. Hughes' allegation that he was certified to work and began working as a quality control inspector at Hatfield within two weeks of his employment is unfounded and incorrect. The preponderance of the evidence is that he did not begin working independently as an inspector until at least November 1, 1982 and that there was sufficient time for training. Contemporary documentation establishes that he received the appropriate classroom training.

D-343. But Intervenors dispute whether Mr. Hughes received the requisite amount of on-the-job training. Mr. Koca, the Hatfield Quality Assurance supervisor, was Mr. Hughes' quality control supervisor during the latter's employment at Hatfield, and testified on behalf of the Applicant. Ff. Tr. 7418. In an effort to demonstrate that Mr. Hughes had 64 hours of on-the-job training before being certified as a Level II inspector, Mr. Koca produced an on-the-job training report on Mr. Hughes, signed by Scott Wagner, the Level II inspector charged with training Mr. Hughes on the job. *Id.*, Exhibit G. The report purports to document thirty-two on-the-job inspections with Mr. Hughes, *each lasting exactly 2 hours* for a total of 64 hours of on-the-job training in Procedure 9A. No convincing explanation was given for this unlikely series of events. We suspect that the record was an after-the-fact estimation but the document itself has little probative value.

D-344. The Region III inspectors, however, were able to satisfy themselves by examining the actual inspection reports, bearing both Mr. Wagner's and Mr. Hughes' signatures, and by interviews, that about 48 hours of on-the-job inspection training was accomplished, which is more than enough (40 hours) to support Mr. Hughes' certification in Procedure 9A. Some of the inspection reports which could have verified the remaining 16 hours of on-the-job training were inadvertently lost or destroyed as a result of the reinspection program. Region III Testimony, ff. Tr. 7801, at 14-15.

D-345. Mr. Hughes' allegation with respect to the amount and timing of his training at Hatfield is unsubstantiated. However, as we find in relation to the Hatfield inspector recertification program, *infra,* at least half of the sixty to seventy Hatfield inspectors were later found to lack the requisite 40 hours of documented on-the-job training.

MR. HUGHES' ALLEGATION OF CRIBBING ON TESTS

D-346. Mr. Hughes' allegation that he had the failed test with corrected answers available to him when he was retested had some corroboration from the stipulated testimony of Irvin Souders, an inspector who worked with Mr. Hughes at Hatfield. Ff. Tr. 7020, at 2. As noted above, the Board was also aware that the allegation that Mr. Hughes was aided in his retesting by the corrected failed test was then under investigation by the Office of Investigations. These circumstances led the Board to include the issue of alleged fraudulent testing at Hatfield in the reopened hearing. Office of Investigations made available to the Board a document which that office regards as the test in question. It purported to show corrections to wrong answers. Intervenors Ex. 27. The matter is still under investigation by the Office of Investigations.

D-347. The circumstances surrounding the questioned document do not provide much assurance either way that it is or is not genuine. Region III's Mr. Forney testified that, during their interview with Mr. Hughes, he reported that the failed test had been thrown away. Later another, unidentified, person provided a copy of the questioned document, Intervenors Ex. 27, claiming it to be Mr. Hughes' failed test. Tr. 7972-74.

D-348. Mr. Koca, Hatfield QA supervisor, explained how tests were administered during the period of Mr. Hughes' employment. A Xerox copy of the master test would be provided to the trainee who would take the test while under observation by a Hatfield official. On failed tests, Mr. Koca would review the answers with the trainee, providing the correct answers. Also, the trainee was to have studied more about the missed questions. Mr. Koca preferred two days to elapse before the trainees retook a failed test, but sometimes both tests would be taken the same day, first in the morning, then in the evening. Mr. Koca would retain failed test papers but only until that test was passed, then the failed test would be destroyed. He could not recall that Mr. Hughes had failed a test but, if he had failed, that test should have been collected from him. Mr. Koca has no knowledge that Mr. Hughes or other trainees had the correct answers available on retesting and such a situation would be contrary to Hatfield procedures. Koca, ff. Tr. 7418, at 10-13; *see also generally* Tr. 7480-7501.

D-349. Mr. Koca testified on cross-examination that the suspected document — the allegedly failed test — Intervenors Ex. 27, was not a test scored by him because his practice is to write the score, his initials and the date on the document, information absent from the suspected document. Tr. 7479. This suggests a fake document. Some of the marks on the test looked like his, some did not. *Id.* He defended his testimony that he would mark the score, date, and his own initials on a failed test even though he would later destroy it, by stating that, in the interim, the boss might want to examine the test — perhaps on appeal by the failing trainee. Tr. 7479. Intervenors urge the Board to find that Mr. Koca's account is not credible.

D-350. The Applicant argues that the document in question supports Mr. Koca's testimony regarding the standard practice; that the document bears the handwritten date of October 8, 1982 but the same test was actually passed by Mr. Hughes on October 12, *ergo,* sufficient time had passed. Applicant's Proposed Finding 770. A copy of the passed test is attached to Mr. Koca's testimony, ff. Tr. 7418, Attachment K.

D-351. The Board cannot arrive at a reliable conclusion with respect to this allegation. The litigation centered around one document, the alleged failed test with corrected answers. Its authenticity has not been established and the portions of it which have any tendency to be helpful are illegible. The Board erred in receiving it. Tr. 7547. We cannot accept Mr. Hughes' allegations, or Mr. Souder's stipulated testimony, as being sufficiently reliable to conclude that the questioned document represents a practice of providing corrected failed tests as cribs in retesting. On the other side of the issue, Mr. Koca has a strong interest in defending whatever practice then existed and his memory is uncertain.

D-352. While further inquiry by the Office of Investigations into the particulars of Mr. Hughes' specific allegation is appropriate, the Board does not believe that the allegation can be tested by regarding Mr. Hughes' accusation and his particular experience as bounding the universe of evidence on the issue. The matter cannot be resolved until the Office of Investigations completes its inquiry, if then.

D-353. In the meantime, however, Hatfield has adopted new and improved testing procedures with sensible safeguards, which even Intervenors believe are comforting as to the future. Intervenors' Proposed Findings 43-44.

D-354. The Board's ultimate finding with respect to Mr. Hughes' allegations is that he has been very unreliable and inaccurate. However, the Board continues to have concern about Hatfield's inspector testing and certification procedures.

Other Worker Allegations

D-355. During the main hearing, Region III's prepared testimony alluded to allegations by three persons concerning Hatfield's work at Byron including references to inspector qualifications and certification, recordkeeping, and QC inspector independence. About half of the allegations had already been identified, but the remaining were then under evaluation by Region III and the Office of Investigations. Region III Testimony, ff. Tr. 3586, at 6. This testimony passed without particular attention until the Board began reviewing the record on the QA contention, especially Mr. Hughes' allegations. We needed to know more about the allegations and requested an evidentiary presentation in the June 21 Reopening Order. A panel of Region III witnesses then addressed the Board's inquiry. Ff. Tr. 7801. They summarized the status of NRC investigations into allegations received from Individuals "A," "B," "C," and John Hughes, from August 2, 1982 to the date of the hearing. Sixty-five unique allegations were received from these individuals. Of these, thirty-four have been inspected by NRC Region III personnel and disposed of (thirty-two are closed and two remain open pending verification of corrective action). The remaining thirty-one, including ten allegations referred to the Office of Investigations, have not been investigated. As a result of inspections then completed, Region III believes that only five allegations were substantiated. Two substantiated allegations remained open pending completion of corrective action by the Applicant and verification by the NRC. The open allegations concern (1) the utilization of former craft personnel as quality control inspectors without having established measures to assure that such inspectors were not inspecting their own work and (2) the acceptance of cable pan hanger connection detail based upon information provided on weld card travelers. The latter matter remained open pending completion of a review by the Applicant to determine whether verification of connection detail has been accomplished as part of the weld inspection and will be followed up by NRC Region III personnel prior to closure. *Id.* at 8-11, 19, 20.

D-356. Pertinent parts of the complete inspection reports were received into evidence as attachments to the Staff's testimony. *Id.*, Attachments B, C, and D. Intervenors followed through on several of the allegations. The Board has reviewed the inspection reports and Region III testimony and include in our findings below those allegations which appear to have significance bearing on the integrity and effectiveness of Hatfield's quality assurance program.

D-357. One allegation was that the Hatfield Quality Assurance Manager was inept and incapable of performing as a quality assurance manager. To investigate this allegation, Mr. Forney of Region III, in February 1983, reviewed the manager's certification package. Neither Hatfield nor the Applicant had performed sufficient review of the manager's educational background. Instead, Hatfield and Applicant relied upon a certification letter from a company in the business of providing managers. That letter suggested that the quality assurance manager was qualified, but would need specific training in the nuclear area. Tr. 7918-20 (Forney).

D-358. The Hatfield Quality Assurance Manager had spent a number of years as a manager of different companies in Rockford and other areas. Because his past experience as a manager involved quality assurance functions only part of the time, Mr. Forney determined that this prior experience should be entitled only to partial credit toward the background requirements of a quality assurance manager. On this basis, Mr. Forney concluded that the Hatfield Quality Assurance Manager did not have sufficient prior work experience to be properly certified. Mr. Forney therefore issued an item of noncompliance. Tr. 7919-20 (Forney).

D-359. The Quality Assurance Manager was removed from his position. He is still with Hatfield. In addition, to correct this situation, Applicant reviewed the certification packages of all Hatfield quality control inspectors. Tr. 7921 (Forney). Mr. Forney personally has reviewed the records of all the Level III inspectors who were working for Hatfield at the time he reviewed the quality assurance manager file. Tr. 7929 (Forney).

D-360. Compounding the problem that the Quality Assurance Manager was unqualified was the finding the following month by Region III that the Quality Assurance Manager reports to the corporate vice president who was located on site at Byron and had direct responsibility for costs and schedule. Applicant was cited for this failure of quality assurance independence. Applicant Ex. 8, Appendix. The problem has since been resolved by having the QA manager report off site to the company president. Stanish, ff. Tr. 2619, at 6.

D-361. Another allegation was that a Hatfield Level II QC inspector, Mr. Wells, had prior experience as a carpenter, implying that he had no previous qualifying nuclear experience. In February 1983 Region III found that the inspector had been erroneously certified. Region III Testimony, ff. Tr. 7801, Attachment D. His previous work at Byron, on cooling towers, was deemed not to be nuclear-related. Other experience as an inspector was not in a formal program. Therefore Region III gave him "zero" credit for experience and issued a Level IV item of noncompliance. It is particularly significant that this type of problem was to have been resolved as of September 1982 in a Hatfield inspector recertification program as a result of Inspection Report 82-05-19 (discussed below). *Id.* This incident provided a basis for concern by the Staff that the problem of inspector certification at Hatfield, thought resolved, was continuing — particularly in Hatfield's interpretation of acceptable work experience. Tr. 7915-17 (Forney). Moreover, as we noted at the outset of our findings on Hatfield, Region III had once before (in 1978) received assurances from Applicant that Hatfield would comply with existing ANSI standards for inspector qualifications. Paragraph D-307, *supra,* Intervenors Ex. 3, Appendix A.

D-362. A third, and very serious, allegation was that inspectors had prepared discrepancy reports to document findings in the power block but that the reports were destroyed by Hatfield supervisors. Region III investigated and, in January 1983, noted discrepancies, as it were, between discrepancy reports and the discrepancy report log book. The NRC inspectors also noted that it was almost impossible to prove or disprove whether the log had been altered because it was of a loose-leaf type. Corrective action requested by the NRC, and instituted, was to use a bound ledger-type log book for discrepancy reports, nonconforming reports and the like. Ff. Tr. 7801, Attachment C, at 7; Tr. 7895-96. The matter was not regarded as a violation by Region III. To the Board, however, the situation seems to be a violation of at least two of the criteria of Appendix B to Part 50. Criterion XVII requires that adequate quality assurance records be maintained, and Criterion XVIII requires auditing of the QA program. In this case, an allegation that appropriate records were not maintained apparently had some objective support, but no reliable audit of the records was possible because of the use of loose-leaf logs. The matter did not reflect well on the Hatfield quality assurance program.

D-363. In March 1983, Individual "C" made allegations to the resident inspectors concerning the handling of quality assurance documents involving potential intentional wrongdoing which were forwarded to the Office of Investigations where the matter pends. Region III Testimony, ff. Tr. 7801, at 19.

D-364. In November 1982, some Hatfield workers, including Mr. Hughes, made general and specific allegations concerning the quality of welding of electrical hanger installations by Hatfield. At the time of the allegations, Region III had already initiated a welding reinspection program at Hatfield as a result of Region III's "82-05-19" inquiry into inspector qualifications. *Id.,* Attachment B. As of June 4, 1983, 818 welding defects, primarily weld undercuts and overlaps, were identified on the 7,753 weld attributes which had then been inspected. Region III has not yet determined the safety significance of the problem as we discuss in greater detail in the following findings. Tr. 7806-09 (Hayes). With respect to these worker allegations, however, the reinspection program is predicated to encompass them. If it does not, additional inspections will be performed to resolve the allegations. Tr. 7955-56.

*The Reinspection Program and Inspector Recertification — More About Hatfield*

D-365. During March, April and May 1982, Region III conducted a special Construction Assessment Team inspection of the Byron units to assess certain aspects of the quality assurance in construction activities. Applicant Ex. 8. The joint inspection reports, 50-454/82-05 (DETP) and 50-455/82-04 (DETP) dated June 16, 1982, particularly Noncompliance Item 19 of the joint reports (Applicant Ex. 8, at 67-70), were the subject of extensive testimony in the reopened hearing. The parties referred to the inspection reports and the subsequent corrective programs as the "82-05" or the "82-05-19" inspection, report, or program respectively, a shorthand reference we have found useful.

D-366. As pertinent to the QA issue, the special Construction Assessment Team concluded:

> Based on a review of training qualification and certification records of a minimum of 10 percent of the QA/QC personnel working for contractors performing safety-related work it is apparent that an effective program does not exist to ensure that a suitable evaluation of initial capabilities is performed, that written certification is provided in an appropriate form, and that qualification criteria are established.
>
> Certain contractor QA/QC supervisors and inspectors were not adequately qualified and/or trained to perform safety-related inspection functions.

Applicant Ex. 8, at 67.

D-367. Examples included Hatfield Electric, and as we discussed earlier, Reliable Sheet Metal, Hunter Corporation, Blount Brothers and others. *Id.* at 68. Although the 82-05-19 inspection report has broad significance to the entire QA program at Byron, the Board's particular concern during the reopened hearing was about Hatfield.

D-368. The special team found that there was a wide variation in the implementation of requirements for QA/QC inspector certification and training by Byron contractors which was attributed to the Applicant's failure to establish a formalized program for contractors to follow. Region III reviewed training, qualification and certification records of some of the QA/QC personnel working for contractors performing safety-related work against the Applicant's FSAR commitments, Regulatory Guide 1.58 (Rev. 1), ANSI standard N45.2.6-1978, and 10 C.F.R. Part 50, Appendix B to determine if the certification/qualification program was adequate. Region III Testimony, ff. Tr. 7801, at 3-4; Tr. 7813-14 (Forney).

D-369. The Staff later expressed the concern that it is difficult for Applicant to perform an audit of a contractor's inspectors' qualifications without a formal program. Furthermore, Applicant's informal program failed to meet the *intent* of the ANSI and Regulatory Guide 1.58 standards for inspector qualifications. Although the Staff did not charge Applicant with willful failure to meet the standards, Applicant had not yet "put all of the commitments together." The dispute between the Applicant and the Staff was more than a matter of interpretation of the standards. It was a matter of whether the Applicant had actually honored its commitments. Tr. 7966-71 (Forney).

Reinspection

D-370. Although no specific hardware problems had been identified during the Construction Assessment Team inspection, Region III was concerned that the use of inadequately qualified inspectors may have resulted in unidentified conditions adverse to quality. Region III recognized that there is a difference between certification and qualification of inspectors, *i.e.*, an inspector who possesses the requisite qualifications but is not properly certified may have a minimal effect on quality and safety, whereas if a certified inspector lacks the requisite qualifications, the impact on quality and safety may be significant. In any event, Region III believes that, in fact, inspectors who were not qualified were employed on safety-related work at Byron. This includes Hatfield's work. Tr. 7840-42, 7847-48, 7860 (Forney).

D-371. The Staff considered an Applicant-reinspection program an appropriate way to determine if inspections were inadequate and if any hardware problems exist at Byron. Region III Testimony, ff. Tr. 7801, at 4-5. Although Inspection Report 82-05 was issued on June 24, 1982 (Applicant Ex. 8), it was not until February 1983 that Applicant proposed the reinspection program acceptable to the Region III Staff.[67] Earlier proposals were rejected. Tr. 7697-99 (Stanish).

D-372. Under the tentative reinspection program accepted by Region III, every fifth inspector from six contractors was selected from a

[67] The basic premise of the reinspection program was tentatively accepted by Region III. Certain aspects are still under Staff consideration. Tr. 7981 (Forney); *see* Paragraphs D-409 to D-410, *infra*.

chronological list based on the certification date of each QC inspector since the beginning of the project. A minimum of three additional inspectors from each contractor was selected by the NRC Senior Resident Inspector, Mr. Forney. Each inspection performed by the selected inspectors during the first 90 days of inspections is being reinspected where the item is accessible. In addition, for two contractors, Powers-Azco-Pope and Johnson Controls, each inspection performed during the first 90 days, by every inspector certified since construction began, is reinspected where the inspected items are accessible.[68] Tuetken, ff. Tr. 7760, at 4-6.

D-373. Other contractors that performed safety-related work were not included in the reinspection program because their work is now inaccessible, was inspected by an independent agency (*e.g.*, the Authorized Nuclear Inspector), was performed by properly certified inspectors or the work could not be re-created. Region III Testimony, ff. Tr. 7801, at 6.

D-374. The reinspection program also provides that another inspector be selected where all or most of an inspector's inspections are inaccessible. In addition, the sample size, both as to the number of inspections made by a selected inspector or the number of inspectors selected, would be increased if an unacceptable number of rejectable defects are identified during the reinspections. *Id.* at 7.

D-375. Mr. Tuetken, the assistant superintendent for construction at Byron, testified that the purpose of the reinspection program is to verify that deficiencies identified by the NRC in past QA/QC personnel training and certification did not result in unsatisfactory work going undetected. Ff. Tr. 7760, at 3-4.

D-376. The reinspection program is intended by Applicant to ensure that the work of each selected inspector attains a 95 percent quality level for objective attributes and a 90 percent quality level for subjective attributes. If the program demonstrates that the selected inspector has failed to meet acceptable quality levels with regard to attributes inspected during his first three months work, the next three months inspection by the inspector of the attribute in question is reinspected. If this sample

[68] The situation with these two contractors was worse than most. Powers-Azco-Pope is a joint venture of three companies established solely for instrumentation work at Byron. Stanish, ff. Tr. 2619, at 6. All eleven of its inspectors were found not to be properly trained, qualified and certified, and five of them were decertified. Tr. 7817 (Forney). It was not possible to verify from the files the QC's supervisor's education, previous employment or qualifications for either Level I inspector or Level II supervisor. Applicant Ex. 8, at 68. Johnson Controls, a contractor for HVAC controls, had one of its two inspectors checked. The records did not demonstrate education or prior work history and his certification testing (oral) was inadequate. *Id.* Both contractors used open-book testing. Tuetken, ff. Tr. 7760, at 5.

also fails, 100 percent of the failing inspector's inspections of the attribute in question is reinspected. In addition, if the first six months work fails to meet acceptable quality levels, the original sample of inspectors whose work is under reinspection is expanded by 50 percent for the attribute in question, *e.g.*, if one out of a sample of ten fails to meet acceptable criteria with respect to an item, the work of five additional inspectors involving the item is reinspected. Tuetken, ff. Tr. 7760, at 5-6; Tr. 7787 (Tuetken); Tr. 7988 (Connaughton). Any improper installation or construction work discovered during the reinspection program is to be reworked or reevaluated to an acceptable level. Tuetken, ff. Tr. 7760, at 6-7.

D-377. Twenty-two Hatfield inspectors or about 25 percent of the population of Hatfield inspectors certified up to September 1982 (the date Hatfield's and other contractors' certification procedures were revised and approved for use by the Applicant) were selected. Eighteen were selected at random (each fifth inspector) and four were selected by the NRC Senior Resident Inspector. Reinspected attributes consist of equipment setting, equipment modifications, conduit and conduit hangers, cable pan and cable pan hangers, bolting, welding, and cable terminations. *Id.* at 7-8.

D-378. At the time of the hearing, the results indicate that one Hatfield inspector will fail to meet acceptable quality levels with regard to his first three months work. The attributes involved concern weld inspection (weld detail, type and profile, size, length, cracks, fusion, porosity, undercut, slag, craters and overlap). The majority of inspection deficiencies identified involve weld undercut and overlap. The inspector will apparently achieve only a 75 percent acceptability level for subjective attributes during his first three months, but inspections during his next three months are expected to meet the 90 percent acceptability criterion. However, if the inspector fails to meet that criterion, all of the inspections performed by the inspector will need to be reinspected and the original random sample of Hatfield inspectors will be expanded by 50 percent (nine inspectors) with regard to the attributes at issue. *Id.* at 9-10. If nine additional inspectors are chosen, approximately 30 percent of the inspectors and 100 percent of the visual welding inspectors' first three months work would be included in the reinspection. Tr. 7774-75 (Tuetken). Reinspection of work by seven other Hatfield inspectors has been completed and results indicate each of the seven exceeded the established acceptable quality level. Tuetken, ff. Tr. 7760, at 10.

D-379. The Intervenors are sharply critical of several aspects of the Applicant's reinspection program and the respective evidentiary presentations. In June and July 1983 the Byron Quality Assurance Group conducted an audit of the contractors' reinspection programs, including Hatfield's. Intervenors Ex. 29. Region III indicated that an earlier audit would have been prudent. Tr. 7699-7700 (Stanish). The audit report came to light in the reopened proceeding only because Intervenors' counsel inquired whether such an audit had been conducted and noticed that the Byron Quality Assurance Superintendent, Mr. Stanish, referred to it during his testimony. Tr. 7642-43.

D-380. Applicant's audit revealed that Hatfield (and other contractors, Hunter Corporation and Blount Brothers) were not issuing discrepancy reports on nonconforming conditions discovered during the reinspection program. Instead the nonconforming work would be corrected by using a "field problem sheet." Intervenors Ex. 29, at A1; Tr. 7702-04, 7750-51 (Stanish). The problem with using this type of documentation is that the appropriate record to identify defective inspections would not be generated and the main purpose of the reinspection program would be defeated because the problems would not show up in a trend analysis. Tr. 7752 (Stanish). In fact Hatfield was not even following its own program designed to identify nonconforming conditions during the reinspection. Tr. 7703-04 (Stanish). This is yet another example of the problem plaguing Hatfield's quality assurance program — the failure to maintain a reliable system of nonconforming documentation control — first observed by Region III in its August 1978 (78-07) inspection. Intervenors Ex. 4.

D-381. The Applicant's audit also established that Hatfield had failed to document the evaluation for the nature of work to be reinspected and Applicant could not verify that the correct evaluation had been made. Tr. 7707-08 (Stanish). Hatfield also misunderstood the requirement to increase the sample population of inspectors when a given inspector fails the first round of reinspections. The misunderstanding led to a smaller increase than required. Tr. 7728-29 (Stanish).

D-382. Yet another audit finding faulted Hatfield for not complying with its QA/QC memorandum #295 where a weld inspection acceptance of cable pan or conduit hangers implies verification of the correct connection detail. This type of acceptance occurred even when the reinspection could not verify the connection detail because fireproofing had been installed over the work. Intervenors Ex. 29, at A2. One of the inspectors selected for reinspection had inspected for bolt torquing. Tuetken, ff. Tr. 7760, at 8. His work was dropped from the program because it could not be determined which of the torquings he had inspected. Hatfield, with Applicant's assent, declined to inspect in this area because the purpose of the reinspection was to test the inspectors, not the inspections. The concern was possible skewing of the statistical trend. Tr. 7790-91 (Tuetken). As a result there will be no inspector qualification verification on this function.[69] Hatfield could have reinspected a sampling of the torquings in any event to test the general reliability of the inspections, but, again, the decision was made in the direction of fewer reinspections.

D-383. Intervenors raise the concern that not all areas inspected by Hatfield will be reinspected. Proposed Finding 85. Applicant disputes this contention, stating that Mr. Tuetken testified that every area or attribute originally inspected by Hatfield will be covered, citing ff. Tr. 7760, at 8. Reply at 14. This was not Mr. Tuetken's testimony. He merely listed the attributes or areas of inspection which were inspected during the example period by the inspectors selected for reinspection. Since these inspectors were selected at random, and by the Senior Resident Inspector on the basis of their experience (Tr. 7994), there is no assurance that each attribute inspected by Hatfield will be covered in the reinspection program. There is no basis to assume that the random selection just happened to cover every attribute — Mr. Tuetken did not testify that that was the case. *Id.* and ff. Tr. 7760, *passim.*

D-384. On August 4, 1983 the Region III staff met with Applicant to express its concerns that the reinspection at Hatfield might not actually be conducted, and that Applicant is not maintaining a rigorous and dedicated control over the reinspection effort. The Staff's concern is founded on the nature of Hatfield's inspection records. They are maintained, not according to inspector, but according to the type of inspection which leads the Staff to fear that the inspectors in the reinspection program may not take the time and effort to sift through the documentation to determine that they are actually inspecting the work of a selected inspector — a process essential to the statistical integrity of the program. Tr. 7758-59 (Tuetken).

Recertification

D-385. As a result of the 82-05-19 inspection, Region III also concluded that the Applicant needed to apply standardized inspector certification requirements, based on ANSI standard N45.2.6-1978 and Regulatory Guide 1.58, to all Byron contractors. Stanish, ff. Tr. 7549, at 2, 3; Tr. 7969 (Forney).

D-386. Since 1978, Region III had experienced problems with Applicant and its contractors, including, as we noted above, Hatfield, concern-

[69] Another reason for excluding bolt torquing from the program was that there was to be an over-inspection of this procedure by Pittsburgh Testing Laboratories. Tr. 7902 (Tuetken).

ing compliance with the ANSI standards and the Regulatory Guide. As of March 1981, the Applicant had formally committed itself and its contractors to the 1978 ANSI Standards, and had informally committed to them before. Applicant took exception only to the ANSI educational requirements for inspectors. Tr. 7819-23 (Forney). Therefore, as of the 82-05-19 inspection, May 1982, Applicant had not honored its commitment. Applicant Ex. 8; Tr. 7968-69 (Forney). In response to Inspection Report 82-05, Applicant reviewed the parameters set forth in standard N45.2.6-1978 and translated its general guidelines into quantified standards to be met by each contractor. The ANSI standard itself is not sufficiently quantified. Stanish, ff. Tr. 7549, at 3; Tr. 7565 (Stanish).

D-387. Site contractors were required to revise their training and certification procedures to incorporate the standard. Revised procedures of each contractor were submitted for approval to Applicant's Quality Assurance Department. Stanish, ff. Tr. 7549, at 4.

D-388. Hatfield's revision of its certification procedures included setting specific minimum hours (40) of on-the-job training required by Applicant for certification to any procedure. In addition, pursuant to Applicant's directive, Hatfield set a minimum number of questions (40) for certification examinations. Tr. 7565, 7580-82 (Stanish); Tr. 7949, 7950 (Connaughton).

D-389. Each site contractor was directed to review all presently employed inspectors to verify that they met the revised standards. Inspectors who did not satisfy the requirement were to be retrained and recertified by the employing contractor. Stanish, ff. Tr. 7549, at 4.

D-390. Applicant's Quality Assurance Department reviewed 100 percent of contractor inspector certification packages but did not independently verify the accuracy of information contained in the certification documents. *Id.* at 4, 5; Tr. 7633-36 (Stanish).

D-391. Applicant's review of the contractors' inspector certification packages was not a smooth process. The review began in October 1982, but the contractors' records were not in a reviewable format. Applicant stopped its review because it was fruitless and continued again in late February 1983. Tr. 7639-42 (Stanish).

D-392. About half of the sixty to seventy inspectors at Hatfield required retesting because Hatfield had administered examinations containing less than forty questions. At least half of the Hatfield inspectors required additional training because they had not compiled at least 40 hours of *documented* on-the-job training. Tr. 7580, 7582 (Stanish).

D-393. As a result of its audit in June-July 1983, Applicant revoked the certification of two inspectors until verification of their high school education was accomplished. Tr. 7726-27 (Stanish). However, as the

Board noted above (Paragraphs D-357 to D-358), neither Hatfield nor Applicant discovered that a Quality Assurance Manager and a Level II Quality Control Inspector were not properly certified in that they both lacked qualifying experience.

Applicant's Response to the Board's Reopening Order and the Allegations of Fraud

D-394. The Board's Reopening Order of June 21, 1983 directed the parties "to present a full evidentiary showing and explanation of the pertinent investigations of Hatfield Electric's quality assurance program and the subsequent reinspections." *Id.* at 3. In our order of July 7, 1983, we clarified (at Applicant's request) the scope of the reopened hearing. There we stated that the evidence may be limited to Hatfield. The Board also directed the parties to:

> Report the results of Applicant's reevaluation of previously trained/qualified/-certified QA/QC personnel employed by or assigned to Hatfield.
>
> [and]
>
> Report the results of the reinspection program regarding Hatfield Electric installations. (CECo letter of February 23, 1983, Attachment A.)

Of most importance, we also informed the parties that we were particularly interested in any fraudulent training, qualification or certification practices. *Id.* at 2.

D-395. As we noted in the foregoing discussions, Applicant presented the testimony of Mr. Stanish, the Byron Quality Assurance Superintendent (ff. Tr. 7549), and Mr. Tuetken, the Assistant Superintendent for Construction at Byron (ff. Tr. 7760) in the reopened hearing. Intervenors complain (particularly Proposed Findings 47-90) that Messrs. Stanish and Tuetken were not forthcoming in their testimonies. The Board cannot adopt entirely the severe criticisms leveled by Intervenors at the direct testimony of these important officials.[70]

[70] For example, Intervenors' Proposed Finding 57 would have the Board find:

> that Edison's offering of Mr. Stanish as a witness in this hearing, when Mr. Stanish failed to be even minimally familiar with the review and recertification aspect of the 82-05-19 program as it relates to Hatfield Electric Company, is an example of Edison's casual attitude toward this hearing process. The Board is distressed at this apparent attempt to keep the information set forth in the Board's July 7 Order from being heard by presenting evidence through a witness with such minimal knowledge of a specific subject about which an evidentiary showing was specifically ordered.

D-396. However, the Board was troubled and puzzled at the very low information content in the prepared direct testimonies of these witnesses and in Mr. Stanish's oral testimony. In particular, Mr. Stanish not once referred to Hatfield Electric Company in his written direct testimony. The Board accepted Mr. Stanish's general testimony, over Intervenors' objections, because, in his brief discussion of inspector recertification programs for all affected Byron contractors, Hatfield's recertification program was necessarily subsumed. But his direct testimony conveyed little information of value.

D-397. On cross-examination, Mr. Stanish was uncertain whether Hatfield was required to do additional work to comply with his department's new certification requirements. Tr. 7562-63. He could not identify all the changes made by Hatfield in its recertification efforts. He recalled only the changes respecting 40 hours of on-the-job training and forty-question examinations. Tr. 7566-68. The best estimate he could give concerning the number of Hatfield inspectors who required retesting was "probably" half. The number of inspectors requiring additional documented on-the-job training was said to be "at least" half. Tr. 7580. Applicant's quality assurance audit of the contractors' reinspection and recertification programs, including Hatfield, was conducted under Mr. Stanish's supervision and was reviewed by him. Yet neither the very material report nor the relevant information it contains would have been in evidence had the Intervenors not requested it from him.[71]

D-398. For his part, Mr. Tuetken directly and fully addressed the reinspection program at Hatfield but made no mention of the Hatfield inspector recertification program. This is understandable because he is a production manager. But the fact remains that the Hatfield inspector certification aspect of the Board's order was ignored by Applicant in the reopened proceeding except to the sparse incidental references to it by Mr. Koca, the Hatfield Quality Assurance Supervisor, called by Applicant to refute Mr. Hughes' allegations. Mr. Koca's testimony (ff. Tr. 7418) was narrowly confined to Mr. Hughes' allegations as was discussed above.

D-399. Our June 21 order reopening the hearing explicitly broadened the issue beyond the Hughes' allegation to encompass the allegations of other individuals referred to in Region III testimony during the main hearing relative to the issue of alleged fraudulent training, testing and certification practices. As Applicant has known at least since early in the main hearing, these allegations have been and are still under investigation and continuing inspections by the Office of Investigations and Region III, respectively. Region III Testimony, ff. Tr. 3586, at 6.

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[71] The Board does not suggest that the audit report was concealed. Mr. Stanish identified it in general terms in his direct testimony. Ff. Tr. 7549, at 5-6.

D-400. Mr. Koca was an appropriate, albeit interested, witness to answer the allegations made against him personally and against his employer. But the Board does not regard him as being an adequate witness to represent Applicant's entire position on this serious issue. No Commonwealth Edison official addressed the Board's broader concern about the general integrity of the Hatfield training and certification procedures. Neither Mr. Stanish nor Mr. Tuetken even referred to the Board's request to be informed about allegations of fraud in their direct testimonies. In fact, Applicant's counsel initially made very strong objections to Intervenors' cross-examination of Mr. Stanish on alleged fraudulent practices on procedural grounds that there was inadequate foundation for such questioning. Tr. 7649-53.

D-401. As far as the Board can determine, the Applicant's entire effort to address the allegations of fraudulent certification practices and the Board's respective order is summed up in the following exchange between Intervenors' counsel and Mr. Stanish:

> Q. Mr. Stanish, has Commonwealth Edison done anything to see whether Hatfield Electric Company engaged in any fraudulent training qualification or certification practice?
>
> A. Yes, we have been involved with their training program, in that in order to verify that inspectors have, in fact, attended training sessions, we do frequently attend Hatfield's training sessions.
>
> We also review on-the-job training as it is being performed by inspection candidate and his trainer. Those are the types of activities we do get involved with.
>
> Q. But are these the only two activities that you undertake?
>
> A. Those are the only two that come to mind right now.

Tr. 7659.

D-402. Mr. Stanish testified also that in reviewing contractors' inspector certification packages (*i.e.*, files) his department accepted the certification documentation at face value, depending upon the contractor's quality assurance supervisor or manager for the integrity of the packages.[72] Tr. 7635 (Stanish). The Board was concerned the Applicant might have felt restrained from making its own fraud-type investigation into Hatfield's training, testing and certification practices because of the pending NRC investigations, but Mr. Stanish assured the Board that

[72] Mr. Koca, the Hatfield Quality Assurance Supervisor, testified that his company, in turn, accepts the certification of yet another employer as evidence of an inspector's experience without inquiring into the previous employer's foundation. Tr. 7428-29.

pending NRC investigations were not a part of his consideration. Tr. 7660.

D-403. The Board does not suggest that Applicant's officials have uncovered evidence of fraud in Hatfield's quality assurance program and that this information has been willfully withheld from the hearing. Mr. Stanish expressly stated that, in all of his experiences in dealing with Hatfield, including direct and indirect participation in audits, he has not received any indications of fraudulent practices. Tr. 7739. Furthermore, the Board has no basis on this record upon which it can find that fraudulent practices existed at Hatfield. But, we had made the determination, within our authority to do so, that the allegations and investigations concerning Hatfield constituted good cause to inquire further in this adjudication. Our concern is that the Applicant has done nothing of any significance to address the issue and has imparted a general sense of disinterest to the Board. The best that can be said for Applicant's response to the inspector certification integrity issue is that Applicant neither found nor looked for indications of certification fraud.

D-404. While the Board recognizes that the Applicant opposed reopening the hearing, Applicant was nevertheless obliged to comply with the Board's order to present evidence on the issues set out in our order, to petition for reconsideration by the Board, or to seek appellate review of that order. It did none of these. Our conclusion is that Applicant's evidentiary response to the issue in the reopened hearing has been weak and borders default.

### NRC Staff's Position on the Reinspection Program

D-405. The reinspection program is very important to the quality assurance contention and to Region III's licensing responsibilities. Region III states that it "is a very extensive and comprehensive program that looks at almost all of the work that has been completed at that plant that is safety-related." Tr. 7956 (Hayes). No other Staff finding at Byron has resulted in a reinspection program of the magnitude of the 82-05-19 program. Tr. 7868-69 (Forney).

D-406. The reinspection program's importance is manifested by the fact that it is being relied upon by Region III to make the basic empirical determination of the qualifications of the contractors' inspectors and whether their work was deficient. Tr. 7843, 7964 (Forney). The Staff will also use it to evaluate the workers themselves — for example, a welding defect would imply that a welder may not have been properly qualified. Tr. 7980 (Forney). To a large extent the reinspection program is relied upon by the NRC Staff to dispose of some of the pending worker allegations about the quality of work at Byron, particularly welding work by Hatfield. Tr. 7954 (Connaughton); Tr. 7809-10, 7955-56 (Hayes). Most of the worker allegations remaining to be investigated do not involve significant hardware problems, but those that do will, for the most part, depend upon the reinspection program for resolution. Region III Testimony, ff. Tr. 7801, at 22.

D-407. Region III will not recommend that a Byron operating license issue until the reinspection program is completed, and the results are evaluated and found to be acceptable. Tr. 7858-59 (Forney). Moreover, all of the pending worker allegations must be resolved, either by the reinspection program or by additional inspections, before the Staff will issue a Byron operating license. Tr. 7882 (Forney); Tr. 7809-10, 7962 (Hayes).

D-408. Region III is concentrating on potential hanger-weld defects in Hatfield's work at Byron partly as a result of worker allegations. *See* Paragraph D-364, *supra*. Tr. 7806-07. As we noted above, as of June 1983, there were 818 defects in 7,753 weld attributes reinspected. Failure of a hanger single weld or the hanger itself would not necessarily lead to a support system failure, but it could be that the hanger weld defects are more serious than is presently recognized by Region III. Weld defects may exist on a series of hangers on a single support system which could affect the ability of that system to perform the design function. Tr. 7809 (Hayes). Region III believes there may be a problem in a few areas and Applicant may be called upon to demonstrate that welds meet design requirements either by reworking or by engineering evaluation. Tr. 7982 (Hayes).

D-409. The NRC Staff was requested by the Board to testify concerning its position on the adequacy of the reinspection program.[73] The Staff was not able to provide assurances to the Board that the reinspection program was adequate. Region III witnesses testified that a final determination whether the reinspection program is successful will not be made until up to three months after the Applicant reports its results and evaluation of the reinspection program. Ff. Tr. 7801, at 7. Staff's review may require additional corrective actions and verification inspections. *Id.* at 21-22.

D-410. As we noted above, the Staff has, to date, accepted only the basic premise of the Applicant's reinspection program. Tr. 7981 (Forney). The Staff may require even greater second-round expansion of failed inspectors' work for reinspection than that proposed by the Applicant and discussed by Mr. Tuetken. Tr. 7980, 7987 (Forney). Also,

[73] Memorandum and Order, July 7, 1983, *supra*.

Region III in all likelihood will take exception to Applicant's criterion of 90 percent acceptability rate for subjective inspection attributes. Tr. 7999 (Forney). Moreover, Region III officials have yet to reach agreement with Applicant about the definition of subjective weld attributes. *E.g.*, Tr. 8001; *see generally* Tr. 7997-8006 (Region III Panel).

D-411. The relevance and importance of the reinspection program at Byron to the licensing review of the Byron plant, and to the quality assurance litigation is, and has been, obvious. Yet, when the Region III Staff first presented its prepared direct testimony on the quality assurance contention on April 7, 1983 it made no mention whatever of a reinspection program. Ff. Tr. 3586. Attachment A to the Staff's early testimony is a sixty-page Byron inspection chronology consisting of a listing of Region III's inspections at Byron since March 1978. On page 38 of the chronology was the terse notation: "Noncompliance (IV) 454/82-05-19 455/82-04-19 QA/QC supervisors & inspectors not adequately qualified and/or trained" — nothing more.

D-412. It was not until after the close of the evidentiary record when the NRC Staff, opposing Intervenors' motion to reopen the record on John Hughes' allegations, informed the Board and the parties by affidavit that Region III depends, in part, upon "this comprehensive reinspection program" to resolve not only the allegations of Mr. Hughes, but the more specific allegations of his co-allegers alluded to in the earlier testimony.[74]

D-413. The Applicant presented a brief discussion of the 82-05-19 inspection findings in its testimony of Mr. Shewski during the main hearing. Mr. Shewski emphasized the favorable findings in the inspection report, alluded to the standardization of the inspector certification process for Byron contractors and made a general reference to a sample reinspection plan acceptable to the NRC to resolve its inspector certification concerns. Shewski, ff. Tr. 2364, at 31-35.

D-414. In the Board's view, the Applicant's presentation was inadequate on the issue — such inadequacies are one of the basic reasons for adverse-party adjudicatory hearings. The Staff's original presentation, totally ignoring the recertification and reinspection program, has never been explained by the Staff nor understood by the Board.

D-415. Subsequently, however, after losing its appeal on the issue of presenting evidence on pending inspections and investigations, the Staff has made what we perceive to be a diligent effort to explain fully the significance and details of the certification and reinspection programs in

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[74] NRC Staff Response to Joint Intervenors' Motion to Reopen the Record, May 9, 1983, Affidavit at 5, 6 and 10.

compliance with our reopening order. The explanation for Staff's initial silence on the issue lies perhaps in its perception of the respective roles of the NRC Staff and adjudicating boards in the licensing process, as we discuss below in the next section.

D-416. In sum, during the reopened hearing we learned from the parties, particularly the NRC Staff, that:

- There was a very thorough inspector recertification program imposed on the Byron contractors by the NRC Staff through the Applicant.
- In response to the 82-05 inspection, the Applicant instituted an extensive, comprehensive, and apparently unusual reinspection program.
- The Staff depends upon the reinspection program to determine whether the contractors' inspectors were qualified, whether their inspections were adequate and whether some of the production workers, particularly welders, were qualified.
- The Staff relies upon the reinspection program to resolve some of the worker allegations about the quality of work at Byron, especially the quality of the welding by Hatfield.
- The reinspection program is being employed to identify work for correction or reevaluation.
- The NRC Staff cannot now provide assurance that the reinspection program will satisfy its concerns about the qualifications of inspectors and the quality of the work at Byron.
- Only the basic concept of the program has the Staff's approval. Final approval may require an expansion of the reinspection sampling and further agreement on the standards for reinspection.
- The NRC Staff regards the pendency of the reinspection program as a basis to withhold an operating license for Byron.

D-417. Nevertheless, the Staff, in its proposed findings on the quality assurance contention, proposes that the Board find that, contrary to the contention, Applicant has the ability and willingness to comply with Appendix B to Part 50. Thus, Staff would have the Board decide the quality assurance contention in favor of issuing an operating license for Byron on the basis of the record presently before us.

Delegation to Staff

D-418. The Staff's initial slighting of the issue of the reinspection program, and its present position that the matter can be left to the Staff for post-hearing verification, can be explained, we believe, by what appears to be the Staff's misunderstanding of the respective roles the Staff and the licensing boards play in the licensing process. The Staff appears to think the Board can delegate to it the responsibility of deciding the essence of the issues raised by the contention on quality assurance. We, however, do not think so.

D-419. In *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-298, 2 NRC 730, 737 (1975), the Appeal Board ruled:

> When governing statutes or regulations require a licensing board to make particular findings before granting an applicant's requests, a board may not delegate its obligations to the staff. The responsibilities of the boards are independent of those of the staff under the Commission's system, and the boards' duties cannot be fulfilled by the staff, however conscientious its work may be.[18]

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[18] *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Station), ALAB-124, 6 AEC 358, 360, 361-62, fn.4 (1973). *See also Washington Public Power Supply System* (Hanford No. 2 Nuclear Power Plant), ALAB-113, 6 AEC 251, 252 (1973).

D-420. The Commission long ago made the definitive statement of the non-delegation rule in the context of operating license hearings:

> As a general proposition, issues should be dealt with in the hearings and not left over for later (and possibly more informal) resolution. See this Commission's decision in *Wisconsin Electric Power Co.* (Point Beach Unit 2), RAI-73-1, p. 6 [CLI-73-4, 6 AEC 6 (1973)]. In some instances, however, the unresolved matter is such that Boards are nevertheless able to make the findings requisite to issuance of the license.[8] But *the mechanism of post-hearing resolution must not be employed to obviate the basic findings prerequisite to an operating license* — including a reasonable assurance that the facility can be operated without endangering the health and safety of the public. 10 C.F.R. 50.57. In short, the "post-hearing" approach should be employed sparingly and only in clear cases. *In doubtful cases, the matter should be resolved in an adversary framework prior to issuance of licenses*, reopening hearings if necessary. [Emphasis supplied.]

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[8] For example, a Board might, after hearing, find an applicant's security plan adequate, except for minor procedural deficiencies. In such a case, the Board could choose to authorize issuance of a license — with the deficiencies to be subsequently cured under the scrutiny of the Director of Regulation.

*Consolidated Edison Co. of New York* (Indian Point Station, Unit 2), CLI-74-23, 7 AEC 947, 951-52 (1974).

D-421. The Commission has also held that the rule against delegation shall apply even to issues a licensing board raises on its own motion in an operating license proceeding:

> Nor would it be an adequate solution, as the applicant and the regulatory staff suggest, to have a Licensing Board which spots an issue merely refer the matter to the staff for resolution. The regulatory staff, to be sure, plays a critical role in this agency's procedures, even aiding our Boards in resolving issues. [citations omitted] But when a Board uncovers an issue, we expect *it* to resolve the matter openly and on the record, after giving the parties (which includes the staff) an opportunity to comment or otherwise be heard. [Emphasis in original]

*Consolidated Edison Co. of New York* (Indian Point Nuclear Generating Unit 3), CLI-74-28, 8 AEC 7, 8-9 (1974).

D-422. By citing *Point Beach* in *Indian Point*, CLI-74-23, the Commission indicated both what the ground of their rule was and how firmly they were committed to it. In *Point Beach*, the Licensing Board had authorized a full-power operating license subject, among other things, to a condition that, before 20 percent power is exceeded, the Staff would resolve a fuel rod problem which the intervenors had identified post-hearing. *See* LBP-72-32, 5 AEC 162, 204. At first, the Appeal Board remanded the proceeding. ALAB-86, 5 AEC 376, 379. But later, the Appeal Board authorized temporary operation at 75 percent, reasoning, in part, that the fuel rod problem would not occur during such restricted operation. ALAB-90, 6 AEC 11, 13, 16.

D-423. The Commission, however, pursuant to 10 C.F.R. 2.786(a), under which the Commission reviews Appeal Board decisions or actions "in cases of exceptional legal or policy importance" (*id.*), said

> [H]owever reasonable or logical that result may have appeared to the Appeal Board, it does not adequately take into account the demands of the Atomic Energy Act and the Administrative Procedure Act. Those statutes provide that whenever an agency is required to conduct an adjudicatory hearing on an operating license application, all parties have the right to an opportunity to participate in the resolution of properly contested issues.

*Wisconsin Electric Power Co.* (Point Beach Nuclear Plant, Unit 2), CLI-73-4, 6 AEC 6, 7 (1973).

D-424. The Commission directed that the Appeal Board's authorization of restricted operation be stayed pending completion of the remand proceeding before a licensing board. *Id.*

D-425. Applied to the quality assurance contention in the Byron proceeding, the rule against delegation would appear to require that the Board decide, rather than the Staff decide, when the reinspection program is adequate. The contention raises "properly contested issues" (*Point Beach*, 6 AEC at 7) about difficulties which are not simply "minor procedural ones" (*Indian Point*, 6 AEC at 951 n.8). Moreover, we see no way yet to manage the resolution of these difficulties in stages so as to delegate any later stages to the Staff. For example, the Staff and the Applicant haven't agreed yet on a full set of standards for the reinspection program. Nor can the Staff yet provide assurance that the reinspection program will satisfy its concerns. Therefore we are not in a position to say that the reinspection program will reliably test the Applicant's ability to maintain an adequate quality assurance program. At the very least, the quality assurance issue is one of those "doubtful cases" the Commissioners have said should be "resolved in an adversary framework prior to issuance of licenses." *Indian Point,* CLI-74-23, 6 AEC at 952.

D-426. Our application of the rule against delegation of quality assurance issues in particular would appear to be confirmed by the Appeal Board in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-124, 6 AEC 358 (1973), cited in *Perry, supra.* There the Appeal Board noted that both the Licensing Board and the Staff were concerned that the Applicant's quality assurance program did not fully comply with the Commission's regulations (6 AEC at 360, 361), but the Appeal Board rejected the Staff's notion that the Licensing Board should have left it to the Staff to resolve the outstanding issues off the record. *Id.* at 362 n.4. Instead, the Appeal Board remanded the proceeding to the Licensing Board. *Id.* at 366-67.

D-427. The Board is mindful that our position with respect to post-hearing verification in the quality assurance issue is in bright contrast to our acceptance of predictive findings and post-hearing verification of the formulation and implementation of many aspects of the Byron emergency plans. As we stated with respect to our citation to *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1103 (1983), *supra,* the Appeal Board there noted that emergency planning issues differ from other issues in hearings, particularly as a result of the amendment to 10 C.F.R. 50.47(a)(1) in July 1982 respecting the predictive nature of emergency planning findings in adjudications. However, the *Waterford* Appeal Board noted its continued and current agreement that issues not involving emergency planning should be dealt with during hearing and not left over for later and possibly more informal resolution except, for example, where minor procedural deficiencies are involved. *Id.*

D-428. It is probably quite evident by now that this Board will not delegate to the Staff the task of determining post-hearing that the Hatfield reinspection program is adequate. This conclusion and the application of the rule against undue delegations with respect to other contractors is one of the essential elements of our ultimate conclusions on the quality assurance contention.

## 6. *Board Conclusions on Quality Assurance*

D-429. The Board concludes that the Intervenors prevail on the essence of the quality assurance contention. Applicant has not, in the language of the contention, demonstrated its "ability or willingness to comply with 10 C.F.R. Part 50, Appendix B, to maintain a quality assurance and quality control program, and to observe on a continuing and adequate basis the applicable quality control and quality assurance criteria and plans . . . ."

D-430. Before the Board sets out the particular factual reasons for this conclusion, those reasons should be considered against the background of the quality assurance litigation and within our perception of the role of adjudicating boards in operating license proceedings. This exercise is necessary for an understanding of the significance and reach of our conclusions.

D-431. As we stated at the outset of the section on the Byron contractors, there is a strong element of randomness in this quality assurance litigation. Worker allegations coming to the Board's attention were necessarily random. The Staff inspection program is based on sampling and audit and, in turn, the details of the Staff inspections did not come to the adjudication in a well-organized manner. We considered in detail only those matters the Intervenors happened to select or the Board happened to notice and deem important to our decision. The quality assurance litigation was not a systematic or complete review of the quality assurance programs of the Applicant and its contractors at Byron.

D-432. In this operating license proceeding, as in others, the Staff, in discharging its licensing obligations, considers many matters not at issue in the adjudication. Some of these functions, including the Staff's quality assurance inspection program, may and probably do, involve considerations at least as important as those that happen to be adjudicated. The resolution of those considerations, no matter how important, is a Staff function, because we do not oversee the Staff's work. Therefore leaving those matters to the Staff is not a delegation because they were not in our sphere of matters to decide. Other matters, especially the quality assurance performance of Hatfield Electric Company, were thoroughly considered in the adjudication, and as we noted in the preceding section, respective problems may not be delegated to the Staff for post-hearing resolution. Still other contractors, or the particular activities thereof, in examples noted below, came to our attention more-or-less incidentally. Whether post-hearing resolution of remaining problems with the incidentally considered matters can be delegated to the Staff is a highly judgmental consideration.

D-433. Therefore, in deciding whether Applicant has met its quality assurance obligations, *as an issue in this proceeding*, the Applicant may be the coincidental beneficiary or victim of whether a particular set of facts came to be litigated and whether problems identified in the litigation may be delegated to the Staff for post-hearing determination. As it turned out, despite the random nature of the litigation, enough information was considered for the Board to conclude that the insufficiencies in the quality assurance programs of the Byron contractors demonstrate an inadequate quality assurance program in Applicant's organization and that the resultant problems cannot all be delegated for resolution.

D-434. The Board does not have confidence that the quality of the work at Byron by Hatfield Electric Company is adequate to provide reasonable assurance that the Byron facility can be operated without undue risk to the public health and safety. The long and bad quality assurance history of Hatfield at Byron persuades the Board that the Applicant has not discharged its responsibility to assure that Hatfield's quality assurance program is effective. Applicant seems to have begun to meet its quality assurance responsibilities with respect to its Byron contractors very late. With respect to Hatfield, at least, we do not have assurance that even today Applicant has met those responsibilities.

D-435. A reinspection program seems to be a logical method by which doubts about Hatfield's quality assurance program can be resolved. But we cannot find, in part because the Staff does not find, that the reinspection program is sufficient to assure that Hatfield's work is good enough. In addition to the Staff's stated conclusions to this effect, the Board is concerned about several unexplained aspects of the reinspection program.

D-436. Who, if anyone, has decided that the reinspection of 25 percent of the Hatfield inspectors is a statistically significant and reliable sample? The Staff apparently won't decide the sampling adequacy until it reviews the complete reinspection program for sufficiency. At least half of the Hatfield inspectors were found to need retesting and about half needed more on-the-job training, but not all of these inspectors' work is being reinspected. Nor is every attribute of the original inspections being sampled.

D-437. Some of the previous inspections are not accessible or re-creatible. A statistically reliable reinspection sampling could provide assurance that the inaccessible and non-re-creatible inspections were adequate, but that assurance has not been provided in this record.

D-438. We are most concerned that Hatfield seems to be perpetually incapable of maintaining reliable records of nonconforming and deviating conditions. Applicant's reinspection audit revealed that Hatfield, as well as other contractors, were not issuing discrepancy reports in the reinspection program, but were correcting work with field-problem documentation instead. This raises the possibility that the trend analysis may be nonconservatively skewed. This is simply not acceptable and reduces still further our confidence in the reinspection program at Hatfield. We are also concerned that, despite all of its troubles, Hatfield still has not developed a practice of carefully assuring and documenting that its inspectors are qualified.

D-439. Apart from the Hatfield reinspection program, we cannot overlook the fact that, as a result of worker allegations against Hatfield, there are several matters still pending with Region III and the Office of Investigations. Most worker allegations have not been substantiated by Region III after careful inspection — a phenomenon we ourselves have observed in this adjudication. Still, some important allegations have been accurate, and they cannot be ignored. This aspect of the hearing process presents problems beyond our power presently to solve with fairness to all. Region III states that the Staff will not authorize Byron to operate until the allegations are inspected or investigated by the cognizant Commission office. This may be an indication of the potential gravity of the allegations, or it might be simply another indication of the Staff's caution.

D-440. Certainly the pendency of the NRC inquiries into the workers' allegations, with nothing more, cannot fairly be a basis for deciding the quality assurance contentions against Applicant and we do not do this. Were it only a question of the outstanding inquiries, the Board would seek either their prompt completion or assurances that either they involve delegable Staff matters, or that they present no safety consideration. The background to and the inadequacies of the Hatfield reinspection program standing alone are sufficient for the Board to rule against Applicant on the contention; the pendency of the inspections and investigations, as they are described in our findings, is simply added concern.[75]

D-441. We conclude that the record concerning Hatfield's inadequate quality assurance program and the attendant circumstances, standing alone, is sufficient to find against Applicant on the quality assurance

[75] On August 9 and 10, 1983 the Board heard from representatives of the Office of Inspection and Enforcement, Region III, and the Office of Investigations, *in camera* and *ex parte*, to learn the status of pending inspections and investigations. We determined that some of the inspections are of no further interest and all of the inspections and the investigations were in stages too early to produce reliable results. Memorandum and Order, LBP-83-51, 18 NRC 253 (1983). Subsequently, and prior to August 26, we again reviewed the transcript of the *in camera, ex parte* session in connection with disclosing nonconfidential portions. The Board has not since reviewed that transcript and we do not use that information in this decision.

issue. Our concern does not stop with Hatfield, however. Several other contractors of safety-related work at Byron have inadequate or questionable quality assurance programs.

D-442. We concluded that the Systems Control Corporation quality assurance program broke down, was unreliable and fraudulent and that Applicant defaulted in its respective oversight responsibility. The inquiry by the Department of Justice into alleged fraud at Systems Control was pending at the close of the record. Problems with Systems Control were still open items with Region III. The Board noted that the 100 percent reinspection of Systems Control work may remove the matter from a direct safety concern. This factor, the reinspection of all of Systems Control's work, which by its nature is accessible for reinspection, points to a somewhat different conclusion than the Hatfield situation. The results of the reinspection can be evaluated by the Staff as a matter of routine procedure as a delegable function. There is nothing left to adjudicate with respect to Systems Control. We allude to the Systems Control experience, however, because it adds additional support to our conclusion that Applicant's quality assurance oversight of its contractors, without more, is not sufficient protection of the public safety.

D-443. Our findings with respect to Reliable Sheet Metal are about the same. Reliable's inadequate quality assurance program is a reflection on Applicant's program. However, the adequacy of the 100 percent reinspection of Reliable's work is a matter appropriately delegated to the Staff.

D-444. We concluded that the 82-05 reinspection program on Hunter Corporation's work, if effective, is essential to a verification of Hunter's quality assurance program. Applicant's 1983 audit of Hunter's reinspection established that Hunter was not employing the documentation for nonconforming conditions needed to test the reliability of the reinspection. The record on Hunter is another reason why the 82-05 reinspection program is a matter too uncertain to delegate to post-hearing Staff verification.

D-445. Our conclusion with respect to Blount Brothers is somewhat different. Blount is one of the contractors subject to the reinspection program. After a large litigation on the allegations of Messrs. Gallagher and Stomfay-Stitz, and looking at the performance history available to us, we concluded that there is insufficient basis to conclude that Blount's quality assurance program was inadequate. The most significant record information adverse to Blount is the Staff's very brief finding that there was a flaw (of no apparent importance) in the certification file of one of the two inspectors checked in the 82-05 Region III inspection. Applicant Ex. 8, at 69. Nevertheless Blount was caught in Applicant's reinspection commitment, apparently for reasons not in evidence. We conclude that the effectiveness of Blount's reinspection program is a matter properly left to the Staff for its post-hearing determination, because the question is not just whether the particular matter is more than a minor procedural one — it is also whether it is within the purview of the Board.

D-446. In Blount's case, even though we had jurisdiction over the general subject of its quality assurance program by virtue of the contention, the Blount reinspection program and related events were not addressed in the litigation before us. Thus, leaving it to the Staff to handle Blount's reinspection post-hearing is not a delegation at all; it is a matter which did not rise to the level of being an issue in our hearing.

D-447. The Board arrives at a similar conclusion with respect to Powers-Azco-Pope and Johnson Controls. The Staff insisted upon a 100 percent reinspection sample of these two contractors because there were large-scale failures in their respective inspector qualification procedures. But almost nothing about them was litigated. In what we acknowledge to be a judgment call, we conclude that the Powers-Azco-Pope and Johnson Controls quality assurance programs were not significant issues in our proceeding. Therefore, leaving the results of their reinspection programs to the Staff is not a delegation. Moreover, even if these two contractors' programs were at issue before us, the fact that all of their work is to be reinspected would render the acceptability of those reinspections a procedural matter properly delegated to the Staff, as we found with respect to Systems Control and Reliable Sheet Metal above.

D-448. In sum we have concluded that Applicant's quality assurance performance with respect to Hatfield Electric and Hunter Corporation has been inadequate and resolution of those matters may not be delegated to the Staff to resolve after the hearing. Applicant's performance with respect to Systems Control Corporation and Reliable Sheet Metal was inadequate, but their reinspection programs may be left to the Staff to consider post-hearing. The facts surrounding Systems Control and Reliable Sheet Metal, however, support the Board's ultimate conclusion in our order below. Although aspects of the quality assurance program of Blount Brothers were litigated, Applicant prevailed on those aspects. The circumstances of the Blount reinspection program were not in issue. Although there were summary indications of large-scale failures in inspector qualifications at Powers-Azco-Pope and Johnson Controls, their quality assurance programs were not issues considered by the Board, and, in any event, their 100 percent reinspection program would be a matter appropriately delegable to the Staff.

D-449. Contrary to the implications of the contention, we do not conclude that Applicant is institutionally incapable or unwilling to maintain an adequate quality assurance program. Although the underlying reasons for Applicant's failures with respect to the contractors' quality assurance programs were not litigated during the hearing, we believe that the record as a whole indicates that the very large quality assurance task at Byron simply got ahead of Applicant's quality assurance organizations. It may be a matter of timing. As the evidence unfolded at the hearing, Applicant was catching up.

D-450. Finally, the Board notes again its earlier conclusion that the various quality assurance organizations within Applicant's corporate structure were suitably designed to carry out their functions; that they possess sufficient independence from costs and scheduling considerations, and that Applicant prevailed on that aspect of the quality assurance contention charging insufficient independence of the quality assurance function.

## E. Groundwater Pathway

E-1. Two League of Women Voters' Contentions about contamination of groundwater by radionuclides were admitted for litigation. By stipulation dated December 6, 1982, these two contentions, Nos. 39 and 109, were revised and consolidated and now read as follows:

> Since the groundwater system underlying the Byron Nuclear Power Station site has not been characterized adequately, the consequences of radionuclide releases to the underlying aquifer cannot be predicted with confidence. In consequence, no proper NEPA analysis of this important subject can be made. In addition, as a result of this serious and unresolved problem, the findings required by 10 C.F.R. 50.57(a)(3)(i), 50.57(a)(6) and 10 C.F.R. 50.34(b)(4) cannot be adequately made.

E-2. The League is chiefly concerned that the Staff and the Applicant have underestimated the velocity with which radionuclides released to the groundwater under the Byron plant by a major accident would travel to points where humans draw from the groundwater system for their uses. *See* Wood, ff. Tr. 6879. Only if the contaminants travel slowly enough can engineers stop them or the contaminants decay to safe levels before humans come in contact with them.

E-3. The League was moved to question the Staff's estimation of contaminant velocity partly by reliable documents which show that traces of cyanide dumped at the Byron site (before the Applicant owned it) had traveled to wells between one and two miles from the site with a velocity twenty times the groundwater velocity calculated by the Staff. *Id.* at 5.

E-4. The League contends that the reason why the Staff may have underestimated the velocity of the contaminants is that the Staff has inadequately characterized the geology of the groundwater system under the Byron site. The means for calculating the velocity of contaminants in a given groundwater system depend on the geology of the system. Velocity in uniformly porous bedrock is easily calculated by using Darcy's equation, but that equation cannot be used to calculate velocities in bedrock in which, for example, there are fractures centimeters wide and kilometers long. Therefore, a good estimation of contaminant velocity, and thus a good estimation of the risks certain accidents pose to water supplies, depend on an adequate characterization of the geology of the system. The League contends that the Staff and the Applicant have not adequately studied the extensive fracturing of the limestone in the area of the plant. League's Proposed Finding 12.

E-5. The League contends that as a result of what it claims is an inadequate characterization of the groundwater system under Byron, the Board must make two rulings. The first of the two is that the Staff has performed improperly its obligation under the National Environmental Policy Act of 1969, 42 U.S.C. § 4321, *et seq.* (NEPA), under which the Staff must prepare Final Environmental Statements (FES) for all construction permit and operating license proceedings. *See* 10 C.F.R. Part 51.

E-6. Several sections of the FES for the operating license proceedings for Byron (Byron FES-OL), NUREG-0848, Staff Ex. 2, consider the impacts on the groundwater system at Byron of both normal operation of the plant and accidents. Section 5.9.4.5(5) sets out numerical estimates of the impacts radionuclides released into the groundwater under the plant by a core meltdown would have on the water supplies around Byron. Those impacts and their costs depend to a great extent on how quickly the radionuclides can travel through the groundwater system. Thus, one of the most crucial numbers in this section of the FES is the number of years the Staff calculates it would take these radionuclides to travel from a point underneath the reactor core to an offsite spring which would carry the radionuclides to the Rock River. If that number is grossly wrong, then the FES calculation of the costs and benefits of operating the plant may have to be revised.

E-7. The League also contends that if the Board finds that the groundwater system at Byron has not been adequately characterized, the Board must also rule that the requirements of three of the Commission's Regulations, 10 C.F.R. 50.57(a)(3)(i) and (a)(6), and 50.34(b)(4), have not been met. Section 50.57(a)(3) requires that there be "reasonable assurance . . . that the activities authorized by the operating license can be conducted without endangering the health and safety of the public." Section 50.57(a)(6) requires that the "issuance of the license will not be inimical to the common defense and security or to the health and safety of the public."

E-8. Section 50.34(b)(4) requires the Applicant to submit with its application for an operating license a final safety analysis report (FSAR), which, among other things, must assess the risk which operation of the plant poses to public health and safety, determine margins of safety during normal operations and anticipated transient conditions, and determine the adequacy of structures, systems and components for mitigation of consequences of accidents. If 50.34(b)(4) is not met, then 50.57(a)(3)(i) and (a)(6) are not met either.

E-9. One of the accidents considered in the FSAR is related to the issue raised by this contention. Section 2.4.13.3 of the FSAR analyzes the consequences of a rupture in one of the plant's boron recycle holdup tanks. One of these tanks can hold 125,000 gallons of a boron solution containing fission products introduced into the primary loop coolant by contact with the reactor core during normal operation. Lahti, ff. Tr. 6750, at 3-4; Tr. 6835-36 (Lahti). The FSAR analysis assumes, among other things, that the contents of one such tank could leak into the groundwater under the plant through a substantial crack in the floor of the auxiliary building, which houses the tank. In order to comply with Commission rules, the FSAR analysis must show that concentrations of radionuclides in the water at points where humans draw it out for their uses will not exceed the limits set out in 10 C.F.R. Part 20, Appendix B, Table II, Column 2. *See* 10 C.F.R. 20.106(a), and NUREG-0800 (*Standard Review Plan*), § 15.7.3, Acceptance Criterion 2, at 15.7.3-2.

E-10. Whether the radionuclide concentrations exceed these limits or not depends in part on whether the radionuclides travel slowly enough through the groundwater system to decay to safe levels before humans use the contaminated water. Thus the adequacy of the FSAR analysis, like the FES NEPA analysis of a meltdown, depends on an adequate characterization of the groundwater system. If that characterization is inadequate, then the velocity of radionuclides released by a boron tank rupture cannot be predicted, and there is no reasonable assurance that health and safety would not be endangered by that accident.

E-11. All the parties presented testimony by witnesses well qualified to speak to one or more crucial aspects of assessing the consequences of releases of radionuclides to the Bryon groundwater system. The Applicant presented testimony by three witnesses: George C. Klopp, a

General Design Engineer with Edison, testified on the adequacy of the FES's assessment of the consequences of contamination of the Byron groundwater system by a core meltdown. Mr. Klopp has had a great deal of experience as a member of both private and government groups working on risk assessment. Lawrence L. Holish, head of the Geotechnical Division of Sargent & Lundy, the firm which is the architect-engineer of the Byron plant, testified about the methods used to study the groundwater system at Byron, and about the reasoning behind the FES's and FSAR's estimates of the velocity of radionuclides in the groundwater system. Mr. Holish has taken part, often as supervisor, in the design of the foundations of forty power plants, fifteen of them nuclear. Gerald P. Lahti, who as Assistant Division Head of Sargent & Lundy's Nuclear Safeguards and Licensing Division supervises the Shielding and Radiological Safety section of that Division, testified about the radiological consequences of a rupture of one of the boron recycle holdup tanks at Byron and release of its contents into the groundwater under Byron. Mr. Lahti has been evaluating radiation hazards and designing radiation protection since 1963. He wrote the radiological part of Section 2.4.13.3 of the Byron FSAR, which estimates the consequences of a rupture of one of the plant's boron tanks.

E-12. The NRC Staff presented joint testimony by two witnesses: Dr. Richard Codell, Senior Hydraulic Engineer in the Hydrologic and Geotechnical Engineering Branch of the Division of Engineering in the Office of Nuclear Reactor Regulation; and Gary Staley, Hydraulic Engineer in the same branch. Dr. Codell and Mr. Staley testified about the adequacy of the characterization of the groundwater system at Byron, and about the use of Darcy's equation to calculate the velocity of radionuclides in the groundwater at Byron. Each of the two men has nearly 10 years' experience analyzing the effects a nuclear power plant and the water in the environment of the plant can have on each other. Together, Dr. Codell and Mr. Staley wrote Section 5.9.4.5(5) of the Byron FES-OL (NUREG-0848), Staff Ex. 2. That section contains the Staff's analysis of the consequences of contamination of the Byron groundwater system by a core meltdown. Dr. Codell made a significant technical contribution to NUREG-0440, Liquid Pathway Generic Study (1978), which deals generically with the subject of Section 5.9.4.5(5) of the Byron FES-OL.

E-13. The Intervenor presented testimony by Dr. Bernard John Wood, professor in the Department of Geological Sciences at Northwestern University. Dr. Wood testified about indications that the characterization of the groundwater system at Byron might be inadequate. Ff. Tr.

6879. As part of his participation in nuclear waste storage projects, Dr. Wood has studied the behavior of radionuclides in groundwater.

### *1. The FES Calculation of Travel Time*

E-14. To assess the consequences of contamination of the Byron groundwater system by a core meltdown, the Staff had to calculate how long radionuclides released by the meltdown would take to travel to the closest points where humans use the water. The shorter the travel time, the greater the consequences, since contaminants have less time to decay to safe levels, and engineers have less time to stop the spread of the contaminants. This travel time depends on the velocity of the radionuclides in the groundwater, and on the distance between where they are released to the groundwater and the closest point where they threaten human uses of the water. We show first how the Staff determined the distance in the Byron case.

E-15. A melted core which penetrated the basemat under the reactor in Unit 1 would release radionuclides into the dolomite and limestone of the Ordovician-age Galena and Platteville groups under the plant, but the radionuclides would not sink below that dolomite and limestone, for in the area of the plant, though not regionally, they are separated from lower layers of rock by a layer of shale, called the Harmony Hill Shale Member of the Glenwood Formation, which acts as a barrier to vertical water movement. Holish, ff. Tr. 6750, at 8. Therefore, to find the distance radionuclides would have to go to threaten humans, the Staff had only to find the groundwater sink which was nearest the reactors and supplied with water by the dolomite and limestone above the Harmony Hill Shale.

E-16. Most of the water for the region around the plant comes from the rock layers under the Harmony Hill Shale. *Id.* at 9-10. The dolomite and limestone above it supply water only to some wells near the town of Byron, to springs near the plant, and to the Rock River. Staff Ex. 2 (Byron FES-OL), at 5-57. Of these, the nearest to the Byron reactors is an unnamed spring whose water flows into the Rock River. *Id.* The spring is 3600 feet from the reactors. *Id.*

E-17. To make the final calculation of velocity the Staff used a formula called Darcy's equation. It expresses the velocity ($v$) of groundwater in a porous medium as directly proportional to both the permeability ($k$) of the medium and the slope ($i$) of the uppermost level of the groundwater, and inversely proportional to the "effective" porosity ($n_e$) of the medium. The porosity of a medium is the ratio of the volume of void in a given amount of that medium to the total volume of that amount. The "effective" porosity of a given amount of a medium is always less than the porosity of that amount, for pores which have no outlets have no effect on the movement of groundwater. Tr. 6689 (Codell). Darcy's equation puts these relations of proportionality succinctly: $v = ki/n_e$. Codell and Staley, ff. Tr. 6549, at 11.

E-18. All the relations the equation states, except one, seem plausible. It is not hard to see that the more permeable a medium is, the faster the groundwater can seep through it, and that the steeper the top of the groundwater — the groundwater "table" — in a medium is, the faster the groundwater flows down through the system. Moreover, it seems plausible that a given ratio of increase or decrease in permeability would be matched by the same ratio of increase or decrease of velocity, that this proportionality would also hold between slope and velocity, and that the effects of permeability and slope would multiply each other. But one might expect effective porosity ($n_e$) to be directly proportional to velocity; that is, the wider the path, the faster the flow. But groundwater systems are in equilibrium; that is, the quantity of flow in them is roughly constant. It is this equilibrium which makes it possible for a groundwater table to have a topography which does not match the topography of the ground surface. Now, quantity of flow, whether constant or not, is equal to the area of flow times the velocity of flow. Thus, when the quantity of flow is constant, but the area is increased, say by an increase in effective porosity, the velocity of flow must decrease. Thus, $n_e$ is in the denominator of Darcy's equation. *See* Tr. 6844 (Holish, Cole). So much for the equation taken as a whole. We shall now show how the Staff determined the numerical values of equation's parts.

E-19. To determine the numerical value of *k*, which represents the permeability of the bedrock under the plant, the Staff used pumping test data submitted by the Applicant in its FSAR. Those data yielded an average permeability of 1.82 feet per day. Codell and Staley, ff. Tr. 6549, at 10-11.

E-20. The value of 1.82 is conservative, for new data indicate a much lower permeability. Mr. Holish, one of the Applicant's witnesses, testified that he had recently decided that the pumping test data the Staff used to determine a value for *k* were not suitable for that purpose. Ff. Tr. 6750, at 15. The data come from tests performed in 1974 on two wells drilled on the western edge of the Byron site to find out whether the water in the wells had been contaminated by the cyanide which had been dumped at a salvage yard nearby. *Id.*; Tr. 6753 (Holish). Mr. Holish decided not to stretch the data beyond their original purpose, for although pumping tests can yield numerical values for *k*, the tests performed on those two wells could not. The pumping did not last long enough to achieve equilibrium, and certain supplemental measurements were not made. Holish, ff. Tr. 6750, at 16.

E-21. To acquire better data for determining *k*, the Applicant performed water pressure tests on thirty-one bore holes drilled to various depths in the bedrock under the area of the plant. The data from these tests yielded an average permeability of 0.52 foot per day. *Id.* at 17. The Staff calculated 0.42 foot per day using data from twelve of the thirty-one boreholes, twelve which gave good coverage of the main plant area. Codell and Staley, ff. Tr. 6549, at 12. These new values for *k* are less than the old one, 1.82, by factors of more than 3. Used in Darcy's equation, they would decrease velocity, and in turn, increase travel time, by the same factors. Such longer travel times would greatly reduce the risks posed by contaminated water. But, the Staff has prudently decided to continue to base its calculations of velocity and travel time on the older, more inaccurate, but also more conservative, figure, 1.82. Codell and Staley, ff. Tr. 6549, at 12.

E-22. The Staff was also conservative in determining a value for *i*, the slope of the groundwater table. "Slope" here means simply a quotient obtained by dividing the vertical distance from one point in the table to another by the horizontal distance between the same two points. The two points in the Byron case are, of course, the unnamed spring and the reactors, the latter taken as one point. The horizontal distance between these points is 3600 feet, as we've said. The vertical distance varies and permits a conservative choice of numbers. The elevation of the unnamed spring is 780 feet above sea level, but the height of the water table at the main plant area varies with rainfall and other factors. Eight hundred and forty feet was the elevation of the water table in a year of abnormally high recharge, before the ground surface was altered by construction. Now, construction, paving, improved surface drainage, and grouting of the bedrock under the plant have made the slope of the water table in the Byron site area almost zero. *Id.* at 10. An almost flat water table implies an almost zero velocity, for velocity and the slope of the table are directly proportional. Nonetheless, the Staff has chosen to rely on the more conservative, pre-construction figure of 840. Therefore, the value of "*i*" in Darcy's equation is (840 − 780)/3600, or 0.0167. *Id.* at 11 (footnote).

E-23. The Applicant determined the effective porosity, $n_e$, by geophysical logging techniques during site exploration. The Applicant also compared the values it got by these techniques with values published by the Illinois State Geologic Survey. Determined this way, $n_e$ varied between 0.02 and 0.10. Holish, ff. Tr. 6750, at 18. The Staff chose 0.075, the measured mean value, for its calculation of velocity. Codell and Staley, ff. Tr. 6549, at 11 (footnote).

E-24. Thus, setting $k$ equal to 1.82 feet per day, $i$ equal to 0.0167, and $n_e$ equal to 0.075, the Staff calculated the average velocity of the groundwater through the Galena-Platteville dolomite and limestone to be 0.4 foot per day. At this rate, groundwater would take 24.4 years to traverse the 3600 feet between the reactors and the unnamed spring which is the groundwater sink nearest the plant. Staff Ex. 2 (Byron FES-OL), at 5-57.

E-25. This travel time of 24.4 years is one of many factors which led the Staff to conclude in the cost-benefit summary in the Byron FES-OL that the impact on human health of accident risks at Byron was "small," meaning, "in the reviewers' judgment," "of such [a] minor nature, based on currently available information, that [it does] not warrant detailed investigations or consideration of mitigative actions." Staff Ex. 2, NUREG-0848 (Byron FES-OL) at 6-3. Since contamination of groundwater by a core meltdown is only one of several accidents the Staff analyzed before it came to this conclusion, the impact of the risk of this accident is very small indeed. *See generally id.*, Section 5.9.4, at 5-32 to 5-67.

E-26. A number of things make this conclusion seem secure. It rests not only on the conservative assumptions which went into the calculation of travel time but on other conservative assumptions as well. For example, the FES assumes that all the radionuclides released into the groundwater by a core melt would travel to the unnamed spring, when in fact, since the Byron site is on high ground, contaminated water would flow in all directions from the damaged reactor. Codell and Staley, ff. Tr. 6549, at 8-9. The FES also assumes that the radionuclides released into the groundwater would travel with the velocity of the water, when in fact, Sr-90 and Cs-137, the radionuclides which after 24.4 years would be the most important contributors to dose, would be slowed down by absorption into the media through which they would travel. Staff Ex. 2 (Byron FES-OL), at 5-57 to 5-58; *see also* Codell and Staley, ff. Tr. 6549, at 10; and the FES also assumes that every core meltdown would contaminate the groundwater, when, in fact, there is good reason to think that very few would; *see* Klopp, ff. Tr. 6750, at 6-8, and Tr. 6801. There are still other such assumptions.

E-27. Moreover, the FES's conclusion that the impact of contamination of the groundwater by a meltdown is small is relatively immune to large changes in the travel time estimate. For example, the Staff's witnesses testified that halving the travel time estimate would increase the percentages of Sr-90 and Cs-137 reaching surface waters by a factor of about 0.5, but those increased doses would still be less than an order of magnitude greater than the doses the Liquid Pathway Generic Study, NUREG-0440, the Staff's generic study of radionuclide contamination of water, predicted as consequences of a meltdown at a typical small river site. Codell and Staley, ff. Tr. 6549, at 13.

E-28. Last, the small impact the FES concludes groundwater contamination by a meltdown would have can be greatly reduced. Even though a judgment that the impact of the accident on human health would be "small" means that "consideration of mitigative actions" is "not warranted" (Staff Ex. 2 (Byron FES-OL), at 6-3), mitigative actions have already been considered, and in the FES the Staff notes that 24.4 years allow "ample time" to use grouting and well-point dewatering — standard engineering measures — to minimize the impact of the accident. *Id.* at 5-59.

E-29. Nonetheless, despite the conservative assumptions in the FES and the insensitivity of the FES's conclusion on impact to large changes in the travel time estimate, without time for mitigation or radioactive decay the impact of groundwater contamination on human health might not be small. The soundness of the FES analysis of that impact depends in part on whether the method the FES uses to estimate travel time is reasonably sound.

E-30. A sound way to estimate travel time is also crucial to the Applicant's conclusion that if the groundwater at Byron were to be contaminated by a rupture of one of the plant's boron recycle holdup tanks, there is reasonable assurance that the health and safety of the public would not be endangered. Applicant's Proposed Finding 344. The Board's Finding E-9 describes the accident and the regulations which deal with it. The Applicant calculates that radionuclides which leaked out of one of the boron tanks would take about 30 years to travel from the point of release to the nearest well, which is on site. Lahti, ff. Tr. 6750, at 5. According to the Applicant, in 30 years all the radionuclides released by the rupture, except Cs-134, Cs-137, H-3, and Sr-90, would decay to negligible levels; and these four exceptions, because the tank fluid would be diluted as it leaked into the groundwater, would, if not interdicted, appear in water in unrestricted areas in concentrations well within the applicable limits, which are set out in 10 C.F.R. Part 20, Appendix B, Table II, Column 2. Lahti, ff. Tr. 6750, at 6, Table I. The Staff thinks it is conservative to predict that a tank rupture could contaminate the groundwater, for if the water table were ever to be higher than it is now, fluid from the ruptured tank would meet groundwater at a point below the water table, and groundwater would flow into the building through the postulated basemat crack, rather than tank fluid flowing out. Tr. 6834-35 (Lahti); Staff's Proposed Finding G-65.

E-31. With one exception, the League does not criticize the numbers the Staff used with Darcy's equation; or the Staff's conclusion in the FES that, given a travel time of 24.4 years, or a similarly large number of years, the impact on human health of groundwater contamination by a core meltdown is small; or the Applicant's conclusion that, given a travel time of about 30 years, there is reasonable assurance that the health and safety of the public would not be endangered by groundwater contamination by a rupture in a boron tank. Indeed, the League's Proposed Findings do not even mention tank ruptures. The League saves almost all its criticisms for the linchpin of the Applicant's and the Staff's conclusions: That enough is known about the geology of the groundwater system at Byron to support sound estimates of radionuclide travel time. After we discuss the one criticism the League does not save for the linchpin, we shall discuss the rest of the League's criticisms.

E-32. The League argues in its Proposed Finding 17 that the results the Applicant obtained by using a second method of determining $n_e$, effective porosity, should have led to a travel time estimate 10 times shorter than the one the Applicant reported for contaminants released by a rupture of boron recycle holdup tank. This second method was developed by Dr. D.T. Snow and is reported by the American Society of Civil Engineers (Snow, *Journal of Soil Mechanics* (Foundation Division), Vol. 94 (1968)). The effective porosity as determined by Dr. Snow's method was less than the effective porosity reported in our Finding E-23 by a factor of somewhere between 2 and 10. *See* Holish, ff. Tr. 6750, at 25-26. It appears to the League that since effective porosity and velocity are, by Darcy's equation, inversely proportional, a tenfold decrease in effective porosity would mean a tenfold increase in velocity and a tenfold decrease in travel time. Tr. 6773-74 (Holish, Thomas). But the Intervenor has not understood the nature and purpose of Dr. Snow's method.

E-33. Applicant states that Intervenor is incorrect in its implication that the estimate of contaminant velocity is ten times too slow, pointing out that, in response to the question "So that means it [travel time to the nearest well of groundwater that might be contaminated from radionuclides as a result of a postulated accident involving a rupture of a boron recycle holding tank] could be three years rather than 30; right?" Applicant's witness answered "No, sir, it does not." Tr. 6775. The Snow analysis was apparently used for a different purpose than the evaluation of design basis accident scenarios, such as tank rupture scenarios. The Snow analysis was one of two qualitative evaluations of the hydrogeological aspects of a postulated core melt event, which is not a design basis accident scenario. The aperture size used in the Snow analysis was based on theoretical considerations, and the analysis was used only as the basis of comparison to determine the effects of aperture size and fracture size. Applicant's witness L.L. Holish stated that Dr. Snow's techniques were designed for granite bedrock in an unweathered zone with a relatively clean joint, a situation that does not exist at the Byron site. The witness further stated that he relied more heavily on Darcy's equation than he did on procedures prepared by Dr. Snow. *See* Holish, ff. Tr. 6750, at 21-27, esp. 24-26; and Tr. 6773-75.

E-34. The Board agrees with Applicant that application of the Snow analysis to contaminant travel time estimates is not appropriate since it was not proposed for that purpose and other more appropriate procedures and measurements were used to estimate contaminant travel times.

### 2. *Cyanide Migration*

E-35. During the hearings on groundwater contamination, the main case the League made against the Applicant's and the Staff's estimates of travel time was based on data the League's witness, Dr. Wood, interpreted to mean that groundwater contaminated with cyanide had moved through bedrock near the Byron site with a velocity about 20 times the velocity the Staff had calculated using Darcy's equation. *See* Wood, ff. Tr. 6879, at 5.

E-36. Between 1969 and 1972, cyanide was dumped in various forms and at various locations in the Byron salvage yard, which is a few miles from the Byron site.[76] Tr. 6605 (Codell). During 1974 and 1975, a study of the water contamination the dumping caused was conducted. The results were published in an article[77] written by four people from the Illinois State Geological Survey (ISGS), R.H. Gilkerson, K. Cartwright, L.R. Folmer, and T.M. Johnson. While the study was being conducted, cyanide was found in wells 1.2 and 1.8 miles northwest of the salvage yard.

E-37. On the assumption, conservative for purposes of calculation, that the cyanide found in the well 1.8 miles from the yard was dumped in 1969 and did not reach the well until just before the investigators found it in the well water, the League's witness, Dr. Wood, calculated that the cyanide would have traveled "at least" 8 feet a day. Wood, ff.

[76] The Applicant had nothing to do with the dumping. Tr. 6684 (Codell).

[77] "Contribution of Surficial Deposits, Bedrock and Industrial Wastes to Certain Trace Elements in Groundwater," 15th Annual Symposium on Engineering Geology and Soil Engineering Proceedings, 1978.

Tr. 6879, at 5.[78] On the basis of this calculation, Dr. Wood argued that the fractures which all parties agree exist at the plant site must be longer than the Applicant and Staff had thought, and that therefore some method other than Darcy's equation, which is designed for use with uniform porous media, should be used to determine the velocity of contaminated groundwater. *See* Wood, ff. Tr. 6879, at 5-6.

E-38. The League's Proposed Findings do not mention the cyanide migration. Arguably, then, the applicability of the cyanide migration data to the Byron plant site is no longer a matter of disagreement among the parties. Nonetheless, the Board will briefly consider the applicability, for the data on cyanide migration were a major part of the League's case (*see* Wood, ff. Tr. 6879), perhaps even an inspiration for the League's contention on groundwater.

E-39. There are two principal obstacles to treating Dr. Wood's calculation of the velocity of the cyanide as an indication of the velocity of radionuclides released into the groundwater by an accident at the Byron plant. The first obstacle is that, despite the thorough study by ISGS (Tr. 6703 (Codell)), too little is known about what happened at the salvage yard, and about how the cyanide got to the wells, for the velocity of the cyanide to be calculated at all. Tr. 6602-03 (Codell). The second obstacle is that, even if enough were known to calculate that velocity, the result could not be applied to Byron, for the geology of the salvage yard and the geology of the plant site are not the same. *Id.*

E-40. First, it is not certain just how the cyanide was disposed of at the yard, but there are strong indications that a large amount of it was disposed of in ways which permitted it to flow out of the yard by surface routes. Much of it was buried in barrels, and other containers, under only a little ground cover, and some of the containers were punctured. Codell and Staley, ff. Tr. 6549, at 17; Tr. 6674 (Codell). Some of the cyanide was stored on the surface near stream channels and other impressions in the ground. *Id.*; Codell and Staley, ff. Tr. 6549, at 18. Some was dumped in liquid form into lagoons, some of which were broken later, thus releasing cyanide to surface water. *Id.* The cyanide in these lagoons was in concentrations so high that some cattle which drank from the lagoons died. Tr. 6770 (Holish). Some of the cyanide was sprayed on roads and thus may have been carried away from the yard by the wind. Codell and Staley, ff. Tr. 6549, at 17. Indeed there is some evidence that

[78] Dr. Wood said that to get 8 feet a day, he divided 1.8 miles by the number of days in five years. Tr. 6889. Performing the same calculation (9504 feet divided by 1826 days), the Board gets 5.2 feet a day, a large figure to be sure, but one 13 times, not 20, the 0.4 foot per day the Staff got using Darcy's equation, and thus more in keeping with Dr. Wood's testimony that the Staff's figure might be too slow by a factor of 10. Tr. 6922-23.

very little cyanide left the yard by way of the groundwater under the yard; there was little or no cyanide in the wells in and around the yard. Tr. 6771 (Holish).

E-41. Given these strong indications that the cyanide traveled out of the yard on the surface, the velocity of the cyanide cannot be calculated; for the cyanide which reached the well 1.8 miles away may have traveled a considerable distance in surface water and then gone down into the groundwater. Unless the distance it traveled on the surface is known, the distance it traveled underground cannot be known, and therefore its velocity underground cannot be calculated. Tr. 6867 (Holish).

E-42. Second, even if that velocity could have been calculated, the result could not be applied to Byron. Any groundwater movement there may have been between the salvage yard and the wells 1.2 and 1.8 miles away would be faster than the movement between the plant site and the unnamed spring 3600 feet away. It was the League's witness who testified that whatever cyanide there was in the groundwater moved at a depth of only tens of feet at most, in the shallow groundwater system. Tr. 6886-87. But radionuclides from accidents at Byron would enter the groundwater system at a lower point in the bedrock, where the rock is less fractured. Holish, ff. Tr. 6750, at 23. Moreover, both the Applicant's witness Mr. Holish, and the League's witness, Dr. Wood, agree that the rock under the salvage yard is more fractured than the rock under the plant site. Tr. 6749 (Holish) and 6911 (Wood). Last, the ground surface at the salvage yard is much steeper than at the site. Tr. 6602 (Codell). Since flow in the shallow groundwater system is controlled by the topography of the land (Holish, ff. Tr. 6750, at 23), the steeper the land, the faster the flow. Therefore shallow groundwater flow at the yard will be faster than shallow groundwater flow at the site.

E-43. These, then, are the arguments which appear to have persuaded the League not to mention the ISGS study of cyanide migration in its Proposed Findings and which persuade the Board to find that study not applicable to the Byron site. However, the League does make a half-hearted attempt to apply another study to the Byron plant site. The League's Proposed Finding 17 begins, "NRC Staff conceded that it is possible that contamination could move as Dr. Wood calculated," a sentence which imports much less than it appears to. During oral testimony, in answer to a question from the Board about what velocities contaminants had been known to have in fractured limestone, the Staff's witness Dr. Codell said that velocities of a few feet a day had been recorded in Florida fractured limestone, but that he did not know how that limestone compared to the limestone at the Byron site. Tr. 6700. The League's Proposed Finding 17 goes on to say that this Florida figure "corroborates Dr. Wood's speculation that contaminants in limestone can move a few feet a day."[79]

E-44. However, Dr. Wood's "speculation" was not simply that contaminants can move a few feet a day in limestone. No one in this proceeding has denied that, and thus to say that the Staff concedes it is to say something of no probative force. Indeed what the League calls speculation is no speculation at all. Darcy's equation itself would confirm it for contaminants in groundwater under a water table of a suitable gradient, in limestone of suitable permeability and effective porosity. Dr. Wood's speculation, rather, was that contaminants in the limestone at the Byron salvage yard in particular, but also at the Byron plant site, can move a few feet a day. This speculation is hardly corroborated by the Florida data, for it is not known how the site of the Florida measurements compares to either the Byron salvage yard or the plant site. There may be as little similarity between the Florida site and either of the Byron locations as there is between the two Byron locations.

### 3. *The League's Argument Now*

E-45. Having dropped, apparently, any argument based on the migration of cyanide from the Byron salvage yard, the League now argues simply that the Staff and the Applicant, in determining travel times of radionuclides released into groundwater by accidents, have given inadequate consideration to the fracturing which exists in the bedrock at the plant site, and are therefore not justified in using Darcy's equation to calculate travel times for the radionuclides. The League's words are these: "The Applicant and the NRC differ from the Intervenor on two points. The first is how to characterize the fracturing that all parties agree exists. The second point of contention is on the appropriate investigatory measures which provide the best bases for that characterization [citations omitted]." Proposed Finding 12. The League's witness, Dr. Wood, testified that the bedrock under the plant is "extremely fractured and jointed." Wood, ff. Tr. 6879, at 4. He said that one sign of the extensive fractures is the large number of springs which flow radially from the site. *Id.* at 6. The Byron bedrock is not a uniform porous medium, and the Staff and the Applicant have never claimed it was. The question is whether, for all practical purposes, it transports water much as a uniform porous medium would. Even the League's witness, Dr. Wood, agrees that the fracturing in a bedrock formation does not, of itself, mean that

[79] This sentence is the only sign in the League's proposed findings of its attempt to apply the cyanide migration study to the plant site.

Darcy's equation cannot be applied to it. He argued that where the fractures are parallel, water flows mainly in the direction of the fractures, and Darcy's equation will underestimate the velocity of the flow. Tr. 6942. For example, in a long, perfectly straight, continuous, clean fracture, 1/40 of an inch wide, having as much surface area as fractures in the Byron area have, water can flow as quickly as 3000 feet per day. Wood, ff. Tr. 6879, at 5-6; Tr. 6698 (Staley). But Dr. Wood also said that if the length of the fractures is small in relation to the distance which is being used to calculate travel time, and if the fractures change direction often enough, and mesh into a network which disperses flow in many directions, then, as experience has shown, the velocity of the flow can be estimated with Darcy's equation, even though the equation was derived from experimental work with uniform porous media. Tr. 6941 (Wood).

E-46. The principal technical question, then, the answer to which determines the outcome on this contention, is whether the Applicant has presented enough evidence to show that the extensive fracturing in the bedrock at Byron is of the sort which permits the application of an equation developed for use with uniform porous media.

### *4. The Applicant's Investigation of the Groundwater System at Byron*

E-47. No party contends that the Applicant did not make a thorough investigation of the geology at Byron, one fully in accord with the Staff regulatory guidance in effect at the time of the investigation — Sections 2.4 and 2.5 of Regulatory Guide 1.70 — and even consistent with the methodology suggested by two later Guides, 1.138 and 1.132. *See* Holish, ff. Tr. 6750, at 4; Staff's Proposed Finding G-11. The League's witness, Dr. Wood, had no criticisms to make of the actual work done during the investigation. Tr. 6913-15. The only question about the investigation is whether its results will justify the use of Darcy's equation. The scope and principal results of the investigation the Applicant made of the geology of Byron are reported in Chapter 2.5 of the Byron FSAR. Here the Board will discuss only those results which bear on whether Darcy's equation can be applied at Byron. These results come mainly from the Applicant's mapping of the site (*see* Holish, ff. Tr. 6750, Exhibit II); drilling, sampling and selective water pressure testing of 154 borings which varied in depth from 10 to 330 feet (*id.* at 4); and direct observation of the bedrock at various elevations during excavations for the foundations of the plant. Tr. 6864 (Holish).

E-48. The investigation showed that there are four sets of parallel fractures in and around the site. One set is parallel, another perpendicular, to the regional structure of fractures. The other two sets of parallel fractures are neither parallel nor perpendicular to the regional structure, but they make right angles with each other just as the first two sets do. Holish, ff. Tr. 6750, at 11-12. On the surface of the ground, parallel fractures are from 200 to 500 feet apart; below the surface they are closer together. *Id.* at 12. Some of the fractures are clean; these range from 1/16 to 1/4 inch in width, others are filled with clay produced by weathering and rock solutioning. *Id.* Fracturing and weathering decrease below the uppermost formation of the Galena-Platteville dolomites and limestone, the Dunleith formation. *Id.*

E-49. Although the Byron bedrock is clearly very fractured, there are no indications that anything like Dr. Wood's hypothetical long, straight, clean, 1/40-inch-wide fracture in which water can flow 3000 feet a day exists at Byron. The mapping of the ground surface in the area does not show such fractures. *See* Holish, ff. Tr. 6750, Ex. II. The maps of the piezometric surface — that is, the water table — at and around the site, do not show the significant depressions which such fractures would cause (Codell and Staley, ff. Tr. 6549, at 11; Tr. 6655-56 (Staley); Staff Ex. 2, NUREG-0848 (Byron FES-OL), at 4-25, 4-29). And the slope (*i*) of the water table would probably be lower than the 0.0167 the Staff calculated if there were a fracture extending from the site to the Rock River. Tr. 6938 (Wood, Cole). Also nothing in the 154 borings the Applicant made shows continuous, large fractures which could provide a direct pathway to the River. Codell and Staley, ff. Tr. 6549, at 14.

E-50. Despite the extensive fracturing at Byron, there is nothing incredible about these results. To the contrary, Dr. Wood's hypothetical fracture was the incredible thing here. As a thing of the imagination, a long, straight, clean fracture 1/40 inch wide can be visualized, and theoretical velocities of water flowing in it can be calculated. Indeed, according to the Staff's witness, Mr. Staley, the velocity of water flowing in such a fracture can even exceed the 3000 feet a day the League's witness, Dr. Wood, calculated. Tr. 6698. But in nature, there are not long, perfectly smooth, perfectly straight, perfectly clean fractures. Dead ends, rough walls, sharp bends, and absorbants like the clay which fills many of the fractures at Byron, all greatly retard the flow of water. Tr. 6697-99, 6701, 6703-08 (Codell, Staley).

E-51. Of course, long fractures resembling Dr. Wood's hypothetical one are not the only geological features which permit water to flow at a high velocity. There can be underground rivers and caves in limestone, and water can flow between the layers of stratified rock. But the Byron geology rules out rivers and caves (Tr. 6700 (Codell)), and large-scale flow between rock layers is unlikely at the elevation at which radionuclides would be released into the groundwater by a core meltdown or tank rupture: The highly fractured rock is above elevations 810 to 815. Tr. 6711 (Staley). But the base of the reactors is at 800, and the bottom of the basemat under them is, of course, even lower. Tr. 6743 (Holish). Also, the boron recycle holdup tanks are 54 feet below the surface, which is at 869 (NUREG-0876 (Byron SER), at 2-1); and the concrete floor underneath a tank goes down another 8 feet. Lahti, ff. Tr. 6750, at 4. Radionuclides would be released at these lower elevations, where the bedrock is only slightly to moderately fractured. Tr. 6743, 6745-46 (Holish). There is enough weight above these elevations to make extensive spaces between rock layers very unlikely. Codell and Staley, ff. Tr. 6549, at 16.

E-52. Not only is there no sign of Dr. Wood's hypothetical fracture at Byron, the fractures that are there go in four directions and mesh into a net which disperses the flow of groundwater across the whole site, and thus prevents the water from flowing in only one direction. Tr. 6865-66 (Holish). Thus, there would appear to be at Byron an example of the kind of pattern of fractures which, Dr. Wood testified, permits the use of Darcy's equation. *See* our Finding E-21. Indeed, if the bedrock at Byron were not fractured, or if the fractures there were distributed randomly, there would be little groundwater flow at the site, for the bedrock at Byron, mostly dolomite, a dense limestone (Tr. 6688 (Codell)), is not considered to be a porous medium. Tr. 6866 (Holish). The fracturing at Byron, then, not only doesn't prevent the application of Darcy's equation, it makes the application possible in the first place, and it is the principal factor which determines the permeability ($k$) of the bedrock. (*See* Holish, ff. Tr. 6750, at 22).

### 5. *The League's Requests for Relief*

E-53. Even though the League's witness, Dr. Wood, had no complaint with the actual investigative work the Applicant did, he claimed that more work had to be done to achieve an adequate characterization of the groundwater system at Byron. In his written testimony, he called for a model of the fissure system at the plant site and a tracer study to confirm that the model was correct. He thought that, not only would the model and the tracer studies yield accurate predictions of travel times, they would also yield information on which sound plans for mitigating the consequences of releases to the groundwater could be based. Ff. Tr. 6879, at 8. Dr. Wood stood by these recommendations during his oral testimony despite the arguments the Staff and the Applicant raised against applying the results of the cyanide migration study to Byron, which application had been a central part of his testimony.

E-54. In its proposed findings, the League does not repeat Dr. Wood's call for a model of the fracturing, or for a tracer study. Apparently the League was persuaded by the Staff's and the Applicant's arguments that modeling and tracer work would be highly impractical to do now. The Staff's witness, Mr. Staley, testified that because it is difficult to know what factors should be built into the model for bed losses due to friction, it might take months to model a mere 100 feet of fractures. Tr. 6595-96. The Applicant's witness, Mr. Holish, testified that tracer studies are very difficult to perform and very time-consuming, and that the techniques used in them are known by only a few specialists. Tr. 6768.

E-55. Nonetheless the League does seem to think that these things should have been done. In its Proposed Finding 15, it says, "[t]he Applicant has not used the time of its involvement with the site to actually trace migration of contaminants from the site . . ." When this proposed finding is read together with the League's not calling for models and tracers in its proposed findings, it appears that the League is suggesting that the Applicant has been negligent in not conducting such studies at the site but that since they are so time-consuming, it is too late to start them now. Insisting, despite all the evidence to the contrary (*see* the Board's Findings E-49 and E-50), that "it is not possible to say that there is not a continuous joint running from the site to the Rock River" (Proposed Finding 9), and thinking, it would seem, that it is too late to start making models or releasing tracers, the League calls on the Board to deny the Applicant a license to operate the Byron plant.

E-56. The Board will not on these grounds deny the Applicant a license for Byron. The evidence shows, and therefore the Board finds, that the Applicant's account of the fracturing in the bedrock at Byron has been adequate to justify the use of Darcy's equation to calculate the velocity of radionuclides in the groundwater. With such evidence, and in the absence of any rule or regulation which requires models or tracers, it cannot be said that the Applicant has been negligent, or that it should begin work on models and tracers now. The evidence also shows that the Staff and the Applicant have been conservative in determining numerical values for the terms in Darcy's equation, and thus that the radionuclide travel time estimates reported in the Byron 1982 FES analysis of the impact of a core meltdown on groundwater, and the travel time estimate reported in the Byron FSAR analysis of the consequences of a ruptured boron recycle holdup tank, are adequate for the purposes of those documents. Therefore, the Board finds that the NEPA analysis in the FES of the impact of a core meltdown on groundwater is adequate. The Board also finds that the analysis in the Byron FSAR of the consequences of release of radionuclides into the groundwater by a rupture in a boron recycle holdup tank is adequate, and that there is reasonable assurance that this release would neither endanger the health and safety of the public nor be inimical to the common defense and security.

E-57. Even Dr. Wood, the League's expert witness, thinks that the fracturing in the bedrock at Byron does not give cause to postpone operation of the plant. When asked whether he thought it made sense to keep the plant idle while a tracer study was carried out, when such a study might take decades, he answered that the study should be done while the plant was in operation. Tr. 6921.

### 6. *The League's Proposed Finding on Interdictive Measures*

E-58. The League may have proposed a fallback remedy. In its proposed "Conclusions of Law," the League asks only that the Board deny the Applicant a license. But in its last proposed finding, the League says that because the Byron groundwater system has been inadequately characterized, "the Applicant's interdictive measures must be redesigned." League's Proposed Finding 22. This redesign was the third and last of Dr. Wood's recommendations, and the only one which the League has carried over from testimony to proposed findings. But at first glance, the League appears self-contradictory in retaining Dr. Wood's third recommendation, and at the same time asking that an operating license be denied. The only way the Board has been able to make sense of the League's call for redesign of interdiction is to regard it as the League's saying what it wants if the Board doesn't deny the Applicant an operating license. The Board, however, will not grant the League this fallback relief either.

E-59. The recommendation that the Applicant redesign its interdictive measures made more sense coming from Dr. Wood. Not only did he not urge simultaneous redesign and denial of a license. In his scheme of recommendations, redesign of interdiction was the natural consequence of modeling and tracer work; he thought that the information he expected those methods to yield would serve as the basis for redesigning interdiction. Wood, ff. Tr. 6879, at 8. But the League did not carry Dr. Wood's recommendations for modeling and tracer work over into its Proposed Findings, nor has it said very clearly what else might guide redesign.

E-60. More important, there is no evidence in the record that the interdictive measures the Staff and the Applicant now have available are inadequate, either in and of themselves, or in relation to the geology of the Byron groundwater system. Witnesses for the Staff and the Applicant testified that there are two standard interdictive measures, both of which could be used at Byron. One would be to make an impermeable barrier in the rock by pressure rock cement grouting the Galena-Platteville formation down gradient from the spill. Holish, ff. Tr. 6750, at 27; Staff Ex. 2, NUREG-0848 (Byron FES-OL), at 5-59. The other procedure would be to drill wells into the bedrock at the perimeter of the site. The wells would be close enough to each other to overlap in influence and cause a drawdown great enough to reverse the hydraulic gradient at the perimeter, and the contaminated water would be pumped out of the wells and stored for treatment.[80] *Id.*

E-61. Both of these procedures are routine engineering methods. Tr. 6640-41 (Staley). The Applicant's witness, Mr. Klopp, a General Design Engineer with the Applicant, knows both procedures. He has had experience in dealing with radioactive spills (Tr. 6727-29 (Klopp)), and he helped plan the bedrock grouting that was done at Byron to strengthen the foundations of the plant.[81] Tr. 6730 (Klopp).

E-62. Finally, there is no evidence that these measures do not suit the geology of the groundwater system at Byron. When the League says, in its Proposed Finding 22, that interdictive measures must be redesigned because the groundwater system at Byron has been inadequately characterized, the League may mean that there may be more fracturing at Byron than the Staff and the Applicant think there is, that there may even be a fracture extending from the plant to the Rock River. On this reading of the League's Proposed Finding 22, the League's call for redesign of interdictive measures makes some sense, for the League suggests what might guide redesign, namely, the mere possibility of more fracturing than the Applicant thought was there. But, as we noted above, there is no evidence that there is more fracturing in the bedrock at Byron than the Staff and the Applicant think there is, and there is good evidence that there is no fracture which extends from the plant to the Rock River. *See* Board Findings E-49 through E-51.

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[80] *Also, some wells would be drilled down gradient from the spill for use in monitoring the* groundwater. Holish, ff. Tr. 6750, at 27.

[81] Both the Staff and the Applicant were conservative in ignoring the effect of this grouting when they calculated travel times. *See* Codell and Staley, ff. Tr. 6549, at 11; Tr. 6733-34 (Holish). The Applicant's witness, Mr. Holish, thinks that in reducing by a factor of 10 the permeability ($k$) of a part of the path radionuclides released into the groundwater would take (*see* Codell and Staley, ff. Tr. 6549, at 14-15), the grouting increases the travel times of the radionuclides by several months. Tr. 6829-31 (Holish).

E-63. Indeed, there is no evidence in the record that even if the travel times estimated by the Staff and the Applicant were less by a factor of 10, as Dr. Wood thought they could be (*see* Tr. 6922-23), the interdictive measures described in the Board's Finding E-60 would not be adequate. The Staff's witness, Mr. Staley, had enough confidence in these measures, and in the grouting that was done to give the plant a firmer bedrock foundation, to assert that they by themselves gave adequate protection to health and safety. Tr. 6656 (Staley). The Board, however, is obligated by the contention to make findings on the larger question of whether the Byron groundwater system has been adequately characterized. The Board accordingly adopts as the ground of its decision the same larger consideration that the Staff adopts in its proposed findings: that the groundwater system at Byron has been adequately characterized.

### 7. *Conclusions on Groundwater Pathway*

E-64. Contrary to the League's contention on groundwater, the Board concludes that the Staff and the Applicant have adequately characterized the groundwater system at Byron. Therefore, the Board concludes that the Staff has made an adequate NEPA analysis of the impact of a core meltdown on the groundwater at Byron, that the Applicant's FSAR analysis of the consequences of a rupture in a boron recycle holdup tank meets the requirements of 10 C.F.R. 50.34(b)(4), that there is reasonable assurance that such a rupture would not endanger the health and safety of the public (*see* 10 C.F.R. 50.57(a)(3)(i)), and that such a rupture will not make issuance of a license inimical to the common defense and security (*see* 10 C.F.R. 50.57(a)(6)).

## F. Seismic Analysis of the Byron Site

F-1. Intervenor League of Women Voters alleged that the seismic design of the Byron nuclear plant is inadequate to assure its safe operation. Contention 106, as admitted for litigation, states:

> There exist serious seismic-related site problems discovered subsequent to the construction permit herein which indicate that the seismic design for Byron is not such that there exists assurance that these problems are adequately resolved in accordance with applicable regulations, including but not limited to 10 C.F.R. 50.57(a)(3)(i), 50.57(a)(6) and 10 C.F.R. Part 100, Appendix A. Specifically, the Rockford League of Women Voters contends that due to the lack of reliable information regarding the causes of earthquakes which have been experienced in northern Illinois, Edison should be required to perform strain gage tests on faults cutting basement rock located in the northern Illinois region where earthquakes of modified

Mercalli VII or greater intensity are expected to occur. Further, recent evidence from the central portion of the United States shows that neither the Byron-designated safe shutdown earthquake peak ground acceleration value of 0.20(*g*) nor the operating basis earthquake peak ground acceleration value of 0.09(*g*) are sufficiently conservative. Ground acceleration significantly greater than both of these values is possible at the Byron site. In addition, it is not known if the recently discovered Plum River Fault is a capable fault. This fault is known to approach the Byron site within 5.3 miles and may even be closer if the fault extends further to the east.

### *1. Applicable Law*

F-2. Commission regulations require that nuclear power plants shall be designed, constructed and operated with reasonable assurance that the health and safety of the public will be preserved. 10 C.F.R. 50.57(a)(3). Specific to this contention is the General Design Criterion establishing minimal requirements for protection against natural phenomena including seismic events, 10 C.F.R. Part 50, Appendix A, Criterion 2, which says, in pertinent part, that "[s]tructures, systems, and components important to safety shall be designed to withstand the effects of . . . earthquakes . . . without loss of capability to perform their safety functions." The regulations require detailed investigations of the geology of the plant site and an analysis of the historic record of seismic activity of the area.

F-3. Phenomena proposed as the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) have been established in the Regulations as guides in the determination of the structural requirements of a nuclear plant necessary to counter ground motions and to give reasonable assurance of the preservation of the health and safety of the public.[82] 10 C.F.R. Part 100, Appendix A, Section III(c) and (d).

F-4. The specifications of the Safe Shutdown Earthquake are based on an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. Certain structures, systems and components shall be designed to remain functional when subjected to the maximum vibratory ground motion produced by an SSE. These structures, systems and components are those necessary to assure:

[82] These phenomena were also described by the Staff in the testimony. The SSE is the ground motion defined by a spectrum (amplitude of ground motion as a function of frequency) at which the plant has to be capable of closing down, and does close down without release of contaminants; the OBE, also defined by a spectrum, is the ground motion at which the plant is required to be shut down in an orderly manner and an investigation initiated to determine the occurrence of damage. Tr. 759 (Rothman).

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures set out in 10 C.F.R. Part 100.

10 C.F.R. Part 100, Appendix A, Section III(c).

F-5. An Operating Basis Earthquake is one which could reasonably be expected to affect the site during the operating life of the plant. It is the earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. 10 C.F.R. Part 100, Appendix A, Section III(d).

F-6. The Regulations define a capable fault as a fault which has exhibited one or more of the following characteristics:

(1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years.

(2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.

(3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

10 C.F.R. Part 100, Appendix A, Section III(g).

F-7. Additionally, the Regulations provide that the maximum vibratory ground acceleration of the Operating Basis Earthquake shall be at least one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake. 10 C.F.R. Part 100, Appendix A, Section V(a)(2). A departure by an applicant from one or more of the criteria established by General Design Criterion 2 is permitted by the Regulations for good cause shown. 10 C.F.R. Part 100, Appendix A, Section II. Specifically, in this instance, the ratio of the ground accelerations of the Operating Basis Earthquake and the Safe Shutdown Earthquake may be established at a value different from one half.

F-8. Five witnesses provided evidence on this contention. Applicant's case consisted of the testimony of Alan K. Yonk and Anand K. Singh, a geologist and a structural engineer, respectively, employed by Sargent & Lundy, the architect-engineer for the Byron Station. Witnesses for the Nuclear Regulatory Commission Staff were Ina B. Alterman and Robert L. Rothman, a geologist and a seismologist, respectively, both of the Office of Nuclear Reactor Regulation. The League presented the testimony of Henry H. Woodard of the Department of Geology at Beloit College in Wisconsin.

F-9. The Byron area is located on the Central Stable Region tectonic province, a region of relative consistency of surface geologic structural features. It is rather extensive, reaching essentially from the Rocky Mountains well into New York State and south to Oklahoma, a region characterized, in general, by a relatively low level of seismicity. A few areas within the province have, however, experienced earthquakes. On the order of ten earthquakes per 4,000 square miles were reported between 1800 and 1977. Since earthquakes in the Central and Eastern United States typically occur 5 to 20 kilometers below the ground surface, their cause must be sought in the structural features at those depths rather than in the surface characteristics. Applicant Ex. 1, at 2-24 and 2-25; Tr. 847-48 (Rothman).

F-10. The Byron site is on a till plain comprised of a 4- to 27-foot-thick layer of loess and glacial drift in place for 15,000 to a million years. This till rests on an Ordovician dolomite established 500 Mybp[83] which, in turn, is supported by the primarily granitic Precambrian 800-My-old bedrock. Applicant Ex. 1, at 2-22.

## 2. *The Sandwich Fault*

F-11. The Sandwich Fault Zone located approximately 6 miles southwest of the Byron site was also known and considered at the construction permit stage. *See* Finding 96 of LBP-74-87, 8 AEC 1006 (1974), at 1036. It was deemed not capable at that time.

F-12. Since the issuance of the construction permit, the Illinois State Geological Survey (ISGS) has performed a detailed investigation of the Sandwich Fault Zone to determine its extent, amount of offset, age, and the nature of faulting. The investigation included detailed field mapping, well records, drill cores, sample studies, seismic refraction work, down-hole geophysical logging and earth resistivity profiles determined by the latest techniques. The results are presented in "The Sandwich Fault Zone of Northern Illinois" ISGS Circular 505 (1978). Alterman, ff. Tr. 753, at 2, 5.

F-13. The detailed ISGS investigation confirmed that neither glacial material nor subjacent residual soil was offset anywhere along the entire

[83] Mybp = Million years before the present.

length of the fault wherever the younger material was observed. Subsequent reexamination of glacial tills strongly supported an Illinoian age (500,000-125,000 ybp) for the tills in the Byron area. This would require the undisturbed residual soil beneath the glacial till to be of the Yarmouth interglacial period (600,000 ybp). *Id.* at 2.

F-14. The witnesses are in agreement about the noncapability of the Sandwich Fault Zone. Yonk, ff. Tr. 432, at 6-8; Alterman, ff. Tr. 753, at 4; Tr. 567 (Woodard).

### 3. *The Plum River Fault*

F-15. The Plum River Fault, trending east-west to within 5.3 miles northwest of the site, was originally thought to be an anticlinal structure. A detailed study done by the ISGS following the issuance of the Byron construction permit provided evidence that it was a fault zone. The study which identified this fault zone is documented in "Plum River Fault Zone of Northwestern Illinois" ISGS Circular 491 (1976). Alterman, ff. Tr. 753, at 3; Yonk, ff. Tr. 432, at 6; SER (Staff Ex. 1), at 2-23. The proximity of the fault to the Byron site was determined by ISGS from observations of the decrease, in the easterly direction, of the offset in the bedrock. Tr. 812 (Alterman).

F-16. Field observations, well records, cores and seismic refraction data indicate the fault zone is generally less than half a mile wide with strata displaced 100 to 400 feet vertically on the north. ISGS Circular 491, at 1, 2.

F-17. The fault was identified through core drillings at two locations through the glacial deposits to the top of bedrock and by seismic refraction techniques. Alterman, ff. Tr. 753, at 3; Tr. 791, 832. That these indirect observations were required to locate the fault indicates that the surface materials remain undisturbed. Alterman, Tr. 762-64, 831, 834.

F-18. The cores from the two borings, taken about ½ mile apart in a north-south direction, in the Plum River area, identified differently aged bedrock structures at the same elevation above sea level. Each of these structures is covered by similar glacial deposits. The structure to the north is Silurian, to the south, Ordovician, some 50 My older. The fault was thereby determined to be between the bore holes.[84] These conclusions were confirmed by seismic refraction tests. Tr. 763, 832-834 (Alterman). The till-bedrock interface can be located to an accuracy of ±1 foot by seismic tests. Tr. 842 (Rothman).

[84] An excerpt from ISGS Circular 491, sketches from p. 16, purporting to show these details is bound following Tr. 822.

F-19. The elevation of the top of the bedrock is relatively constant across the fault zone indicating that, after the south side was uplifted 100 to 400 feet relative to the north side, erosion, largely glacial, leveled the bedrock surface leaving little evidence of a scarp. ISGS Circular 491, at 17. A Staff witness had personally observed the absence of an escarpment. Tr. 822 (Alterman). This absence indicates that there has been no vertical movement at least since the time the glacier passed over the area. Tr. 820-21 (Rothman).

F-20. In more recent conversations and correspondence between the Illinois State Geological Survey and the Staff, the Survey reaffirmed that it has never found disturbed till overlying any of the faults in northern Illinois. Tr. 764, 787, 860-61 (Alterman).

F-21. The glacial till overlying the fault has also been determined to be of Illinoian age and, being undisturbed, establishes, through its age (0.40 to 0.13 Mybp), a limit on the recency of the Plum River Fault. Staff Ex. 1, Section 2.5.1.1.

F-22. Applicant and Staff witnesses agreed with the conclusions drawn by the ISGS that the Plum River fault zone predates the deposition of Illinoian-age soils. Yonk, ff. Tr. 432, at 6; Alterman, ff. Tr. 753, at 3.

F-23. Knowledge of the regional tectonics supports the conclusion that the Sandwich and Plum River Faults are Paleozoic (600 to 250 Mybp) with later movement probably not after the Cretaceous period (65 Mybp). *Id.* at 2; Staff Ex. 1, at 2-23, 2-24.

F-24. Additionally, a Staff witness concluded that the Plum River Fault was not capable, basing her opinion upon evidence documented in ISGS Circular 491: (1) the Illinoian glacial till overlying the fault zone is undisturbed; (2) there is no seismicity associated with the fault zone; (3) there is no fault escarpment, and (4) the regional tectonic history indicates that faulting in Illinois is at least 65 My old. Tr. 788, 818 (Alterman).

F-25. This absence indicates that there has been no vertical movement at least since the time the glacier passed over the area. Tr. 820, 821 (Rothman).

F-26. The Intervenor's witness Woodard opined that the noncapability of the Plum River Fault has not been demonstrated and he faults the ISGS for basing its conclusion on techniques insufficiently accurate to determine the displacement of the till at the fault line. Further, he contended that a direct observation of the region should have been made in an excavation. Tr. 565-68, 571, 574 (Woodard).

F-27. The Intervenor's witness was not aware of any evidence of a fault which had displaced overlying northern Illinoian-age soil deposits or of any characteristics showing the Plum River Fault to be capable. Tr. 560-61, 564, 582 (Woodard).

### 4. *Minor Displacement Faults on the Byron Site*

F-28. The Byron site is basically a rock site. The plant foundations extend into the upper bedrock which are part of the Ordovician-age Galena Group dolomites. These dolomites are jointed and fractured in the upper formations of the Galena Group. Some solution activity has taken place among the joints causing widening. Minor offsets in some of the joints technically qualify them as faults. Yonk, ff. Tr. 432, at 3.

F-29. These offsets, observed during excavation of the Byron site were found to have vertical displacements typically 1 to 6 inches, with lateral extents of as much as 1800 feet. Whether these subsurface features should be designated as capable faults was the subject of considerable investigation and study, and the issue was ventilated at length in evidentiary sessions as a part of the litigation of the construction permit for Byron. The then-presiding Licensing Board concluded that these faults underlying the site are not capable. *See* Findings 13-21 of LBP-75-64, 2 NRC 712 (1975) at 715-18 and Findings 15-19 of LBP-75-74, 2 NRC 972 (1975) at 977-79. None of the parties, including the League, disputes the fact that the minor displacements are not capable faults according to the criteria established in 10 C.F.R. Part 100, Appendix A. Yonk, ff. Tr. 432, at 4, 5; Alterman, ff. Tr. 753, at 3, 4; Tr. 567 (Woodard).

F-30. Although the Illinoian till that overlies the faults in the Byron area (including the Sandwich and Plum River Faults) is undisturbed, displaced glacial till and blocks of bedrock near the Plum River Fault have been observed. The ISGS interprets these displacements as "ice shove" structures attributed to glacial movement during the Pleistocene period, and not of tectonic origin. ISGS Circular 395, at 17; Staff Ex. 1, at 2-23.

F-31. In oral testimony directed to the characteristics of rock formations to be tested to establish the capability of a fault, as specified by the NRC Regulations, Intervenor's witness could not present evidence of surface ground motion at or near faults in northern Illinois within the past 35,000 years; he was not aware of recurring motion along any fault within the past 0.5 My; he could not agree with reports of recorded macro-seismicity in the Byron area detected by appropriate instrumentation; he could not testify that no observed fault was capable. His reservation, concerning the Plum River Fault, arose from the absence of direct excavation to show the absence of disturbance of the overlying till and the concomitant dependence of "indirect" observations such as seismic refraction. To the contrary, the witness could not say that the Plum River Fault is capable. Tr. 561-75 (Woodard); Yonk, ff. Tr. 478 (432), at 7.

F-32. According to Applicant's witness, there is no evidence of motion within any fault at or near the Byron site within the past 0.2 My. Yonk, ff. Tr. 432, at 6.

F-33. There are no known capable faults in the United States east of the Rocky Mountains. Tr. 862 (Rothman); Tr. 869 (Alterman).

F-34. The Board heard no conclusive evidence that any one of the fault areas identified as being near the site of the Byron Station is capable as defined in Commission Regulations. These fault areas are the Plum River, the Sandwich and the minor displacements observed during excavation on the site itself. The evidence that none is capable is persuasive. All parties agreed that the last two were definitely not capable. Of the aspects stated in the Regulations as being necessary to identify the capability of a fault, the only one seriously challenged by the Intervernor as possibly existing at Plum River was whether recurring motion existed within the past 0.5 My. Even Intervenor agrees that there has been no movement at Plum River during at least the last 35,000 years. While presenting no evidence that would demonstrate the capability there, Intervenor contended that neither the ISGS nor the Applicant made what Intervenor considers to be the critical observation necessary to demonstrate noncapability. That observation should have been a direct examination of the Illinoian till overlying the fault made at an excavation. Then the absence of any displacement of the till at the fault would have conclusively demonstrated its noncapability. While it is true that this observation was not made and its accomplishment might well have resulted in this issue not being litigated, the information presented was considerable and convincing. The Board relied principally on testimony presented by the Staff based on and supported by the observation and analysis of data by the ISGS and reported in its Circular 491. These arguments by the Staff, leading to the conclusion that the overlay of till has not been disturbed in recent geologic times, include the absence of an escarpment at the fault, the equality of the elevation of the bedrock strata bordering the fault even though those strata are of different ages, and the tectonic history of the region which includes no record of local seismicity. Additionally, the finding of no fault in northern Illinois which has displaced overlying Illinoian-age soil and that there are no known capable faults in the United States east of the Rocky Mountains assisted the Board in concluding that the noncapability of the Plum River Fault Zone has been sufficiently demonstrated to support our decision that no movement has occurred at Plum River within the past 0.13 to 0.40 My.

*5. Application of Strain Gages*

F-35. Dr. Woodard described the purpose of strain measurements as being to find out whether there is differential strain in the rock on opposite sides of the faults because if there is, it is that differential strain that causes the fault motion. He had no opinion on what strain rate to expect, assuming the Plum River Fault were capable. Tr. 622.

F-36. Dr. Woodard testifed that the testing needs to be performed at depths greater than 3500 feet. However, he has no definite information that strain gage testing was ever performed at this depth (Tr. 584, 611 (Woodard)), nor at what depths it could be performed (Tr. 717, 729 (Woodard)). Dr. Woodard has never used strain gages in the field (Tr. 534 (Woodard)) nor does he know specifically how to install a strain gage in a rock (Tr. 627 (Woodard)). The specific type of strain gages that Dr. Woodard has used in the laboratory could not be applied to downhole strain measurements. Tr. 618 (Woodard).

F-37. The evidence indicates that given the present state of technology, strain gage testing such as recommended by Dr. Woodard at a depth of 3500 feet or greater is not feasible. Tr. 717, 729-32, 734, 742-43 (Woodard); Applicant Ex. 1.

F-38. A technique for translating strain measurements to predicting faults, given the many factors involved in strain and in straining a particular rock, is beyond the current state of knowledge. Tr. 783 (Alterman).

F-39. Even if such techniques were available, the fact that there has not been movement along the Plum River or Sandwich Faults in at least the last 125,000 years, and most likely not since Pennsylvanian time (290 Mybp), coupled with the lack of historic earthquake occurrences, indicates that strain is minimal and therefore that neither earthquakes nor movement is likely to occur on this zone. Alterman, ff. Tr. 753, at 6, 7.

F-40. The League does not address this issue in its proposed findings. The Board assumes the issue has been abandoned. In any event, the record developed on the need and application of strain gages to measure differential strain in rock on opposite sides of faults in the Byron area indicates that such applications even if within the state of the art (which they are not) would be of limited or no value because of the current state of knowledge concerning what to do with the results of such testing. Intervenor witness Woodard testified that the strain gage testing would need to be performed at depths greater than 3500 feet. However, as noted, Dr. Woodard has never used strain gages in the field and had no information that strain gage testing was ever performed at this depth or at what depth the tests could be performed. The additional fact that there has been no movement on this zone in at least the last 125,000 years and most likely not since Pennsylvanian time (290 Mybp), coupled with the lack of earthquake occurrences, indicates that strain is minimal and that neither earthquakes nor movement is likely to occur in this zone.

### 6. *Seismic Design*

F-41. The Byron plant is designed for a Safe Shutdown Earthquake (SSE) peak ground acceleration value of 0.20*g* and an Operating Basis Earthquake (OBE) peak ground acceleration value of 0.09*g*. Singh, ff. Tr. 479, at 3; Rothman, ff. Tr. 760, at 3-4.

F-42. Seven earthquakes have occurred in northern Illinois between 1804 and 1972. The intensities of these earthquakes were estimated by Paul C. Heigold and ranged from IV to VI on the Modified Mercalli (MM) scale. Tr. 445-46 (Yonk).

F-43. At least one of the seven northern Illinois earthquakes has been reevaluated. The 1909 earthquake near Beloit, Wisconsin was evaluated by Nuttli of St. Louis University as being an MM Intensity VII. Tr. 446-47 (Yonk).

F-44. There is no evidence of any earthquake in northern Illinois with an MM Intensity greater than VII. Tr. 558-59 (Woodard).

F-45. The controlling earthquake for the Byron plant is the 1937 Anna, Ohio MM Intensity VII-VIII earthquake. Singh, ff. Tr. 479, at 5.

F-46. The SSE for Byron is based upon an earthquake with an MM Intensity of VIII, which is higher than any earthquake ever recorded in either northern Illinois or in the entire Central Stable Region. *Id.*; Tr. 849 (Rothman).

F-47. Using studies which considered the intensity versus magnitude of earthquakes experienced in the Central United States, Applicant ultimately selected as the SSE an earthquake magnitude value of 5.8. This value is conservative due to the fact that the studies indicate that for earthquakes in the Central United States, an MM Intensity VIII earthquake corresponds to a magnitude of 5.75. Singh, ff. Tr. 479, at 5.

F-48. The magnitude of the 1937 Anna, Ohio earthquake is estimated to range from 5.0 to 5.3. The magnitude of the largest historical earthquake in the Byron area, the May 1909 northern Illinois earthquake, is estimated to be 5.1. *Id.*

F-49. In order to demonstrate the appropriateness of the 0.2*g* ground motion value selected for the SSE, it was compared with the site-specific response spectrum calculated for TVA's Sequoyah Nuclear Power Plant. Singh, ff. Tr. 479, at 6.

F-50. Based on a comparison of the Byron SSE ground motion value (0.2*g*) and the Sequoyah site-specific response spectrum, Applicant determined it was not necessary to prepare a site-specific spectrum for Byron. *Id.*

F-51. The Sequoyah site spectrum was generated for a 5.8 magnitude earthquake, based on real accelerograms of earthquakes recorded at rock sites, at epicentral distances of less than 25 kilometers. A Byron site-specific design basis response spectrum would have utilized these same parameters and the results would have been the same. *Id.*; Tr. 497.

F-52. A comparison of the Byron design basis response spectrum (0.2*g* acceleration anchoring a Regulatory Guide 1.60 spectrum at the foundation level of the structures founded on rock) with the Sequoyah site response spectrum showed that the Byron SSE-based spectrum was conservative in that the Byron SSE response spectrum exceeded the Sequoyah site-specific response spectrum at all frequencies. Rothman, ff. Tr. 760, at 3; SER at 2-27.

F-53. On voir dire examination, Intervenor's witness, Dr. Woodard, admitted he is not a seismologist and does not consider himself an expert with respect to determining the appropriate ground acceleration for which a structure should be designed based upon the geology and seismology of the site. Tr. 522-23 (Woodard). Dr. Woodard candidly disavowed any knowledge on how to calculate the seismic design basis for a nuclear power plant. Tr. 528-29. Finally, Dr. Woodard testified that he did not know how earthquake intensity, magnitude or peak ground acceleration parameters are utilized in developing the seismic design for a nuclear power plant. Tr. 589-90.

F-54. On July 5, 1982 at 04:13:49.81 GMT (July 4, 1982, at about 11:14 p.m. CDT) there was a magnitude 3.8 earthquake with an epicentral location of 35° 11.1′ North latitude, 92° 13.72′ West longitude near the town of Enola, Arkansas. This earthquake was one of over 20,000 small earthquakes which have occurred in the area since about January 12, 1982. An SMA-1 strong motion seismograph was located about 200 meters from the epicenter and recorded a peak acceleration of 0.59*g* on its east-west component. Another strong motion seismograph, a DR-100 which was co-sited with the SMA-1, recorded a peak horizontal acceleration of 0.19*g*. Rothman, ff. Tr. 760, at 6.

F-55. The discrepancy in acceleration between the co-sited SMA-1 and the DR-100 instruments is currently unexplained. The Tennessee Earthquake Information Center, the agency which monitored the earthquake, has stated: "A distinct possibility is that the high SMA-1 acceleration is an installation effect and does not represent a true field acceleration." The entire earthquake recording had a duration of about 3 seconds and the high acceleration had a frequency of about 14 Hz. *Id.*

F-56. Staff witness Rothman testified that no significance can be attached to the high SMA-1 acceleration and that, if indeed this acceleration is not due to installation effects, then it would represent a very close (near-field) high-frequency, short-duration record of an earthquake with little energy. There was no damage reported from this earthquake to the shed in which the SMA-1 instrument is located or to any other building. Since there was no damage to these buildings which were not designed to withstand earthquake motion, there is no reason to believe that earthquake motion of this type could cause damage to a nuclear power plant which is designed using a broad-band response spectrum which encompasses the wider frequency range and higher energies of larger earthquakes. Rothman, ff. Tr. 760, at 7; Tr. 807.

F-57. Other small earthquakes with high peak accelerations in the near field have been recorded. For example, a peak acceleration of 0.25*g* was recorded from an earthquake of magnitude 2.7 in South Carolina, a peak acceleration of 0.7*g* was reported from a magnitude 4.75 earthquake in California, and small mine tremors due to rock bursts have had recordings of 12*g* in near field from these events. These are small events which are recorded in the near field and "[t]hey don't really have any energy." Tr. 810 (Rothman).

F-58. Intervenor witness Woodard testified that he would not expect that the Enola, Arkansas earthquake would do anything to a nuclear power plant because it was such a low-energy event. Tr. 587-88.

F-59. In order to determine the appropriate ground acceleration value for the Operating Basis Earthquake (OBE), the Applicant focused on the earthquake and associated ground acceleration, which could reasonably be expected to affect the plant during its 40-year life. The earthquake selected has an MM Intensity of VI and peak ground acceleration of 0.09*g*. Singh, ff. Tr. 479, at 6.

F-60. The expected recurrence of the OBE was calculated by Applicant to be approximately 2,150 years. The Lawrence Livermore National Laboratory (LLNL), a consultant to the NRC Staff, estimated the recurrence interval to be in the range of 200 to 1,000 years. A third estimate by Dr. Robert B. Hermann of St. Louis University predicts a return period on the order of 1000 years for peak accelerations of about the OBE level in the Byron site area. *Id.* at 6, 7; Rothman, ff. Tr. 760, at 5; Tr. 757-58 (Rothman).

F-61. The difference in recurrence interval estimates between the Applicant and the LLNL are most probably due to the different methods and assumptions used. *Id.*

F-62. To meet the better definition of the OBE as specified in 10 C.F.R. Part 100, Appendix A, Section III(d), the NRC Staff has accepted OBE acceleration values of less than one-half the maximum vibratory ground acceleration of the Safe Shutdown Earthquake for some sites. This is done when supporting data, such as probabilistic analyses of earthquake hazards, justify it. Additionally, in the Byron case, from a seismological viewpoint, the difference between a Regulatory Guide 1.60 spectrum anchored at 0.09*g* and one anchored at 0.10*g* is less than the scatter of the data. *Id.*

F-63. The Board agrees that Applicant has provided adequate supporting data to justify an OBE of 0.09*g*.

F-64. Intervenor asserts that Applicant ignored the only instrumentally measured data that directly correlate a magnitude value of an earthquake occurring in similar kinds of rock at Byron with a ground acceleration value and therefore the ground acceleration values selected for the SSE and OBE are not conservative enough. Woodard, ff. Tr. 548, at 3; Tr. 601-04. As pointed out in both Applicant and Staff testimony, the Sequoyah site-specific spectrum which was utilized in determining the ground acceleration value for the Byron SSE is based on instrumentally measured data, *i.e.*, accelerograms of earthquakes recorded at rock sites similar to Byron and at epicentral distances of less than 25 kilometers. Singh, ff. Tr. 479; Rothman, ff. Tr. 760, at 3. The record clearly demonstrates the applicability of the Sequoyah spectrum to the Byron site and while the high SMA-1 acceleration discrepancy at Enola, Arkansas is not totally understood, and there have been some other small earthquakes with anomalously high peak accelerations, it appears that such near-field, high-frequency, short-duration earthquakes would not cause any damage to a nuclear power plant such as Byron, which is designed to resist the broad-band acceleration spectrum associated with the larger-energy earthquakes. Intervenor's allegation that the ground acceleration values selected for the SSE and the OBE are not sufficiently conservative is not supported by the record. The Board finds that Applicant has demonstrated compliance with 10 C.F.R. 50.57 and 10 C.F.R. Part 100, Appendix A regarding the seismic design of the Byron plant.

## G. Emergency Planning

G-1. On February 21, 1983, Intervenors filed their "Amendment and Consolidation of DAARE/SAFE Contention 3 and Rockford League of Women Voter's Contentions 19 and 108" (hereafter the Revised Contention) which raised various emergency planning issues in thirteen separate paragraphs. By stipulation of the parties, it was agreed that Intervenors would withdraw previously accepted DAARE/SAFE Contention 3 and League Contentions 19 and 108, that certain paragraphs of the Revised Contention would be litigated, and that the remaining paragraphs of the Revised Contention would be resolved informally outside of the hearing process. On August 22, 1983 the Commission approved a proposed settlement and Board recommendation extending the Board's jurisdiction to conduct any further hearings on the remaining paragraphs if necessary after an initial decision and the issuance of a full-power license.

G-2. The parties agreed to litigate three subparagraphs of paragraph 2 of the Revised Contention which concerns Applicant's "Evacuation Time Estimates Within the Plume Exposure Pathway Emergency Planning Zone for the Byron Nuclear Generating Station." Applicant Ex. 18. The parties also agreed to litigate paragraph 3 of the Revised Contention which concerns emergency medical facilities; paragraph 8 concerning emergency protective actions; paragraph 10 concerning reliance on volunteers during emergencies and paragraph 13 which concerns emergency planning coordination and communications. At the outset, we recognize that, in contrast to the findings we make on other issues in the Byron proceeding, some of our findings on emergency planning issues are only predictive. For example, we make findings on paragraphs 8 and 13 even though at the time of the hearing, school evacuation plans were not yet in place, and final plans for communications with emergency response organizations were still being formulated. To make these findings, we've had to rely to some extent on the commitments the Applicant made as part of the settlement agreement on certain paragraphs of the Revised Contention, and on the strength of the showing certain witnesses made about what will be in the final plans.

G-3. Although the treatment we give emergency planning issues in our findings is unusual when compared with the treatment we give other issues here, we are only reflecting the treatment the Commission's regulations accord emergency planning issues. This treatment was clearly expounded recently by the Appeal Board in *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1103-04 (1983):

> With respect to emergency planning, however, the Commission takes a slightly different course. At one time, the agency's regulations required a finding that "the *state* of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." 10 C.F.R. § 50.47(a)(1) (1982) (emphasis added). In July 1982, the Commission amended this provision by clarifying that "the findings on emergency planning required prior to license issuance are predictive in nature" and by eliminating the reference to the "state" of emergency preparedness. 47 Fed. Reg. 30232,

30235 (July 13, 1982), *petition for review pending sub nom. Union of Concerned Scientists v. Nuclear Regulatory Commission*, No. 82-2053 (D.C. Cir. filed Sept. 10, 1982). The notice of proposed rulemaking that preceded this amendment expressed the Commission's intent that "full-scale emergency preparedness exercises [be] part of the operational inspection process and [be] required prior to operation above 5% of rated power *but not for a Licensing Board, Appeal Board or Commission licensing decision.*" 46 Fed. Reg. 61134 (Dec. 15, 1981) (emphasis added). *See also* 47 Fed. Reg. at 30232. The Commission emphasized, however, that "there should be reasonable assurance prior to license issuance that there are no barriers to emergency planning implementation or to a satisfactory state of emergency preparedness that cannot feasibly be removed." 46 Fed. Reg. at 61135. Thus, while the plan need not be "final," it must be sufficiently developed to permit the board to make its "reasonable assurance" finding in a manner nonetheless consistent with the guidance of [*Consolidated Edison Co. of New York* (Indian Point Station, Unit No. 2), CLI-74-23, 7 AEC 947, 951 (1974), discussed, *supra*, Paragraph D-420] and its progeny. *See* [*Cincinnati Gas & Electric Co.* (Wm. H. Zimmer Nuclear Power Station, Unit No. 1), ALAB-727, 17 NRC 760 (1983)], at 770, 773; *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), ALAB-717, 17 NRC 346, 380 n.57 (1983).

G-4. The Commission was moved to make these amendments to 10 C.F.R. 50.47(a) in part because of the analogy it saw between emergency preparedness exercises and "the many other preoperational, startup, or operational tests required by NRC regulations or license conditions." 46 Fed. Reg. 61,134 (1981). It was in recognition of the fact that emergency planning cannot proceed on the same schedule other litigable aspects of the Applicant's responsibilities can proceed on, that the Byron Intervenors agreed to defer litigating many emergency planning issues in exchange for certain commitments from the Applicant as to the content of the final plans.

G-5. These findings are limited to those issues which the parties agreed to litigate. The applicable law for each litigated paragraph of the Revised Contention will be set out when we discuss each paragraph. That law is drawn from the general standards in 10 C.F.R. 50.47(b) and the more specific evaluation criteria of NUREG-0654, FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, November 1980.[85] But there is one rule which applies to all the paragraphs and so is stated now: No operating license will be issued

[85] Although FEMA-REP-1, NUREG-0654, is not a Commission regulation and is not enforceable as such, it is the federal guidance referred to in the Commission emergency planning regulations (10 C.F.R. 50.47(b) n.1, and 10 C.F.R. Part 50, Appendix E, IV.C & nn. 1, 4) and has been accepted by the parties and the Board in this proceeding, with minor exception, as reasonable emergency planning guidance. The Commission recently gave its blessings to FEMA-REP-1, NUREG-0654 as reasonable federal guidance by requiring precise adherence to its standards. *See Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), CLI-83-22, 18 NRC 299, 307-09 (1983).

unless "there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." 10 C.F.R. 50.47(a)(1).

### *1. Paragraph 2, the Evacuation Time Study*

G-6. The Evacuation Time Study was prepared by an independent consultant, Stone and Webster Engineering Corporation, under contract to Applicant. The litigated portion of paragraph 2 states:

> In violation of 10 C.F.R. Section 50.47(b)(10), Commonwealth Edison's "Evacuation Time Estimates for the Plume Exposure Pathway Emergency Planning Zone of the Byron Nuclear Generating Station" does not conform to NUREG-0654, Appendix 4 and will not provide accurate or useful guidelines for the choice of protective actions during an emergency because the study:
>
> * * * * *
>
> (c) does not address the relative significance of alternative assumptions;
>
> * * * * *
>
> (e) does not consider the impact of peak populations, including behavioral aspects;
>
> * * * * *
>
> (k) does not use site weather characteristics as presented in the FSAR, . . .

Section 50.47(b) requires that:

> The onsite and offsite emergency response plans for nuclear power reactors must meet the following standards:
>
> * * * * *
>
> (10) A range of protective actions have been developed for the plume exposure EPZ [emergency planning zone] for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

This standard is addressed by a specific criterion in NUREG-0654, which states, *inter alia*:

> 10. The organization's plans to implement protective measures for the plume exposure pathway shall include:
>
> * * * * *
>
> l. Time estimates for evacuation of various sectors based on a dynamic analysis (time-motion study under various conditions) for the plume exposure pathway emergency planning zone (*See* Appendix 4); . . .

NUREG-0654, at 61-63. The referenced Appendix 4 gives "an example of what shall be included in an evacuation times assessment study and how it might be presented." NUREG-0654, at 4-1.

G-7. Applicant presented the testimony of Ms. Jean L. McCluskey and Dr. Thomas J. Horst, Stone and Webster employees, who appeared as a panel. Ms. McCluskey is the Project Manager for the Byron Station Evacuation Time Study. Dr. Horst is responsible for the technical aspects of the Study. Ms. McCluskey and Dr. Horst testified as to the purpose, assumptions, and limitations of the Study. Dr. John Golden, Applicant's Supervisor of Emergency Planning, also answered questions pertaining to the Study as did Mr. David Smith, who is Chief of Field Services for the Illinois Emergency Services and Disaster Agency (IESDA).

G-8. The NRC Staff presented the testimony of Dr. Thomas Urbanik with the Texas Transportation Institute of Texas A&M University, who is responsible to the NRC for reviewing evacuation time estimates for nuclear facilities. Mr. Monte B. Phillips, an Emergency Preparedness Analyst with the NRC, also addressed the Study.

G-9. Mr. Paul Holmbeck, who has a well-informed layman's knowledge of radiological emergency response plans was the coordinator for the Intervenors' emergency planning contention and was their lead witness. His affidavit (Intervenors Ex. 13) includes comments on the Evacuation Time Study.

G-10. The primary purpose of the Evacuation Time Study is to analyze the feasibility of evacuation of the Byron Plume Exposure Pathway Emergency Planning Zone (Plume EPZ). The Study is not an evacuation plan which would be implemented in an emergency. It is an assessment of representative time frames for the evacuation of various areas around the Byron Station for a range of seasonal, time-of-day, and weather conditions. It identifies the approximate time frames associated with evacuation based on a detailed consideration of roadway network and population distribution. It also identifies the assumptions upon which the time estimates are based. The authors anticipate that the Study will be useful to state and local emergency officials to assist them in determining the relative feasibility of evacuation as a protective action. McCluskey and Horst, ff. Tr. 4834, at 4, 5.

G-11. There are various reasons for making evacuation time estimates. During the process of making the estimates, one identifies potential bottleneck or congestion areas where queuing or backup could occur. Most importantly, these estimates provide decisionmakers with information on which to base a protective action choice between sheltering and evacuation during an emergency. Phillips, ff. Tr. 5509, at 2-3.

*2(c) Significance of Alternative Assumptions*

G-12. The phrase "relative significance of alternative assumptions" within the meaning of this subcontention is found on page 4-7 of NUREG-0654, Appendix 4. The term has special relevance to time-dependent traffic loading of segments of the evacuation roadway network. *Id.* Appendix 4 also identifies the following alternative assumptions: (1) normal versus adverse weather conditions, (2) day versus night, (3) workday versus weekend, (4) peak transient versus off-peak transient, and (5) evacuation of adjacent sections versus nonevacuation.

G-13. However, in their emergency planning proposed findings quoted below, Intervenors correctly acknowledge that the significance of alternative assumptions is limited because of the rural character of the Byron Plume EPZ:

> 7. The analysis in the Byron evacuation study is not sensitive to many assumptions to which an analysis for a more densely populated site might be sensitive. (Urbanik Tr. 5399.)
>
> 8. The largely rural Byron area has a low population density. Roadway capacity is not a constraining factor on evacuation time. The sensitivity of the study is to other assumptions namely preparation and mobilization time. Preparation time controls the evacuation time. (Urbanik Tr. 5400, 5403, 5410, 5414.)

G-14. Intervenors point to Dr. Urbanik's testimony that preparation time for the Byron site may be "an inordinately long preparation time compared to most sites in the U.S." (Tr. 5414), apparently as an indication that the estimates may be inaccurate. Intervenors' Proposed Finding 9. However, Dr. Urbanik rejected this implication, and on reflection, indicated that "inordinate is a bad choice of words" preferring "larger than average." This circumstance he attributes to the fact that there are many farm people in the Byron EPZ who take longer than average to prepare. *Id.* Accordingly, we do not find that the longer-than-average preparation time for Byron is, in itself, an indication of inaccuracy.

G-15. Intervenors identify several additional factors which, in their view, bring the estimates of preparation time for Byron into question: First, although the evacuation study identifies major employers in the area, there is no surveying of who is employed at the employers' sites or how long those employers would require for shutdown. Proposed Finding 12. A related criticism is that Dr. Horst and Ms. McCluskey have gathered no data on where people work relative to their respective homes, this factor being relevant because the travel time to home must be added to preparation time. Proposed Finding 13. The third criticism is that there has been no indication that the time required to evacuate farms has been considered in the evacuation study. Proposed Finding 14.

G-16. On the other hand, Dr. Horst testified that a major source of data for the Study included information on the number of people that live in the area as well as how they are distributed throughout the area and the population of special facilities and transients. Tr. 5013. In acquiring the data concerning the population around the Byron plant, County and State officials who are familiar with the area were consulted to determine how long it would take the local population to prepare for an evacuation. Tr. 5013-14 (Horst). Further, the average public response times for receiving warning, leaving work, traveling home, and evacuating the home have been favorably reviewed by Ogle County and IESDA officials. Applicant Ex. 18, at 4-3. Ms. McCluskey also indicated that the Evacuation Time Study incorporated the experience of evacuations in similar types of rural areas. Tr. 5016.

G-17. The Study applied approved probability distribution techniques to Byron-specific data. A normal distribution was assumed for the time spread of specific events, *i.e.,* receiving warning, leaving work, travel time from work to home, and leaving home. Normal distribution represents the situation in which most persons respond in the average time for the given event and lesser numbers of individuals respond earlier and later than the average time. Therefore, the cumulative probability distribution of each of these events is an "S" shape. The curves have been derived by using standard mathematical techniques based on NUREG-0654 recommendations. Nighttime curves have been derived by combining two events: Receive Warning and Evacuate Home. Daytime curves have been derived by combining the Receive Warning, Leave Work, Travel Home, and Evacuate Home time distributions. The cumulative distribution of these different events combined has an "S" shape similar to the curves for the individual events, and represents the spectrum of public response times. Applicant Ex. 18, at 4-3, 4-4.

G-18. The Board is satisfied that the preparation-time estimates for the Byron Evacuation Time Study are reasonably accurate and that they satisfy the expectations of NUREG-0654 with one exception. Intervenors' concern that the preparation estimates do not include time to shut down employment sites has not been directly answered by any portion of the evidentiary record we can identify. For that matter, however, neither can we discern that Intervenors raised the issue during the hearing.

G-19. The Study estimates that the average time from receiving a warning to leaving work in the Byron area would be 15 minutes. Applicant Ex. 18, Table 4-1. Whether this includes the employment center shutdown time estimate is not clear. Given the basically rural nature of the EPZ and the absence of heavy industry in the area, it would seem that the 15-minute estimate would include shutdown time. Even if some employment centers need more than 15 minutes to shut down, not all employees would be required for that purpose and the 15-minute average could still pertain.

G-20. The Stone and Webster witnesses reported that the major employers of the area have been identified. Tr. 5014. Therefore, the uncertainty can be easily resolved and should be. If the estimate includes shutdown time, that fact should be stated. If not, perhaps reliable inferences can be drawn based upon the nature of the employment centers. Where uncertainty remains, specific inquiry would not be burdensome. Identification of employers with extended shutdown times would also be useful in identifying employees who should be regarded as emergency workers. The Board will require that the Evacuation Time Study be clarified and amended if necessary to reflect employment center shutdown times.

### *2(e) Consideration of Peak Populations and Behavioral Aspects in the Evacuation Time Study*

G-21. Intervenors adopted this portion of their emergency planning contention from NUREG-0654, at 4-10 which requires that behavioral aspects be considered when considering the impact of peak populations.

G-22. The Byron Evacuation Time Study considers peak populations in two ways. First, the Study considers summer and winter populations. Summer populations include transient populations resulting from recreational facilities in the area. Second, special events which attract significant numbers of additional transients are analyzed in separate simulations. Certain special events in the plume exposure pathway EPZ could attract significant numbers of additional transients. These special events are the Autumn on Parade Festival and the Byron Dragway and Motosport Speedway events. These events were analyzed in separate simulations. Based on these simulations, the analysis determined that the presence of additional transient populations associated with the special events do not increase the time required to evacuate. McCluskey and Horst, ff. Tr. 4834, at 6; Applicant Ex. 18, at 6-2.

G-23. Intervenors would have the Board infer that the Evacuation Time Study estimate with respect to certain special events is flawed because queuing would occur at those times. Proposed Finding 19. The Time Study considered queuing in its conclusion, however (Applicant

Ex. 18, at 7-1), and Intervenors are too late in their effort to challenge the conclusion.

G-24. The principal thrust of Intervenors' "peak population" subcontention is that the NETVAC II Dynamic Route Selection model used in the Time Study employs two invalid assumptions:

1. While peak population transients are correctly assumed to know the routes taken into the evacuation zone, this knowledge would not extend to knowing the proper evacuation route. Proposed Findings 25, 26.
2. The assumption that persons unfamiliar with an evacuation route will follow others out is not consistent with the assumption that persons will choose the less congested path. Proposed Findings 23, 29-32. And in any event, the assumption that the lead car driver knows where to go is invalid. Proposed Findings, Opinion, at 10.

G-25. The persuasive testimony of the Time Study authors does nothing more than observe ordinary human behavior when they testified that transients will not be evacuating in a vacuum, that they will not travel opposing a crowd, and will not necessarily join in a congestion when a close-by alternative is observed and is used by others. Tr. 4879 (McCluskey and Horst). The Board believes that the situation postulated by the Intervenors' proposed findings is simply not logical. Transients with the intelligence, information, resourcefulness and temperament to venture into an area unfamiliar to them can be expected to employ the same faculties in finding their way out of the area, albeit by a different route. The same resourcefulness will permit them to follow others out of the area, to adjust from time to time to minimize congestions and to find routes on their own if necessary. The scenario suggesting that a crowd, some of whom are familiar with the area, will continue to follow a lead car in the wrong direction for evacuation is implausible. Moreover, Intervenors' postulation ignores the contribution toward orderly evacuation which would be made by police and other traffic control workers in the event of a radiological emergency.

G-26. We find that the behavioral aspect of peak populations in the Byron EPZ has been adequately addressed in the Evacuation Time Study.

### Aberrational Behavioral Aspects

G-27. Although by its terms, Intervenors' subcontention in subparagraph 2(e) narrowly charges that the Evacuation Time Study does not "consider the impact of peak populations, including behavioral aspects," their proposed findings somewhat unfairly enlarge the scope of the issue by asserting that aberrational behavior, especially panic, in the population at large, has not been properly evaluated. Intervenors presented no evidence, electing instead to analyze perceived faults in the evidentiary presentation by the Applicant and Staff. Proposed Findings 43-50.

G-28. The Evacuation Time Study made several behavioral assumptions, including those at issue here, *i.e.*, persons within the Byron evacuation zone will leave when requested, and will comply with traffic rules. Ms. McCluskey and Dr. Horst testified that:

> These assumptions are based in part on the findings represented in an Environmental Protection Agency publication entitled "Evacuation Risks — An Evaluation" published in June 1974. This publication analyzes information regarding human reactions to actual evacuations, and concludes "the idea that people will panic in the face of great threat or danger is widespread. However, it is not borne out in reality. Insofar as wild flight is concerned *the opposite behavioral pattern in most disasters is far more likely.*" [Emphasis in original.]

McCluskey and Horst, ff. Tr. 4834, at 7.

G-29. Contrary to Intervenors' assertion that "opposite behavioral pattern" could mean "complete passivity and non-responsiveness" (Proposed Finding 44), Ms. McCluskey clearly stated that by "opposite behavioral pattern" she meant that people will not panic and not flee. Tr. 4867. Ms. McCluskey, when questioned by Intervenors' representative, added her own experience during a Nor'easter with accompanying flooding, winds, snow and rain as a basis for believing that people act rationally and responsibly during emergency evacuations. Tr. 4890-93 (McCluskey, Holmbeck).

G-30. Dr. Urbanik testified that experience with large-scale evacuations does not reveal any aberrant behavior on the part of evacuees despite their large numbers and stressful environment. This conclusion is based on examination of the literature concerning evacuations, including large-scale evacuations in Texas and Louisiana during Hurricane Carla and the evacuation of Missasauga, Canada (216,000 people) following a train derailment involving hazardous chemicals. Urbanik, ff. Tr. 5391, at 6.

G-31. The Board does not regard the evidence presented to it to constitute a fully litigated, adversarial evidentiary presentation on whether persons would flee in panic, or refuse to move at all. Intervenors, as we noted, had no evidence. But, nothing in the Evacuation Time Study subcontentions required a greater evidentiary showing by Applicant or the Staff. The evidence presented was reliable, probative and substantial, and constituted the preponderance, in fact all, of the evidence on the issue. Moreover, the implicit assumption in Intervenors' proposed findings that panic or its opposite, total immobility of the general population is, in our view as finders of the fact, inconsistent with normally expected human behavior under evacuation conditions. The Evacuation Time Study has given sufficient consideration to the possibility of aberrational behavior.

Parents of Schoolchildren

G-32. The Study assumes, as we noted above, that persons will evacuate when requested to do so. Intervenors assert that parents of school-age children will not leave without their children then in school. Proposed Findings 33-35. Ms. McCluskey testified that the parents are no exception to the general rule provided that they have been informed beforehand concerning the plans for evacuating their children. *E.g.*, Tr. 4997. Intervenors, however, meet this testimony by pointing out — accurately as is presently the case — that there is no evidence that arrangements for such timely reassurances have been made. Proposed Finding 39. This is not a completely fair argument. The adequacy of the evacuation plans for schools is provided for in Commitment I of the emergency planning stipulation, and the adequacy of the public information and educational programs is the subject of Commitments Q through T. Applicant acknowledges in its Reply Findings that information for parents is to be included in its public information and education commitments. Reply Findings at 6.

G-33. Therefore, assuming the adequacy of the schoolchildren evacuation plans and the effectiveness of the plans to communicate the school evacuation information to parents, we find no basis to conclude that the Evacuation Time Study is deficient with respect to parents of schoolchildren.

G-34. Nevertheless, the Board is sympathetic to the special concern implicit in Intervenors' proposed findings. Schoolchildren and their parents present a special case for evacuation planning. We observed, with some concern during the hearings and in the limited appearance statements, that school evacuation plans were not in place and there was virtually no public understanding of school evacuation expectations. We will regard this aspect of Applicant's commitments to be an especially important consideration in the event of a request for hearing.

Adverse Weather Conditions

G-35. According to NUREG-0654, an evacuation time analysis must include:

> Two [weather] conditions — normal and adverse — .... Adverse conditions would depend on the characteristics of a specific site and could include flooding, snow, ice, fog or rain. The adverse weather frequency used in this analysis shall be identified and shall be severe enough to define the sensitivity of the analysis to the selected events. These conditions will affect both travel times and capacity. More than one adverse condition may need to be considered. That is, a northern site with a high summer tourist population should consider rain, flooding, or fog as the adverse condition as well as snow with winter population estimates.

*Id.* at 4-6, 4-7.

G-36. The Byron Evacuation Study selected several adverse conditions, light snow, icing, rain, and fog, all of which come within the definition of adverse weather which reduces road capacity to 70 percent of normal weather capacity. Applicant Ex. 18, at 2-3. Translated into time differences, for normal weather, summer and winter, the general population evacuation time estimates for the full plume EPZ range from about 191 minutes during the day to 111 minutes at night. In adverse weather, the estimates range from about 227 minutes for day and 123 minutes for night. *Id.* at 1-1, 1-2. Other time estimates for differing scenarios, *e.g.*, special facility evacuation, were also calculated. *Id.*

G-37. The Intervenors level two criticisms of the use of the adverse weather factors in the Study. First they state that there is doubt that decisionmakers will know what 70 percent roadway capacity looks like. Responding to this, Applicant states that the decisionmakers need not know what 70 percent of roadway capacity looks like, that it is the sensitivity of the time estimates to adverse weather conditions which controls. We agree with Applicant but for simpler reasons. The decisionmakers may not know what 70 percent of road capacity looks like, but they know what rain, icing, fog and light snow look like and they can recognize variations of those conditions. As noted in the preceding finding, the reduced roadway capacity factor has also been set out in the form of time variations for the guidance of the decisionmakers.

G-38. Intervenors also complain that more adverse weather conditions, such as flooding in the summer and heavy snow in the winter, should have been included as additional assumed adverse weather conditions. Indeed, NUREG-0654 contemplates that more than one adverse weather condition may be needed, for example, in northern sites. Ms. McCluskey and Dr. Horst explained that the more adverse conditions were not used because:

> Obviously snow and icy pavements in the extremes identified as "snowfall in excess of six inches and often accompanied by damaging glaze" can effectively reduce the capacity to zero. However, because such conditions occur, on the average, about once per year, it was decided that the evacuation time estimates should address the most common adverse conditions, thereby providing officials a more useful aid in making decisions regarding protective actions.

Ff. Tr. 4834, at 8.

G-39. But this reasoning does not preclude the use of additional adverse weather conditions if such information would be useful. However, both Ms. McCluskey and Dr. Horst in their testimony and Dr. Urbanik in his testimony point out that the feasibility of no evacuation at all must also be considered. In heavy snow, for example, evacuation would not begin until roads are cleared. *Id.* at 8, 9; Urbanik, ff. Tr. 5391, at 5. The adverse weather condition used in the Study and the assumed reduced capacity factor is intended to represent the upper limit where roads are passable but not in good condition. Tr. 4396 (Urbanik). Accordingly the Board finds that the Evacuation Time Study has employed reasonable adverse weather conditions in its assumptions.

G-40. The Board, however, is concerned about another aspect of the adverse weather assumptions, apparently overlooked by Intervenors in their proposed findings. In Applicant's view the assumed 30 percent reduction in capacity for adverse weather is "conservative." Proposed Finding 260. According to Dr. Horst, who selected the capacity reduction assumption, the reduction for rainfall, for example, could actually be as little as 15 percent. The overall study indicated only a reduction to 80 percent capacity. But Applicant's study went "a little bit further and used 70 percent instead." Tr. 4965-66; 4981-88. He stated, at one point, that assumed traffic time, increased as a result of the extra 10 percent capacity reduction, was perhaps only "a relatively small percentage." Tr. 4966 (Horst). But Dr. Horst testified inconsistently that at the 70 percent ("conservative") roadway capacity assumption, there may be a major increase in time estimate, but that one may not *see* the increase at 80 percent (realistic) assumed roadway capacity. Tr. 4986.

G-41. It was troubling to the Board that Dr. Horst was not able to explain why conservatism in emergency planning lies in the direction of assuming greater-than-realistic evacuation traffic times. Tr. 4984-88.

G-42. Dr. Urbanik approved the use of an assumed 30 percent reduction in capacity as appropriate to account for site-specific conditions, but he acknowledged that available research indicates a capacity reduction range of only 8 to 24 percent for a variety of conditions including wet weather or light snow. Urbanik, ff. Tr. 5391, at 4, 5. He did not explain why he approved a more-than-realistic assumption in capacity reduction.

G-43. The Board finds no basis for the Applicant's and Stone and Webster's conclusion that reducing the assumed roadway capacity, with its attendant increase in assumed traffic time, is conservative. If evacuation is the only course open to the emergency decisionmaker, overestimating the traffic-time assumptions would not help in making a decision, and in fact evacuation time studies would have greatly reduced importance. Where the decisionmakers must select from more than one protective action, any departure from realistic evacuation time estimates could influence their decisions away from safety. Accordingly the Board will require the Applicant to modify its Evacuation Time Study to reflect realistic traffic time estimates. Conservatisms may remain in the Study provided that they are clearly identified as such and quantified.

## 2. *Paragraph 3, Emergency Medical Facilities*

G-44. The parties agreed to litigate paragraph 3 of the Revised Contention which states:

> In violation of 10 C.F.R. Section 50.47(b)(12), the emergency planning for the ingestion exposure EPZ of the Byron Station does not sufficiently address the fact that there are inadequate medical facilities to provide the required bed space for an evacuation; that there is an insufficient number of medical and para-medical personnel to render medical assistance during an evacuation; that there are insufficient procedures for the screening, treatment, and isolation of persons sustaining radiological injuries; and that there is an insufficient number of materials, supplies, equipment, and vehicles to provide for the transportation of injured persons during a radiological disaster.

Section 50.47(b)(12) provides:

> (b) The onsite and offsite emergency response plans for nuclear power reactors must meet the following standards:
>
> * * * * *
>
> (12) Arrangements are made for medical services for contaminated injured individuals.

G-45. The Applicant presented the testimony of Dr. John C. Golden, Supervisor of Emergency Planning for Commonwealth Edison Company, David Smith, Chief of Field Services for the Illinois Emergency Services and Disaster Agency (IESDA), and David D. Ed, Nuclear Safety Executive with the Illinois Department of Nuclear Safety (IDNS). Monte P. Phillips, Analyst with the Emergency Preparedness Section, NRC Region III, and Gordon L. Wenger, a community planner with the Technological Hazards Branch, Federal Emergency Management Agency, Region V, testified for the NRC Staff.

G-46. The Intervenors presented testimony and exhibits pursuant to stipulation; the affidavit of Paul Holmbeck (Intervenors Ex. 13) and the testimony of James L. Murphy, a public health specialist (Intervenors Ex. 20). Mr. Holmbeck conducted an ambulance survey (Intervenors Ex. 14), admitted in part and rejected in part. Mr. Murphy prepared the questions in the survey based on his experience in the *Indian Point* proceeding.

G-47. The Commission has provided detailed and exact guidance concerning the requirements of 10 C.F.R. 50.47(b)(12) that emergency planning must include arrangements for medical services for "contaminated injured individuals." In *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-83-10, 17 NRC 528 (1983), the Commission clarified the phrase "contaminated injured individuals" and the scope of "arrangements . . . for medical services" to be provided for the public in the event of a nuclear plant accident.

G-48. The Commission stated that the scope of "medical services" to be provided was focused on the radiation hazards which fell into two categories. The first involves individuals who sustain a traumatic (nonradiation) injury requiring emergency medical care and are also externally contaminated with radioactive materials. The second category involves individuals subjected to dangerous levels of radiation and in need of medical treatment for that purpose (without regard to nonradiation trauma).

G-49. With respect to the individuals who become injured and are also contaminated, the Commission concluded that arrangements that are currently required for onsite personnel and emergency workers provide emergency capabilities which should be adequate for treatment of members of the general public. These would include: "(a) local and backup hospital and medical services having the capability for evaluation of radiation exposure and uptake, including assurance that persons providing these services are adequately prepared to handle contaminated individuals, (b) onsite first-aid capability, and (c) transportation capability." The Commission concluded that no additional medical facilities or capabilities are required for the general public. Facilities with which prior arrangements have been made or which have the capability to treat contaminated injured individuals should be identified. The Commission also stated that the number of individuals both onsite and offsite who may become contaminated and injured is expected to be very low.

G-50. With respect to individuals who may be exposed to dangerous levels of radiation, the Commission concluded that medical treatment required less advance planning and can be arranged on an "as-needed" basis because, in regard to radiation injury, while medical treatment may be eventually required in cases of extreme exposure, patients are not likely to need emergency medical care. The Commission determined, however, that emergency plans should identify those local or regional medical facilities which have the capability to provide appropriate medical treatment for radiation exposure. It determined that diagnosis and treatment could take place in most existing medical facilities. The Commission emphasized that no contractual arrangements are necessary and no additional hospital or other facilities need be constructed.

G-51. Applicant's arrangements for treatment of its Byron Station personnel who may suffer a traumatic injury accompanied by contamination include agreements with the Rockford Memorial Hospital for medical services and with the Byron Fire Protection District for ambulance transportation. The Rockford Memorial Hospital was, at the time of the hearing, constructing a new emergency room which will be adapted to facilitate treatment of contaminated injured persons. It was selected, in part, because of that adaptability. Golden, ff. Tr. 5035, at 3-5.

G-52. Personnel from the Rockford Memorial Hospital and the Byron Fire Protection District who may be involved in treatment of contaminated injured personnel will receive annual training in treatment of such injuries from Radiation Management Corporation (RMC), a nationally recognized expert consultant in health physics. *Id.* at 5-6; Ed, ff. Tr. 5174, at 9.

G-53. As part of the RMC service program, RMC will provide inventories of plant and hospital equipment and supplies for use in handling radiation victims. RMC also provides emergency expert consultation and access to its own Radiation Emergency Medical Team and access to a medical center equipped for definitive evaluation and treatment of radiation injuries. Golden, ff. Tr. 5035, at 5, 7, and (Golden) Ex. 6 thereto.

G-54. In addition, the Radiation Protection and Chemistry Department from the Byron plant and Applicant's other nuclear power plants will provide radiation protection and contamination control assistance needed by hospital personnel in treatment of contaminated injured persons. *Id.* at 6.

G-55. The Illinois Department of Nuclear Safety (IDNS) will provide any support needed by hospitals in the treatment of contaminated injured persons. Its staff includes at least six health physicists. The IDNS staff would be on hand in the event of a radiological accident at

Byron. Also IDNS maintains a Standard Operating Procedure for Radiological Decontamination of Personnel that would apply for treatment of members of the public who are contaminated and injured. Ed, ff. Tr. 5174, at 4-5, 8, 9.

G-56. As part of the procedure, IDNS maintains a list of hospitals near nuclear plants capable of handling contaminated injured persons. At the time of the hearing, IDNS was investigating those hospitals in the Byron area to determine which are capable of handling contaminated injured persons. The list of hospitals also includes those under agreement with Applicant to provide medical service for its personnel at all of its nuclear power plants. These are hospitals which receive the RMC training and services. The local emergency support organizations will be apprised of the hospitals capable of handling contaminated injured patients. The next revision of the list will include Rockford Memorial Hospital. *Id.* at 8-10 and Attachment 4, Table 2.

G-57. The hospitals most appropriate to treat contaminated, injured individuals are those with both an IDNS license for radioactive materials with specialized training provided by RMC. Tr. 5368 (Ed). The second category consists of those hospitals licensed to handle radioactive materials and with a staff knowledgeable of radiation and nuclear materials and licensed by IDNS to deal with radioactive materials. Rockford has two well-equipped hospitals with very fine nuclear medicine departments, Rockford Memorial Hospital and Swedish American Hospital. Tr. 5368-69 (Ed). Applicant has no contract with the latter.

G-58. The Rockford Memorial Hospital does not have the capability to treat anyone receiving a life-threatening or dangerous dose. Tr. 5052 (Golden). That type of treatment would probably be carried out at either Northwestern Memorial Hospital in Chicago or at a number of universities throughout the country that have been involved with treatment of people who have received large doses of radiation. *Id.* The screening might be done at Rockford Memorial Hospital. Tr. 5074-75 (Golden).

G-59. The Applicant has contracted with Northwestern Memorial Hospital to provide treatment for its onsite Byron personnel who have been exposed to dangerous levels of radiation. *Id.* at 4.

G-60. Six ambulance services in or near the Byron plume EPZ (five within, one outside) can provide support in a radiological emergency including transportation of contaminated injured persons to one of the hospitals. These services have a total of nine ambulances and three rescue-squad vehicles. All will receive training in radiation protection and dosimetry equipment. Illinois Emergency Service and Disaster

Agency is pursuing mutual aid agreements with other ambulance services outside the EPZ with extensive transportation resources that could provide backup support. Smith, ff. Tr. 5170, at 3-5.

G-61. In view of the very strong showing that Applicant has made, or is making, excellent arrangements for emergency medical facilities, Intervenors' criticisms seem to be trivial and premature. Proposed Findings 61-79. For example, Intervenors imply that because Applicant's supervisor for emergency planning, Dr. John Golden, "does not possess a medical background," he is not competent to assess the adequacy of Rockford Memorial Hospital to treat contaminated injured persons. The fact is that Dr. Golden has a bachelors degree in physics, masters and doctors degrees in public health with a major in radiological and environmental health. He also has professional experience as a health physicist, including service with Sandia Corporation. Moreover he has the benefit of advice from Applicant's medical department headed by a medical doctor, Dr. Mehn who, as it happens, is also chairman of the Radiation Emergency Committee at Northwestern Memorial Hospital. These facts were well established at the hearing. *E.g.*, Golden, ff. Tr. 5035, at i, and Golden Ex. 6; Tr. 5071 (Golden). Intervenors' persistence in making the argument while ignoring Dr. Golden's qualifications and the support of Dr. Mehn's medical department, diminishes the Board's confidence in the Intervenors' proposed findings on this subissue.

G-62. Similarly, the state of readiness of the six ambulance services in or near the Byron EPZ as indicated by Intervenors' prehearing ambulance survey (Intervenors Ex. 14) has little relevance to their competence after the prospective training by the Illinois Emergency Services and Disaster Agency. In addition, Intervenors' arguments (Proposed Finding 73), and their ambulance survey, fail even to acknowledge that one of the ambulance services, the Byron Fire Protection District, which did not respond to the survey and which is not included in the analysis, is actually under contract with Applicant.

G-63. The Board finds that there is reasonable assurance of adequate arrangement for medical services for contaminated injured individuals as required by 10 C.F.R. 50.47(b)(12) as that section has been interpreted by the Commission in *San Onofre, supra.*

### *3. Paragraph 8, Local Protection*

G-64. The parties also litigated paragraph 8 of the emergency planning contention which alleged:

> In violation of 10 C.F.R. 50.47(b)(10), emergency plans are incapable of offering sufficient guidance for the choice of protective actions during an emergency since

Applicant and state planners have yet to adequately determine the local protection afforded (in dose reduction) by various protective measures including evacuation, sheltering, and radioactive prophylaxis.

G-65. The Applicant's case on paragraph 8 consisted of the prepared testimony of Dr. Golden, Mr. Smith, and Mr. Ed. The Staff's testimony consisted of the prepared testimony of Mr. Phillips and Mr. Wenger. The Intervenors' case on paragraph 8 consisted of the following stipulated exhibits: Affidavit of Paul Holmbeck (Exhibit 13, at 4-12), Affidavit of Gary Montel, Administrator of the Pine Crest Manor Home of the Aging (Exhibit 15, at 2), Affidavit of J. Michael Maloney, Superintendent of Schools for the Leaf River Community Unit No. 270 (Exhibit 16, at 10), Affidavit of Charles Lamb, Director of the Ogle County Education Cooperative (Exhibit 17, at 6), Affidavit of David Turner, Superintendent of Schools for the Mount Morris Community Unit 261 (Exhibit 18, at 3-12), and Affidavit of David Miller, Superintendent of Schools for the Meriddean Community Unit No. 223 (Exhibit 19, at 10).

G-66. Since Intervenors had a separate paragraph on evacuation time estimates, the principal feature of their paragraph 8 litigation was the allegation that Applicant and the State have not adequately determined the protection afforded by sheltering and radioprotective prophylaxis.

G-67. Section 50.47(b)(10) provides that emergency plans must demonstrate that: "a range of protective actions have been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place . . . ."

G-68. Federal guidance as it relates to sheltering may be found in NUREG-0654, at 64, where the evaluation criteria for protective responses includes:

m. The bases for the choice of recommended protective actions from the plume exposure pathway during emergency conditions. This shall include expected local protection afforded in residential units or other shelter for direct and inhalation exposure, as well as evacuation time estimates. [Footnote omitted.]

*See also id.,* Appendix 1, at 1-7, 1-16.

G-69. The meaning of "sheltering" as a radiological emergency protective response is not in dispute. Mr. Phillips, the Staff's emergency preparedness analyst, described it in the following terms:

Essentially, sheltering is a protective action consisting of doing the best you can with what you have. We are not talking about ensuring that everyone has a basement, or lives in a fallout shelter. What we are talking about is closing the doors and windows, going inside, turning off the ventilation system (or for most houses the furnace fan), and staying away from any outside openings if possible. Having a basement would be ideal, but it certainly isn't a requirement for licensing that all homes have basements and be made of brick. Also, this criterion does *not* mean that a house-to-house canvas or survey must be conducted to determine how many have basements, how many are made of brick, and how many are office buildings, etc. [Emphasis in original.]

Phillips, ff. Tr. 5509, at 10.

G-70. The last sentence in the quoted testimony was an apparent reference to Staff's perception of the Intervenors' contention as set out in Mr. Holmbeck's affidavit, Intervenors Ex. 13, at 11-12. However, in their proposed findings, Intervenors apparently now agree that a house-to-house survey is not necessary in that decisionmakers need not know the sheltering capacity of every structure in the EPZ. *E.g.*, Proposed Finding 90. Nevertheless, as we explain below, Intervenors persist in their view that Applicant has not adequately determined the sheltering protection available in an emergency.

G-71. Applicant's emergency plans and studies relating to protective measures are designed to conform to the state programs for radiological emergencies. Mr. Ed of the Illinois Department of Nuclear Safety (IDNS) described the state's approach to sheltering *vis-a-vis* evacuation:

[T]he goal of [Illinois Plan for Radiological Accidents] IPRA is to totally eliminate or maximally reduce the dose commitment accumulated by the general population during an accident involving radiation or radioactive materials. Evacuation is clearly favored as the most effective protective action since it reduces radiation exposure to zero if timely achieved. Sheltering is utilized as a protective action only when it is estimated to be more effective in dose reduction than evacuation, *i.e.*, only when timely evacuation is impractical or impossible. For such circumstances, DNS has developed a standard operating procedure which would guide DNS in choosing between evacuation and sheltering as recommended protective action. This procedure has been approved by FEMA and NRC. The procedure reduces the factors that must be considered in selecting the appropriate protective action to a set of complex mathematical formulate. This mathematical operation considers, among other things, the dose commitment reduction afforded by sheltering. The factors for dose commitment reduction afforded by sheltering are derived from the EPA report entitled "Protective Action Evaluation Part II, Evacuation and Sheltering as Protective Actions Against Nuclear Accidents Involving Gaseous Releases" (EPA 520/1-78-001B). *If the predominant type of structure is unknown or of a mixed type, the dose reduction factor used for sheltering is a conservative value assuming a single-story wood frame building, the least protective type of sheltering provided by a permanent structure.* The use of such a conservative value for the dose commitment reduction

afforded by sheltering is consistent with our policy which favors evacuation. The procedure compares the dose commitment reduction afforded by evacuation with the degree of dose commitment reduction that would be afforded by sheltering. As stated earlier the dose commitment reduction factor provided by timely evacuation is 100 percent. However, if evacuation cannot be completed before exposure to the plume, then the effectiveness of this action is decreased. The dose commitment reduction afforded by delayed evacuation is simply a ratio of the amount of time it would take to complete evacuation versus the duration of exposure. For example, if the release lasts four hours and evacuation requires two hours, and both commence simultaneously, the dose reduction afforded by evacuation is 50 percent.
[Emphasis added.]

Ed, ff. Tr. 5174, at 11-13.[86]

G-72. Intervenors, however, take issue with the State's (and Applicant's) assumption that the use of least protective type of sheltering provided in a permanent-type structure, *i.e.*, single-story wood frame building, is a conservative assumption.[87] Proposed Findings 87, 88. As they note, to bring about a premature evacuation under the so-called "conservative" assumption that the assumed average generic sheltering in the Byron case is less than that actually provided there, is not necessarily in the direction of safety, and in fact may be in the opposite direction. We agree.

G-73. Mr. Phillips for the NRC Staff explained that

Guidance for determining *this* value ... [for sheltering] is presented in three documents referenced on page 64 of NUREG-0654. Both the Applicant and the Illinois Department of Nuclear Safety have chosen to use EPA-520/1-78-001B, "Protective Action Evaluation Part II, Evacuation and Sheltering as Protective Actions Against Nuclear Accidents Involving Gaseous Releases." [One of the three documents.] The determination of the average shielding factor may be done by estimating the percentage of various building types and multiplying by the appropriate shielding factor to determine an average, or by using the guidance documents listed on page 64 of the NUREG. For example, Table 5 of SAND 77-1725, "Public Protection Strategies for Potential Nuclear Reactor Accidents: Sheltering Concepts With Existing Public and Private Structures," [another of the three documents] defines a weekly average shielding factor for both cloud and surface deposited radioactive

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[86] We learn from Mr. Phillips' testimony that the Applicant has incorporated into its Emergency Plan the *lower* limits of EPA Protective Action Guides (PAGs). The PAGs provide a range of 1-5 Rem whole-body and from 5-25 Rem thyroid dose savings at which evacuation should be conducted. Ff. Tr. 5509, at 9. Mr. Ed's testimony suggests that evacuation would be ordered to totally eliminate or maximally reduce any dose commitment. This apparent inconsistency is not explained, but since either standard is very conservative *with respect to dose savings*, it is beyond the Board's purview and the scope of this subcontention.

[87] However, Intervenors' citations to these specifically mentioned affidavits do not support the statement that many structures are neither one story nor wood. It is not necessary to have a specific record citation to support the statement, however. Mr. Ed's testimony recognizes that the assumption of single-story wood structures is not realistic.

material for seven geographical areas of the country, including the Midwest and Great Lakes area.

Phillips, ff. Tr. 5509, at 8.

G-74. Clearly the State has not estimated the percentage of various building types in the Byron EPZ. Mr. Ed's testimony, cited above, seemed to have used the document "EPA-520/1-78-001B" as support for his statement "[i]f the predominant type of structure is unknown or a mixed type, the dose reduction factor used for sheltering is a conservative value assuming a single-story wood frame building, the least protective type of sheltering provided by a permanent structure." The Board reviewed the EPA document and could find no basis for so-called "conservative" assumptions.[88] To the contrary, the EPA study considers the difference in the sheltering values provided by small shelters (SS) and large shelters (LS). *E.g.*, EPA Study at 30. Nowhere does the EPA study paint sheltering values with so broad a brush as does the Illinois Department of Nuclear Safety. We were also struck with the insensitivity of the State's generic sheltering value selection in that it did not even mention whether the assumed generic structure had a basement.

G-75. Accordingly the Board will require that the Applicant provide information to the emergency planning officials, particularly the Illinois Department of Nuclear Safety, which realistically reflects the average sheltering values of the structures in the Byron EPZ. This may be generic information. Given the largely rural and suburban nature of the Byron plume exposure EPZ, it is our view that the "EPA-520/1-78-001B" document's simple use of small shelter and large shelter values may be too general. We would prefer to *see* an estimation of the percentage of the various building types in the Byron plume exposure EPZ multiplied by the appropriate shielding factor in arriving at the average value. However, we do not foreclose any method approved by the NRC Staff which realistically estimates a generic average sheltering value for structures near Byron Station.

G-76. Intervenors also urge the Board to require specific identification of large buildings with higher sheltering values near Byron for the use of small communities and transients. Proposed Finding 94. This does not seem to be a practical thing to do given the State's policy of using a low threshold for evacuations. As Dr. Golden stated, it would

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[88] By Memorandum and Order dated September 20, 1983 (unpublished), the Board informed the parties that it would take official notice of the EPA document. Applicant responded on October 10 with Mr. Ed's explanation that the EPA report does not refer to or recommend the practice described as "conservative" by Mr. Ed in his testimony. Assuming the least amount of protection as a conservatism is a policy of the State of Illinois, not EPA.

make it more difficult to evacuate persons from the plume exposure EPZ if they have already been evacuated from their homes to a larger building. Tr. 5145. Also the sheltering delay time in relocating from, say, a house with relatively low sheltering values to a building with higher values could cause an increase in dose exceeding any dose savings from transferring to the larger building. Moreover, it is doubtful whether such buildings exist in significant numbers near Byron. *Id.* School buildings come to mind, but as Intervenors' witnesses state in their affidavits, these buildings are generally without basements and have large window areas and are generally poorly suited for sheltering, as are health care facilities. Intervenors Ex. 13-20.

*Prophylaxis*

G-77. In their proposed findings, Intervenors have changed the thrust of this subcontention from their initial allegation that Applicant and State planners *have failed to determine* the protection available from radioactive prophylaxis to an assertion that potassium iodide (KI) *should be provided* to the general population.

G-78. As Intervenors' own proposed findings indicate, the State has in fact made a determination of the protection available from KI. Proposed Finding 98. The State's determination to provide KI to selected groups is reasonable and is not, as claimed by Intervenors, discriminatory. All those in need will be provided. *Id.*; Ed, ff. Tr. 5147, at 13, and Attachment 1. Moreover we are without authority to direct the State to administer any medication to the population at large.

### *4. Paragraph 10, Volunteers*

G-79. The subcontention expressed in paragraph 10 asserted:

> The emergency planning relies too heavily upon volunteer personnel to effect an evacuation. The emergency plans fail to indicate the number of volunteer personnel who are necessary or available to perform the responsibilities assigned to them. Furthermore, the plans do not: (a) assess the availability of volunteers during hours in which many are employed outside the EPZ; (b) take into consideration inevitable personal conflicts in the responses of volunteers who have families in the EPZ; and (c) give consideration to the possibility that some volunteers who might perform well in nonradiological disasters might refuse to participate in a radiological disaster at the Byron Station.

G-80. The Applicant made a particularly strong evidentiary showing on the issue of the use of volunteers by presenting the testimony of two high officials of the Illinois Emergency Service and Disaster Agency,

E. Erie Jones, Director of the Agency, and David Smith, Chief of Field Services. Similarly Mr. Gordon Wenger of FEMA brought to the hearing the benefit of 25 years institutional experience during which FEMA and its predecessor agencies have observed the work of volunteers in crises of all kinds. Intervenors presented the helpful testimony of Mr. Thomas Bowes who has over 20 years experience as a volunteer in emergency situations and is a reserve officer for the Ogle County Sheriff.

G-81. All parties conceded that the emergency plans for the Byron Station depend heavily on volunteer services. *E.g.*, Jones, ff. Tr. 5444, at 4. The State and Federal witnesses testified generally that, if volunteers are adequately trained, they will function as well as paid personnel in emergencies. *Id.* at 6. They are well motivated and can be counted on. *Id.* They will be trained and properly included in the plans. *E.g.*, Wenger, ff. Tr. 5509, at 6.

G-82. Mr. Bowes testified for Intervenors that, as people become more familiar with how to react in specific instances and are trained, they have a tendency to react as they are trained. He testified that his initial fears about radiation were somewhat alleviated through training and information from Mr. Wenger. Tr. 5628-34. Mr. Bowes also testified that he would perform his duties as a volunteer unless he had a conflict as the administrator of a nursing home, but he would not man his station (a roadblock, for example) if he knew that a plume was heading directly toward him. Tr. 5636. The Board believes that Mr. Bowes presented a realistic picture of the response of a significant but uncounted portion of the volunteers depended upon to staff the emergency response organizations. He did not seem to be timid or reckless, or particularly well informed about radiological plumes. He, as others, may or may not be available to serve in a radiological emergency.

G-83. Perhaps in view of the testimony of the witnesses on this subject, Intervenors have realistically limited the scope of the subcontention in their proposed findings to a question of numbers. They believe there is a need to recruit additional personnel and to demonstrate that assigned volunteers would appear as needed. Proposed Findings at 34.

G-84. Mr. Jones addressed this issue squarely in his testimony. Emergency plans assume that there are a certain number of volunteers who won't show up. He has never had an experience where his agency has not had enough volunteers. Tr. 5458-59.

G-85. Intervenors' contention suggests, however, that there is a difference between nonradiological disasters and radiological disasters which reduce the number of volunteers available to respond in an emergency at Byron. The Board, however, accepts the testimony of the experienced and well-qualified witness for the State Disaster Agency, Mr. Jones, and the testimony of Mr. Wenger for FEMA. *E.g.*, Tr. 5452-54 (Jones); Wenger, ff. Tr. 5509, at 6; Tr. 5574-80. Moreover, one cannot even fairly infer from Mr. Bowes' testimony that he views radiological emergencies as inherently different from nonradiological emergencies.

G-86. In the Board's view, Intervenors' argument that radiological emergencies present a different type of emergency which will adversely affect the response of volunteers is predicated more upon a philosophical viewpoint than upon any evidence. We believe it can be assumed that timid people do not volunteer to be emergency workers in any disaster where their health or safety is perceived to be threatened. Some emergency workers may feel more threatened by radiation than by other dangers, but, on the other hand, some may feel more threatened or dysfunctional in the face of possible physical trauma as in fires, floods, explosions, and other nonradiological disasters. The answer lies in training and the identification of volunteers who prefer not to serve in radiological emergencies. These we assume will tend to identify themselves during planning and training.

G-87. In any event, to the extent that the Byron radiological emergency plan relies on too few volunteers, Applicant's Reply (at 15-16) to Intervenors' proposed finding satisfies the Board that the problem will be addressed. Applicant notes:

> The adequacy of the training program is the subject of a stipulated commitment between the parties. . . . (Commitment "D.") The training sessions for volunteers will provide state planners an opportunity to assess the willingness of individual volunteers to carry out their assigned responsibilities, during a radiological emergency. If it becomes apparent that certain volunteers will not cooperate in such an emergency, it is not reasonable to expect that the State planners will continue to rely on those volunteers who express a reluctance or refusal to cooperate. To the contrary, it is reasonable to expect that the State planners will adjust their plans accordingly. [Citation omitted.] Thus, the State's experience with volunteers and the training process provide reasonable assurance that response organizations will carry out their assigned responsibilities.

G-88. The Board finds that, contrary to the subcontention, the Byron emergency plan does not rely too heavily on volunteers to effect an evacuation.

### 5. *Paragraph 13, Communications with Emergency Response Organizations*

G-89. The final subcontention litigated by the parties is embodied in paragraph 13 which states:

In violation of 10 C.F.R. 50.47(b)(1), the emergency plans, specific tasks, and responsibilities have been formulated without sufficient communication between planning officials and primary and support response organizations so as to enable said organizations to fulfill their assigned roles.

Section 50.47(b)(1) provides in relevant part:

The onsite and, except as provided in paragraph (d) of this section, offsite emergency response plans for nuclear power reactors must meet the following standards:

(1) Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established . . . .

G-90. The essence of Intervenors' subcontention, as set out above, changed to almost the opposite concern by the time their proposed findings were submitted. As we read the subcontention, Intervenors were originally concerned that emergency planning had progressed very far without bringing the emergency response organization into the process, *i.e.*, poor communication. Now Intervenors recognize that the adequacy of actual (as compared to prospective) communications between emergency planners and response organizations is somewhat of a premature consideration because of the early stages of emergency planning relative to the licensing hearing. *See* Intervenors' Proposed Findings, "Discussion" at 35.

G-91. Even so there were two early tests of the communications link between the emergency planners and the local organizations with response responsibilities which, in the view of Intervenors at the time of the hearing, portended poor communications in the final plans.

G-92. Intervenors presented the testimony of three school superintendents, two nursing home administrators, and the director of the county program for handicapped students. All assert that, as of February 1983, communications had been unsatisfactory. Bowes, ff. Tr. 5622, at 2, 8; Intervenors Ex. 15-19; Montel at 2, 9-10; Lamb at 2-3, 6, 8; Turner at 3-4, 8, 9-10; Miller at 3-4, 8, 9-10; Maloney at 3-4, 7-11.

G-93. Several of the school superintendents had questions concerning the liability of emergency response organizations during an emergency. These questions have been brought to the attention of the State Emergency Services and Disaster Agency and that organization promises to address them. Intervenors Ex. 16, 18-19; Tr. 5214, 5220, 5354-55 (Smith). With respect to the liability question, at least, effective early communication has been established, as Intervenors acknowledge in their Proposed Findings 121-22, which we have adopted almost verbatim in this and the preceding finding. *See also* Tr. 5447-48 (Jones).

G-94. School superintendents were also concerned that they were given only about one week to review Revision 0 (*i.e.*, Revision zero) of Volume 6 (Byron-specific volume) of the Illinois Plan for Radiological Accidents (IPRA). Intervenors' Proposed Finding 124.

G-95. The Byron plan, IPRA Volume 6, will probably have four revisions. Tr. 5221 (Smith). During the very week that the Emergency Service and Disaster Agency officials testified, a meeting with school superintendents was scheduled to review a *draft* of Revision 1 of the Byron plan. Tr. 5209-10 (Smith). Thus it appeared that the school superintendents' concerns were in the process of being addressed, with an opportunity for still early contributions.

G-96. Mr. Wenger of FEMA testified that his agency will review the second revision of the Byron emergency plan, IPRA Vol. 6, upon its completion. He recognized that, at the time of the hearing, sufficient time had not yet been allowed for the Illinois State and local officials to complete the planning activities in accordance with their normal progression. Wenger, ff. Tr. 5509, at 7-8; Tr. 5511, 5534-36, 5604-08.

G-97. Although Intervenors presented only two specific instances of perceived inadequate communications, they stated a generalized worry that the operating license hearing was already in session with scant evidence of local involvement in emergency planning. They stated that all of the involved planners, Applicant, Intervenors, and especially the Board, were "at the mercy of awkward time tables in discussing the adequacy of communications at such an early stage in the planning." We noted similar concerns in the public limited appearance statements.

G-98. The stipulation among the parties to defer litigation of many emergency planning issues in exchange for Applicant's respective commitments and the Commission's action extending the Board's jurisdiction for that purpose provided substantial assurance to the public that emergency plans will materialize as promised.

G-99. Even more reassuring, however, was the testimony of Mr. Jones, the director of the Illinois Emergency Services and Disaster Agency, and Mr. Smith, the field director for IESDA, together with Mr. Ed of the Illinois Department of Nuclear Safety. It is clear from their testimony that very careful attention is being given to the Byron plan and we were convinced that excellent communications were, at the time of the hearing, being established with the response organizations. It is also clear from Mr. Wenger's testimony that FEMA is closely following the planning.

G-100. Nor are the communication plans simply left to chance or good intentions. IESDA has developed an Emergency Response Training Plan Matrix (Applicant Ex. 20), which is essentially a guide to all of the organizations with which IESDA already has had initial contacts, and will work with them more extensively in the near future to further develop the Byron plan. These organizations have emergency responsibilities under the Byron plan in the event of a radiological emergency. The Training Plan Matrix identifies the specific aspects of the Byron plan for which each group is responsible and their training requirements. Smith, ff. Tr. 5174, at 6; Applicant Ex. 20. IESDA is working with the organizations and developing the Byron Emergency Plan in accordance with a schedule depicted by a bar graph chart presented in evidence. Applicant Ex. 22; Tr. 5175-76, 5192-94 (Smith).

G-101. The Board finds that the emergency plans, specific tasks and responsibilities are being formulated with sufficient communication between planning officials and the emergency response organizations so as to allow those organizations to fulfill their assigned roles. Further, the Board finds reasonable assurance that the final plans will reflect adequate input from the local response organizations to ensure that they can fulfill their assigned roles.

### *6. Conclusions — Emergency Planning*

G-102. Subject to the condition imposed by the Board that Applicant's Evacuation Time Study be clarified, or amended if necessary, to reflect employment center shutdown times, we conclude that the Study has adequately addressed the relative significance of alternative assumptions, contrary to the claim of Emergency Planning paragraph 2(c).

G-103. Contrary to Emergency Planning paragraph 2(e), the Evacuation Time Study has adequately considered the impact of peak populations, and the behavioral aspects of peak populations as well as the general population. There is also reasonable assurance the behavioral aspects of schoolchildren and their parents will be timely and adequately considered.

G-104. Having required the Applicant to modify the Evacuation Time Study to reflect accurate traffic-time estimates under adverse weather conditions, the Board concludes that adequate considerations will be given to adverse weather conditions by the Study. Contrary to the allegation of Emergency Planning paragraph 2, the Study employs the site weather characteristics of the FSAR.

G-105. Contrary to Emergency Planning paragraph 3, emergency planning for the Byron Station EPZ does assure that there are adequate medical facilities to provide the equipment and trained personnel necessary to care for contaminated injured persons, that there are sufficient procedures for the screening, treatment, and isolation of persons sustaining radiological injuries, and that there are sufficient numbers of materials, supplies, equipment, and vehicles provided for the transportation of injured persons during a radiological disaster.

G-106. Contrary to Emergency Planning paragraph 8, and subject to the condition that Applicant provided to emergency planning officials information which realistically reflects the average generic sheltering values in the Byron plume EPZ, the emergency plans provide sufficient guidance for the choice of protective actions during an emergency.

G-107. Local emergency planning reliance on volunteer personnel is justified and proper, contrary to the claim in Intervenors' Emergency Planning paragraph 10.

G-108. Contrary to the claim in Intervenors' Emergency Planning paragraph 13, the Byron emergency plans have been and are being formulated with a sufficient degree of communication among the planning officials and primary and support response organizations so as to enable such organizations to fulfill their assigned roles.

G-109. Accordingly, subject to the conditions mentioned above, the Board concludes that the Applicant has satisfied the concerns expressed in the litigated Emergency Planning paragraphs and has prevailed on those issues.

G-110. The Board does not believe that the conditions are burdensome or difficult to satisfy. Most, perhaps all, of the information required is already on hand. Therefore, requiring that the conditions be satisfied before Byron exceeds 5 percent of power will provide plenty of time for Applicant to comply without risk to the public safety.

## III. CONCLUSION AND ORDER

The Board withholds authorization for an operating license for the Byron Nuclear Station because of inadequacies in Applicant's quality assurance program. The application is, therefore, denied. It is not within our jurisdiction to foreclose further proceedings on the application, and we recognize that an operating license for Byron may subsequently be granted. Therefore, the Board has decided all other issues before us, and, in the case of emergency planning, we have imposed conditions on any operation of the Byron Station. Similarly, we have specifically noted various commitments by Applicant, particularly in connection with steam generators, which commitments we regard as binding in any operation of the facility.

The Board's decision to withhold operating authority is, of course, a very important result. Therefore, the rationale, scope and significance of our decision should be precisely understood.

This is our final decision in this proceeding. Our jurisdiction passes in accordance with 10 C.F.R. 2.717(a), 2.760, 2.762, 2.771, and 2.785. *See Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-726, 17 NRC 755 (1983).

In *Cincinnati Gas & Electric Co.* (Wm. H. Zimmer Nuclear Power Station, Unit 1), ALAB-727, 17 NRC 760, 776 (1983), the Appeal Board recognized the Licensing Board's implicit intent to retain continuing jurisdiction after the Licensing Board's ruling in a final initial decision that further proceedings are necessary to resolve emergency planning issues. *Zimmer, supra,* LBP-82-48, 15 NRC 1549 (1982). This is not our intent with respect to jurisdiction over quality assurance issues in this proceeding. Except for the unusual provision for continuing jurisdiction over emergency planning issues, jurisdiction over this proceeding will pass from this Board in accordance with the regulations and the *Limerick* case cited above.

Recognizing that the matter may not forever be closed, we explain further the significance of our order. The Board considered the alternative of informing the parties now of the substance of our views on the quality assurance issues, retaining jurisdiction over them, and providing for further proceedings before us when the various inspections, investigations and remedial actions become ripe for consideration. Perhaps a partial initial decision on all other issues could have been rendered.

We have determined, instead, that the remedy most responsive to the circumstances of this case, and the remedy least harsh to the Applicant yet still appropriate, is to decide the issue now. This, we say, is the least harsh appropriate remedy, as compared to the traditional practice of reserving jurisdiction, because it permits the parties to test immediately on appeal the quality of our decision. To reserve jurisdiction and to postpone final decision, in face of the impending completion of construction at Byron, would impose unilaterally upon the parties, particularly the Applicant, our own view of the facts, law and appropriate remedy. Unless Applicant could mount a difficult interlocutory appeal from such a determination (to postpone our decision), it would have been denied due process.

In describing the reach of our order, we have avoided describing it as *res judicata* or collateral estoppel with respect to the quality assurance issues because neither concept, as ordinarily understood, captures our intent. Neither concept neatly fits the unusual situation to be found in the continuum of a licensing proceeding with many aspects. We do not foreclose future proceedings on the quality assurance issue and have no jurisdiction to do so. Recognizing that each party has proposed a final decision to the Board, albeit in differing directions, we have simply decided the issue on the record before us.

We come now to the emergency planning phase of the proceeding. By its order of August 22, 1983 (unpublished), the Commission authorized the Board to conduct, subsequent to an initial decision and subsequent to the issuance of a full-power license, any proceeding that may be provided for in the parties' emergency planning settlement, as set out in the Board's certification to the Commission dated June 17, 1983 (unpublished). Accordingly, we retain jurisdiction sufficient to discharge those responsibilities.

With respect to those emergency planning issues decided in this decision, we direct that any operation of the Byron Station above 5 percent of power be subject to the following conditions:

1. Applicant's Evacuation Time Study must be clarified, and amended if necessary, to reflect employment center shutdown times in accordance with the Board's Finding, Paragraph G-20, *supra*.
2. Applicant's Evacuation Time Study must be modified to reflect realistic time estimates under adverse weather conditions pursuant to the Board's Finding, Paragraph G-43, *supra*.
3. The Applicant must provide information to emergency planning officials, particularly the Illinois Department of Nuclear Safety, which realistically reflects the average generic sheltering values of the structures in the Byron emergency planning zone pursuant to the Board's Finding, Paragraph G-75, *supra*.

IT IS THEREFORE THE ORDER OF THE BOARD, that the Director of Nuclear Reactor Regulation may not issue an operating license for the Byron Nuclear Power Station, Units 1 and 2. The application is therefore denied. If, however, the operating license for the Byron Station is otherwise granted, any operation shall be in accordance with the conditions imposed in this order and in accordance with this Initial Decision.

## IV. FINALITY AND APPEALABILITY

Pursuant to 10 C.F.R. 2.760 of the Commission's Rules of Practice, this Initial Decision will constitute the final decision of the Commission thirty days from the date of its issuance, unless an appeal is taken in accordance with 10 C.F.R. 2.762 or the Commission directs otherwise. *See also* 10 C.F.R. 2.785 and 2.786.

Any party may take an appeal from this decision by filing a Notice of Appeal within ten days after service of this Initial Decision. Each appellant must file a brief supporting its position on appeal within thirty days after filing its Notice of Appeal (forty days if the Staff is the appellant). Within thirty days after the period has expired for the filing and service of the briefs of all appellants (forty days in the case of the Staff), a party who is not an appellant may file a brief in support of or in opposition to the appeal of any other party. A responding party shall file a single, responsive brief *only,* regardless of the number of appellants' briefs filed. (*See* 10 C.F.R. 2.762 as amended December 19, 1983, 48 Fed. Reg. 52,283 (1983).)

THE ATOMIC SAFETY AND
LICENSING BOARD

Dixon Callihan
ADMINISTRATIVE JUDGE

Richard F. Cole
ADMINISTRATIVE JUDGE

Ivan W. Smith, Chairman
ADMINISTRATIVE LAW JUDGE

Bethesda, Maryland
January 13, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Peter B. Bloch**, Chairman
**Dr. Jerry R. Kline**
**Mr. Glenn O. Bright**

**In the Matter of** **Docket Nos. 50-440-OL**
**50-441-OL**

**CLEVELAND ELECTRIC ILLUMINATING COMPANY, *et al.***
**(Perry Nuclear Power Plant, Units 1 & 2)** **January 20, 1984**

The Licensing Board denies intervenor's motion to reopen the record.

## RULES OF PRACTICE: MOTION TO REOPEN

The purpose of reopening the record is for a party to submit or to develop evidence. A motion not made for that purpose does not provide grounds for reopening the record.

## RULES OF PRACTICE: MOTION FOR BOARD TO INVESTIGATE

A licensing board will not conduct its own investigation of quality assurance allegations without proof that Staff offices are unable to conduct such an investigation adequately. Boards are primarily responsible for conducting hearings and should not readily undertake investigative functions.

RULES OF PRACTICE: MOTION TO REOPEN

Newspaper allegations of quality assurance deficiencies, unaccompanied by evidence, ordinarily are not sufficient grounds for reopening an evidentiary record. Such articles do not demonstrate the existence of a "significant safety issue" or a "breakdown of the quality assurance program."

# MEMORANDUM AND ORDER

## (Motion to Reopen: Newspaper Allegations of Q/A Deficiencies)

Ohio Citizens for Responsible Energy (OCRE) has moved to reopen our record on quality assurance, based on newspaper reports that three former L.K. Comstock inspectors have made allegations of deficiencies in their former employer's quality assurance program.[1] The Staff of the Nuclear Regulatory Commission (Staff) and Cleveland Electric Illuminating Company, *et al.* (Applicants) oppose this motion,[2] as amended. Sunflower Alliance Inc., *et al.* (Sunflower) supports the motion.[3] The parties disagree about whether the motion meets the three criteria governing motions to reopen the record.[4]

We depart from the analysis of the parties because we do not consider OCRE's motion to be a true motion to reopen the record. The purpose of reopening the record is for a party to introduce evidence or to conduct discovery leading to the introduction of evidence. That is not what OCRE seeks. In its Motion to Reopen, OCRE states that the inspectors

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[1] Motion to Reopen the Record on Comstock Issues, November 25, 1983 (Motion to Reopen); Susan Hiatt's Letter of November 30, 1983 (First Amendment); and Amendment to Motion to Reopen the Record on Comstock Issues, December 8, 1983 (Second Amendment).

[2] NRC Staff Response in Opposition to OCRE Motion to Reopen the Record, December 22, 1983 (Staff Response); Applicants' Answer to OCRE Motion to Reopen the Record on Comstock Issues, December 19, 1983 (Applicants' First Response); and Applicants' Answer to OCRE Amendment to Motion to Reopen the Record on Comstock Issues, January 6, 1984 (Applicants' Second Response).

[3] Sunflower's Memorandum in Support of "Motion to Reopen the Record on Comstock Issues," December 8, 1983 (Sunflower's Support).

[4] *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-756, 18 NRC 1340 (1983) (need to demonstrate the safety significance of allegations regarding reopening of the record); *Kansas Gas and Electric Co.* (Wolf Creek Generating Station, Unit No. 1), ALAB-462, 7 NRC 320, 338 (1978) (proponent of a motion to reopen has a "heavy burden"); *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-81-5, 13 NRC 361, 363 (1981) (there must be more than bare allegations); *Union Electric Co.* (Callaway Plant, Unit 1), ALAB-740, 18 NRC 343, 346 (1983); *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-138, 6 AEC 520, 523 (1973) (the motion must be timely and must raise significant issues); *Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear-1), ALAB-227, 8 AEC 416, 418 (1974) (a different result would have been reached had the material been considered) (Sunflower also alleges that this case indicates that reopening may be granted with respect to a hotly contested issue).

whose allegations OCRE relies on "are reluctant to communicate with intervenors,"[5] and asks that

> the Licensing Board utilize the NRC Office of Investigations (headquarters personnel) as its investigatory arm to question the two inspectors (and others), secure affidavits from these persons, and in general conduct . . . investigations. . . .[6]

OCRE also states that

> unprofessional conduct of NRC (apparently) Region III Inspection and Enforcement personnel has been alluded to (which discouraged Mr. Mathis from contacting the NRC with his concerns)[7]

and asks that

> the Board conduct an *in camera* exploratory hearing at which Perry workers can, without fear of publicity, reveal their concerns, under oath, to the Board.[8]

Since OCRE does not seek to conduct discovery or to introduce evidence of its own, we do not consider the reopening of the record to be appropriate. There is no need for us to reopen the record to provide assurance that appropriate investigative personnel will investigate the quality assurance allegations, which were published by Cleveland newspapers, are under investigation by the NRC Staff[9] and have been called to the attention of the Office of Investigations.[10]

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[5] Motion to Reopen at 3.

[6] *Id.* at 3-4.

[7] *Id.* at 3.

[8] Second Amendment at 1-2. OCRE cites an unspecified action of the Catawba Licensing Board as precedent and requests that the Board make sure that there be adequate public and private notice of the *in camera* proceeding. *Id.* at 2-3.

[9] Staff Response, Affidavit of James E. Konklin and Cordell C. Williams at 2.

[10] On January 10, 1984, the Board received a telephone call from Mr. Balazs, telling it to expect to receive a letter stating that affidavits filed by the Applicants were incorrect. He also stated that he was pursuing a complaint before the Department of Labor concerning his firing and that he had a lawyer representing him. The Chairman told him that he had not yet received his letter.

On January 11, 1984, the Board received a letter from Steve E. Balazs, dated January 4, 1984, and subsequently served on the parties by the Secretary of the Commission. That letter alleged that facts in Applicants' affidavits were incorrect but it contained no particulars. Consequently, the Chairman called Mr. Balazs' attorney, Mr. Marvin Dworken, and advised him that the information could be: (1) communicated to either of the intervenors in this case, who could present it to the Board, (2) communicated to the Board if Mr. Balazs succeeded in becoming a party to the case, or (3) communicated to the Office of Investigations (OI).

Since Mr. Balazs' attorney, Mr. Dworken, seemed to prefer to call OI, the Chairman telephoned the Director of the Office of Investigations, Mr. Ben B. Hayes, to tell him to expect a call and to inform him of the nature of Mr. Balazs' concerns. Since Mr. Hayes was nearby, he visited with the Chairman on the afternoon of January 11, 1984, and received a copy of Mr. Balazs' letter. Mr. Hayes did not decide whether his office would undertake the investigation itself or refer it to Region III. Apparently, the Office of Investigations is suffering from an acute shortage of personnel.

OCRE's real intention is to request the Board to conduct an investigation of its own, something that the Board might do under unusual circumstances, but that does not appear to be appropriate under the facts presented. Before we would undertake our own investigation, supplanting a Staff function and eroding the separation between fact-finder and prosecutor, we need to be persuaded of the necessity of taking such an action.[11] However, no such showing has been made.

The only indication in our record of investigative inadequacy is the following quotation from a newspaper article:[12]

> [Gene F.] Mathis said he never raised these issues [about the inadequacy of quality assurance at L.K. Comstock] to NRC because he overheard one of its inspectors make fun of Comstock inspectors' complaints to Comstock managers.

Even if true, we find this second-hand, rudimentary and nonparticularized account to be an inadequate reason to lose faith in the integrity of Staff's investigative capacity. Furthermore, this allegation has been given independent scrutiny by the Staff,[13] lending additional credibility to Staff integrity. We therefore conclude that there is no reason for us to undertake our own investigation at this time.

In reaching this conclusion, and deciding to deny OCRE's motion, we do not in any way demean the potential importance of the information contained in the newspaper articles, if it should be true. If quality assurance inspectors have been intimidated, that would have serious implications for plant quality[14] and would cast in doubt the basis for our partial

[11] *See South Carolina Electric and Gas Co.* (Virgil C. Summer Nuclear Station, Unit 1), ALAB-633, 14 NRC 1140, 1163 (1981), *aff'd*, ALAB-710, 17 NRC 25 (1983) (dealing with the somewhat different question of when a Board should call its own expert witness); *Union Electric Co.* (Callaway Plant, Unit 1), ALAB-750, 18 NRC 1205 (1983) (addressing the division of responsibility between the Commission's adjudicatory boards and its Staff).

[12] James Lawless, *Fired Worker Worried About Work at Perry*, The Plain Dealer [Cleveland], Nov. 23, 1983, at 5-B[E].

[13] Staff Response, Affidavit at 3.

[14] One aspect of *The Plain Dealer* article of November 19, 1983, troubled us enough to ask for a clarification. This aspect consisted of two quotations from Max L. Gildner, the NRC's Senior Resident Inspector (Construction) at Perry. Mr. Gildner was quoted as saying, "you've got a company [L.K. Comstock] that treats people like machines, like a tool"; and "[t]hey are not real personnel conscious."

In response to our question, the Staff filed Mr. Gildner's affidavit on January 10, 1984. In that affidavit, dated January 5, 1984, Mr. Gildner confirmed that *The Plain Dealer* story accurately recorded his words. He explained that his talks with large numbers of QA personnel at the site persuaded him that there was "a lack of harmony and lack of effective management as perceived by L.K. Comstock inspectors" (Affidavit at 2) but that, for reasons detailed in the affidavit, this did not constitute either intimidation or harassment. *Id.*

We conclude from Mr. Gildner's affidavit that he has presented a forthright statement of his own views, setting forth criticism tempered by his evaluation of all the facts. This affidavit does not, by itself, call into question the Applicant's QA program, and it does not lend itself to serious adverse inferences unless there were additional evidence concerning specific problems of intimidation. We note also that Mr. Gildner's affidavit lends credence to the accuracy of the newspaper article because it corroborated facts in the article that were within Mr. Gildner's knowledge.

initial decision, which accepted the integrity of Applicants' system for coping with nonconformances.[15] No quality assurance system is any better than the individuals who record the data on which it is based. A failure to record deficiencies would raise serious questions indeed.

However, even were this a genuine motion to reopen the record we would deny it because the bare, uninvestigated and unsworn allegations[16] in the filed newspaper articles do not demonstrate that there is a "significant safety issue" or a "breakdown of the quality assurance program." As we have noted, "[t]he construction of Perry is a massive task."[17] Furthermore, we accept Applicants' assertion that "[c]ommunication problems and disagreements among workers and disciplines in an organization of the size and complexity of the Perry organization are not unexpected."[18] Were licensing boards to consider every allegation by a worker to be grounds for initiating an investigation, it is likely that Boards would become the investigative arm of the agency, a transmogrification that would adversely affect the ability of Boards to attend to the task of deciding important safety and environmental issues.

Nor do we think it beneficial for Boards to supplant investigators. Try as we may, the ability of a Board to assure confidentiality to witnesses is limited because of the necessity of permitting lawyers and representatives of parties to be present during our deliberations. Nor do we have the time to track down leads, examine relevant documents for clues and locate missing witnesses. These are difficult, time-consuming tasks best left to professional investigators permitted to operate within confidentiality constraints. Our refusal to undertake these tasks ourselves does not in any way deprecate their importance. The public and this Board rely on the NRC's investigators to conduct thorough investigations that will assist the Board and the public to understand the importance of quality assurance allegations.

Our dismissal of this motion does not prejudice either intervenor's right to move to reopen this contention in the future, providing that ade-

[15] Partial Initial Decision (Quality Assurance Contention), LBP-83-77, 18 NRC 1365 (1983).

[16] On January 12, 1984, OCRE's representative informed the Board that Applicants had refused to supply it with an address or telephone number for Mr. Ward, one of the allegers quoted in the newspapers. Since Mr. Ward's deposition was included in Applicants' Second Answer, the refusal to permit OCRE access to him could be relevant to the weight we would give to the affidavit. Consequently, we informed Applicants' lawyer on January 13, 1984, that we would disregard the affidavit for purposes of this decision unless we found it to be essential — in which case, the parties would have an opportunity to argue the matter before us.

Since our decision does not rely on the truth of the affidavit, the request for information about Mr. Ward is moot. We assume that NRC investigators would have no difficulty obtaining Applicants' help in contacting Mr. Ward, should that be necessary.

[17] Partial Initial Decision, 18 NRC at 1367.

[18] Applicants' Answer, Riley Affidavit at 13.

quate evidence accompanies the motion. OCRE is cautioned to await the assembly of sufficient evidence before making such motion, however. A party is responsible for satisfying itself that motions are meritorious, in light of applicable law and precedent (including previous Board decisions), before the motions are filed.

## ORDER

For all the foregoing reasons and based on consideration of the entire record in this matter, it is, this 20th day of January 1984,

ORDERED:

Ohio Citizens for Responsible Energy's November 25, 1983, Motion to Reopen the Record on Comstock Issues, as amended, is denied.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Jerry R. Kline (by PBB)
ADMINISTRATIVE JUDGE

Glenn O. Bright
ADMINISTRATIVE JUDGE

Bethesda, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Marshall E. Miller, Chairman**
**Dr. Cadet H. Hand, Jr.**
**Gustave A. Linenberger, Jr.**

**In the Matter of** **Docket No. 50-537-CP**
**(ASLBP No. 75-291-12)**

**UNITED STATES DEPARTMENT OF ENERGY**
**PROJECT MANAGEMENT CORPORATION**
**TENNESSEE VALLEY AUTHORITY**
**(Clinch River Breeder Reactor Plant)** **January 20, 1984**

In a *Memorandum of Findings* the Licensing Board concludes that:

(1) the suitability of the proposed site for the Clinch River Breeder Reactor Plant (CRBRP) for a reactor of the general size and type proposed has been reaffirmed;

(2) from the evidence of record, the CRBRP can be constructed and operated in a manner that would have satisfied the NRC's mandate that the CRBRP achieve a level of safety comparable with that of light water reactor plants. Further, core disruptive accidents need not be included within the spectrum of design basis accidents for the CRBRP;

(3) a comprehensive and detailed quality assurance program was in place and functioning (prior to the termination of the CRBRP program) in accordance with the requirements of Appendix B to 10 C.F.R. Part 50; and

(4) environmental and emergency planning matters were appropriately addressed.

## APPEARANCES

For the Applicants:

**George L. Edgar, Esq.; Thomas A. Schmutz, Esq.** and **Frank K. Peterson, Esq.** for Project Management Corporation, Newman & Holtzinger, P.C., Washington, D.C.

**Leon Silverstrom, Esq.** and **William D. Luck, Esq.** for U.S. Department of Energy, Washington, D.C.

**Herbert S. Sanger, Jr., Esq.; Lewis E. Wallace, Esq.; W. Walter LaRoche, Esq.; James F. Burger, Esq.** and **Edward J. Vigluicci, Esq.** for Tennessee Valley Authority, Knoxville, Tennessee

*For the Intervenors:

**Ellyn R. Weiss, Esq.; Barbara A. Finamore, Esq.; Eldon V.C. Greenberg, Esq.; Dean R. Tousley, Esq.** and **S. Jacob Scherr, Esq.** for Natural Resources Defense Council and the Sierra Club, Washington, D.C.

For the Nuclear Regulatory Commission Staff:

**Sherwin E. Turk, Esq.; Geary S. Mizuno, Esq.;** and **Stuart A. Treby, Esq.,** Washington, D.C.

*For the State of Tennessee:

**Michael E. Terry, Esq.** and **Michael D. Pearigen, Esq.** for the State of Tennessee, Nashville, Tennessee

*For the City of Oak Ridge:

**William E. Lantrip, Esq.** for the City of Oak Ridge, Oak Ridge, Tennessee

*Did not participate in the Construction Permit Hearing.

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# MEMORANDUM OF FINDINGS
**(Construction Permit Phase)**

## Introduction and Summary

Just as the proposed Clinch River Breeder Reactor Plant was the first of a kind technologically, so is this Memorandum of Findings somewhat unprecedented procedurally. The taking of evidence in the Construction Permit phase of this proceeding was concluded August 8-11, 1983. While the Licensing Board was considering the proposed findings submitted by the parties and was drafting its Initial Decision, Congress terminated the appropriation of funds necessary to construct the project.

The Senate voted on October 26, 1983, to table its Appropriations Committee amendment containing a multi-year appropriation for the CRBRP. The effect of this action was to provide no Fiscal Year 1984 funds necessary to continue construction of the plant.[1] The parties to the project then concluded that "there appears no substantial likelihood that such funds will be appropriated."[2] As a result, as described more fully *infra*, the Clinch River project has been terminated and the plant will not be built.

It is undisputed that the Clinch River project cannot now satisfy the Commission's regulations governing requirements for the issuance of a Construction Permit, and that it is not reasonably likely that the project will satisfy all of its programmatic objectives.[3] Accordingly, the Applicants have stated that they will not seek a Construction Permit and they

[1] 129 Cong. Rec. S14,611-44 (Oct. 26, 1983). Congress completed action on the Fiscal Year 1984 Supplemental Appropriations Bill on November 18, 1983 (129 Cong. Rec. H10,529 (Nov. 18, 1983). *See also* 129 Cong. Rec. H9875 (Nov. 15, 1983) and 129 Cong. Rec. S16,588 (Nov. 17, 1983).

[2] Agreement Terminating the Project, Attachment C to November 23, 1983 Motion of NRDC to Intervene, confirmed by Notification Concerning Project Termination, filed by the Applicants December 27, 1983.

[3] Applicants' Response to Motion of NRDC to Intervene, dated December 5, 1983, at 2 and 7-8. *See also* NRC Staff's Response to Motion of NRDC to Intervene, dated December 8, 1983, at 3 and 5.

would accept a condition that nothing in any decision issued by the Licensing Board should authorize the Director of Nuclear Reactor Regulation to issue such a permit. However, both the Applicants and the Staff have requested the Licensing Board to issue a Partial Initial Decision concerning the matters litigated during the Construction Permit phase of the proceeding.[4]

The reasons urged by these parties for the issuance of a Partial Initial Decision include (a) the substantial time, effort and resources which have been expended in developing a full record; (b) the public interest in resolving technical issues fundamental to liquid metal fast breeder reactor (LMFBR) design, and in future advanced reactor development;[5] and (c) the informational benefits which have been recognized by the Commission as flowing to the LMFBR base program.[6]

The Licensing Board takes note of the impressive amount of time, effort and allocation of expert personnel and other resources that have been devoted by all parties since 1975 to the issues involved in this proceeding, and in the development of the detailed evidence produced at our hearings. In the public interest we also recognize the substantial amounts of money expended through the years in this complex proceeding, which costs are ultimately borne by taxpayers and ratepayers. It would be a disservice to the public not to make a comprehensive critique and analysis of the extensive record developed in order to evaluate the numerous issues involved. To that end, the Board has prepared this Memorandum of Findings in order to memorialize its assessment of the issues.

The Applicants and the Staff have suggested the issuance of a Partial Initial Decision covering the matters litigated in the Construction Permit phase of the proceeding. A former intervening party[7] has insisted that no decision should be issued, and that the proceeding should be terminated immediately on ground of mootness.[8]

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[4] *Id.*

[5] *Id.*

[6] *United States Department of Energy* (Clinch River Breeder Reactor Plant), CLI-82-23, 16 NRC 412, 431 (1982).

[7] The Natural Resources Defense Council, Inc. (NRDC) was one of the original intervenors in this proceeding, and it participated fully in the LWA-1 phase of the hearings (LBP-83-8, 17 NRC 158 (1983)). However, NRDC voluntarily withdrew all of its contentions in June 1983, and it was dismissed as a party June 29, 1983 (Tr. 7732-33). NRDC's objections to the issuance of a Partial Initial Decision were set forth at length in its Motion of Natural Resources Defense Council, Inc. to Intervene (filed November 23, 1983), and in its Reply . . . to Applicants' and Staff's Response to Motion to Intervene (filed December 12, 1983 pursuant to leave granted by the Board). Although NRDC has not been permitted to float in and out of the proceeding at will, nevertheless its detailed objections contained in the above filings have been carefully considered by the Board.

[8] *Id.*

The Board considers it advisable to avoid unnecessary technicalities in completing its analysis of a voluminous record under the circumstances prevailing here. It is true that Commission case law establishes that licensing boards are not proscribed from issuing "advisory opinions" by the "case or controversy" clause contained in the U.S. Constitution, although they should issue such decisions only in the presence of compelling circumstances. *See, e.g., Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 54 (1978). However, there is no necessity to belabor the concept of "compelling circumstances" to justify an advisory opinion, because we are not issuing an advisory opinion. Neither are we issuing a Construction Permit or any other kind of license. It is sufficient to issue only a memorandum tailored to the unusual posture of this proceeding, for whatever assistance it may provide to the NRC now or in the future.

Since the issuance of a Construction Permit is no longer appropriate, it is unnecessary for this Memorandum of Findings to address many topics prescribed by Title 10 of the Code of Federal Regulations for inclusion in formal Construction Permit initial decisions. Rather, we have focussed upon those basic ingredients of a first-of-a-kind plant concept that would inherently determine whether that concept has been realistically translated into a responsible project effort, within the NRC's licensing framework. This approach dictated the format of the Opinion Section prior to the project's demise and has been adhered to since. Thus, the topical arrangement attempts to lead the reader through an orderly development of the subject areas necessary to critique the project effort while adhering to the adjudicatory requirement of an evidentiary basis.

The Applicants in this proceeding are the United States Department of Energy (DOE), Project Management Corporation (PMC), and the Tennessee Valley Authority (TVA). The Clinch River Breeder Reactor Plant (CRBRP) was intended to be a Liquid Metal Fast Breeder Reactor (LMFBR) demonstration plant with a rated output of 350 megawatts of net electrical power, proposed to be located on the Clinch River in Oak Ridge, Tennessee.[9]

On February 28, 1983, this Licensing Board issued a Partial Initial Decision addressing those portions of the application for a Construction Permit which are necessary for Limited Work Authorization (LWA) findings under 10 C.F.R. § 50.10(e)(2), namely, findings on all pertinent radiological site suitability and environmental issues (LBP-83-8, 17

[9] Appl. Ex. 86; Staff Ex. 26, at 1-4, 1-12. Citations to the record in this proceeding will be in the following form: (a) Applicants' Exhibits — Appl. Ex., Staff's Exhibits — Staff Ex.; (b) Citations to prefiled direct testimony will include both the exhibit number and page, and the transcript (Tr.) page. Citations to numbered paragraphs of Findings of Fact will be to Finding No.

NRC 158). For the reasons discussed above, that Partial Initial Decision and a pending appeal therefrom were vacated on the ground of mootness by the Appeal Board on December 15, 1983.[10]

The CRBRP was first authorized by Congress in 1970 as a cooperative effort between industry and government to design, construct, and operate the Nation's first demonstration-scale fast breeder reactor (Pub. L. No. 91-273, Section 106). In early 1972, the Atomic Energy Commission (AEC) accepted a joint proposal by the Commonwealth Edison Company of Chicago and the TVA to undertake the design, construction and operation of the demonstration plant as part of the TVA electric system.[11] Under that proposal, PMC, a nonprofit corporation organized and existing under the laws of the District of Columbia, had the overall lead management responsibility for the CRBRP. TVA would operate the plant, and the AEC had lead technical responsibility for the nuclear reactor systems.[12] Over 750 electric systems in the United States pledged more than $250 million in financial payments which were applied to the project by PMC.[13]

On October 11, 1974, PMC and TVA jointly filed an application with the AEC for a Construction Permit and Operating License for the CRBRP pursuant to Section 104(b) of the Atomic Energy Act of 1954, *as amended* (42 U.S.C. § 2011 *et seq.*). After the Energy Reorganization Act of 1974 (42 U.S.C. § 5801 *et seq.*) transferred the developmental and regulatory functions of the AEC to the Energy Research and Development Administration (ERDA) and the NRC, respectively, the NRC assigned the application to its docket for review on April 11, 1975.

On June 18, 1975, notice of receipt of the application and proceedings before the Atomic Safety and Licensing Board was duly published.[14] A timely joint petition for leave to intervene was filed by the Natural Resources Defense Council, Inc., (NRDC), the Sierra Club, and the

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[10] ALAB-755, 18 NRC 1337 (1983). The Appeal Board terminated the appellate proceeding and vacated the partial initial decision on the motion of the Intervenors. However, the Appeal Board further stated that "the issue of revocation of the LWA is better left to the Licensing Board, which still retains jurisdiction over the application for a construction permit." 18 NRC 1339.

[11] Pub. L. No. 92-84. *See* Joint Report of the House Comm. on Science and Technology and the Joint Committee on Atomic Energy (JCAE), H.R. Rep. No. 294, 94th Cong., 1st Sess., 32-35 (1975) (Joint Report); JCAE Authorization Report, S. Rep. No. 104, 94th Cong., 1st Sess., 17-20 (1975) (JCAE Report).

[12] *See* Report on Hearings before the Joint Committee on Atomic Energy on the Basis for the Proposed Arrangement for the LMFBR Demonstration Plant, 92d Cong., 2d Sess. (Sept. 7, 8, and 12, 1972) (JCAE Hearings) at IV-V. *See also* Report on Hearings before the JCAE to Consider Proposed Changes in the Basis for the Cooperative Arrangement for Design, Construction, and Operation of the LMFBR Demonstration Plant, 93d Cong., 1st Sess. (Feb. 28 and May 4, 1973).

[13] Appl. Ex. 86. At the time of its termination on October 26, 1983, more than $1.5 billion had been spent on the facility.

[14] 40 Fed. Reg. 25,708 (1975).

East Tennessee Energy Group (Intervenors), and on October 9, 1975, the intervention petition was granted by this Board. After the East Tennessee Energy Group (ETEG) had become defunct, the other Intervenors requested the withdrawal of ETEG as a party on February 8, 1982. The Board granted the request, leaving the Natural Resources Defense Council, Inc. and the Sierra Club as joint intervenors in the LWA proceedings.

The State of Tennessee Attorney General filed a timely petition for leave to intervene and was admitted as a party on October 9, 1975. On March 29, 1982, the State of Tennessee filed a Motion to Withdraw as a party under 10 C.F.R. § 2.714, but requested leave to continue participating as an "interested state" under 10 C.F.R. § 2.715. The motion was granted.

The City of Oak Ridge petitioned for leave to intervene on July 17, 1975. It amended that petition on January 22, 1976, and was admitted as a party on March 4, 1976. On August 20, 1982, the City of Oak Ridge requested leave to withdraw as a party to the proceeding but to continue participating as an "interested municipality" under 10 C.F.R. § 2.715(c). The Board granted that motion.[15]

On May 6, 1976, pursuant to authorization contained in the 1976 amendments to Pub. L. No. 91-273, *as amended,* the application was amended to include the Energy Research and Development Administration (ERDA) as a co-applicant (with PMC and TVA), and to reflect the realignment of the respective project participants' roles. Under this realignment, ERDA assumed the lead management role in the integrated CRBRP Project Office, which included PMC and TVA personnel, and TVA remained as the operator.[16] DOE is the successor-in-interest to ERDA.[17]

Commencing November 1975, extensive prehearing activities were conducted.[18] By March 1977, the NRC Staff had issued a Site Suitability

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[15] Roane County, which was admitted as a party on October 9, 1975, was granted leave to withdraw from all participation by the Board's Order entered December 13, 1976. The untimely petition to intervene of fourteen counties and municipalities was denied by the Board on August 26, 1976, and the denial was affirmed by the Appeal Board. *Project Management Corp.* (Clinch River Breeder Reactor Plant), LBP-76-31, 4 NRC 153 (1976), *aff'd,* ALAB-354, 4 NRC 383 (1976).

[16] *See* Joint Report at 35; JCAE Report at 19; 122 Cong. Rec. S10,613-22 (June 25, 1976); 122 Cong. Rec. H5835-98 (June 15, 1976).

[17] 42 U.S.C. § 7101 *et seq.*

[18] Intervenors filed fifteen sets of interrogatories, seven sets of requests for admissions, and four requests for production of documents against the Applicants. Intervenors filed twenty-two sets of interrogatories, seven sets of requests for admissions, and three requests for production of documents against the NRC Staff. An appeal arose concerning the admissibility of two Intervenor contentions, and the Commission held that certain programmatic issues previously considered in ERDA's LMFBR Program Environmental Statement would not be reconsidered in the CRBRP licensing proceedings. *See United States Energy Research and Development Administration* (Clinch River Breeder Reactor Plant), CLI-76-13, 4 NRC 67 (1976).

Report (SSR) and Final Environmental Statement (FES) (Staff Ex. 23). On March 28, 1977, the Board issued an Order (unpublished) for commencement of LWA hearings in Oak Ridge on June 14, 1977.

However, on April 20, 1977, the Carter Administration announced its decision to cancel the project. On April 22, 1977, ERDA filed a motion to suspend the proceedings, and on April 25, 1977, the Board issued an Order (unpublished) granting that motion. The Staff suspended its review of the application.

During the next four years, the project continued its design, research and development and procurement activities, although all licensing activities remained suspended. In each of those years, Congress preserved the project by appropriating substantial funding.[19]

In August 1981, President Reagan signed the Omnibus Budget Reconciliation Act of 1981 (Pub. L. No. 97-35), which expressed the intention that the project be expeditiously completed.[20] In a Nuclear Policy Statement of October 8, 1981, the President directed that "government agencies proceed with a demonstration of breeder reactor technology, including completion of the Clinch River Breeder Reactor."[21]

On January 11, 1982, the Applicants filed a motion to lift the suspension of hearings, which the Board granted. The Board entered an Order on February 11, 1982 (unpublished) establishing a schedule for the commencement of evidentiary hearings concerning LWA matters on August 23, 1982. All contentions related to the CP application were identified. The Intervenors restated or revised their original contentions, and filed additional contentions based upon new information. The Board on April 14 (LBP-82-31, 15 NRC 855) and April 22, 1982 (unpublished), ruled upon the admissibility, scope and applicability (LWA vs. CP) of Intervenors' contentions.

Extensive discovery followed.[22] On June 11, 1982, the Staff issued its updated SSR (NUREG-0786), which concluded that the Clinch River site was suitable for a reactor of the general size and type described in the application from the standpoint of radiological health and safety (Staff Ex. 2). The Advisory Committee on Reactor Safeguards (ACRS)

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[19] Pub. L. No. 95-240, March 7, 1978; Pub. L. No. 95-482, October 18, 1978; Pub. L. No. 96-86, October 12, 1979; Pub. L. No. 96-367, October 1, 1980; Pub. L. No. 96-536, December 16, 1980; Pub. L. No. 97-12, June 5, 1981.

[20] *See* H.R. Rep. No. 208, 97th Cong., 1st Sess. (1981); 127 Cong. Rec. S8998 (1981); 127 Cong. Rec. H5817-18 (1981).

[21] 17 Weekly Comp. Pres. Doc. 1101-02 (Oct. 12, 1981).

[22] By April 30, 1982, Applicants and Staff had updated their responses to Intervenors' 1975-77 discovery. As of the close of discovery on June 30, 1982, Intervenors had also filed an additional four sets of interrogatories, four sets of requests for admissions, and three requests for production of documents, and had deposed five persons from the NRC Staff and eleven persons from the Applicants.

issued a letter dated July 13, 1982, which supported the Staff's site suitability conclusion (Staff Ex. 4).

Site suitability hearings were conducted August 23-27, 1982. The Board then reopened discovery on all environmental issues, and held hearings November 16-19, 1982, and December 13-17, 1982 to take evidence concerning the remaining environmental issues.[23] The Board issued a Partial Initial Decision on February 28, 1983, which addressed all pertinent radiological site suitability and environmental issues, and concluded, *inter alia,* that: (1) the Clinch River site is suitable for a reactor of the general size and type proposed in the CRBRP application from the standpoint of radiological health and safety; (2) the contents of the Final Environmental Statement and the Final Supplement to the Final Environmental Statement (Staff Ex. 23 and 24) were affirmed; (3) the requirements of NEPA and 10 C.F.R. Part 51 had been complied with in the proceeding; and (4) an LWA should be issued for the CRBRP pursuant to 10 C.F.R. § 50.10(e).[24]

The Board then opened discovery on all remaining contentions in preparation for CP evidentiary hearings, and on March 11, 1983, the Staff issued its Safety Evaluation Report (SER) for the CRBRP (Staff Ex. 26-28). During the discovery period,[25] Intervenors filed responses to Applicants' and Staff's discovery requests which expressly stated that they wished to withdraw their Contentions 2(f), (g) and (h), 9(a), (b), (d), and (e), 10, and 11(a).[26] The Board granted the Applicants' unopposed motions to dismiss those contentions on May 17, 1983.

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[23] Neither the State of Tennessee Attorney General nor the City of Oak Ridge participated actively in these LWA evidentiary hearings. The Board received the "Position Paper of the Tennessee Attorney General on Socio-Economic Impact Matters and Other Matters Relating to the Clinch River Breeder Reactor Plant," dated November 10, 1982, and "The City of Oak Ridge's Statement Relative to the Socio-Economic Impact of the Clinch River Breeder Reactor Plant," dated November 12, 1982. At the direction of the Board (Tr. 3356-58; Tr. 7104), the Applicants and Staff filed, on January 11, 1983, Responses to the Attorney General's Position Paper and the City's Statement. Neither the Attorney General nor the City conducted cross-examination, presented witnesses, or introduced documentary evidence concerning the socio-economic matters raised by their respective Position Paper and Statement. The Board's February 28, 1983 Partial Initial Decision (LBP-83-8, 17 NRC 158 (1983)), resolved the issues raised in the Position Paper and Statement.

[24] LBP-83-8, 17 NRC 158 (1983). By order of March 28, 1983 (unpublished), the Commission itself determined that it would conduct the "immediate effectiveness" review of the Partial Initial Decision. On May 5, 1983, the Commission found that there was no reason to stay the effectiveness of that decision. (Commission Order dated May 5, 1983 (unpublished)). As noted above, this Partial Initial Decision (LWA) was vacated on the grounds of mootness by the Appeal Board on December 15, 1983 (ALAB-755, 18 NRC 1337).

[25] During this period Intervenors filed five sets of interrogatories and one document request. Applicants filed four sets of interrogatories and requests for admissions, and conducted one deposition. The Staff filed five sets of interrogatories and requests for admissions, and conducted one deposition. In addition, after the close of discovery, the Board granted Intervenors' request for additional discovery on the Staff's HCDA dose calculations (Tr. 7188-7202).

[26] *See* Intervenors' April 19, 1983 Response to Applicants' Eighth Set of Interrogatories and Intervenors' April 22, 1983 Response to NRC Staff's First Set of Construction Permit Interrogatories.

The ACRS issued its report on the CRBRP Construction Permit application on April 19, 1983. The report concluded that, if the matters noted therein and the open items described in the SER were resolved in a satisfactory manner, the CRBRP can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public (Staff Ex. 31). The Staff issued Supplements 1 and 2 to the SER on May 2 and 20, 1983, respectively, which resolved all open issues identified in the SER and ACRS Report (Staff Ex. 29 and 30).

The Applicants moved for summary disposition of Intervenors' Contention 9(g), and for partial summary disposition on Intervenors' Contentions 9(c) and 9(f), which motions were granted on June 29, 1983 (Tr. 7306). The Intervenors moved on June 21 to withdraw all of their remaining contentions from consideration at the CP hearings, and requested permission to submit a written statement. At a June 29, 1983 Conference with Counsel, the Board granted Intervenors' motion and request, and dismissed Intervenors as parties to the proceeding (Tr. 7333). Consequently, this proceeding became and remains uncontested,[27] and only the Applicants and the Staff are parties to the proceeding. The State of Tennessee Attorney General and City of Oak Ridge remained as an "interested state" and "municipality," respectively, under 10 C.F.R. § 2.715(c), but as described above, neither participated in the CP proceeding.

The CP evidentiary hearings were held and completed in Oak Ridge August 8-11, 1983. The record of the CP hearings focused upon the following areas of inquiry: (a) whether a hypothetical core disruptive accident should be a design basis accident; (b) the adequacy of Applicants' and Staff's HCDA analyses; and (c) seventeen (17) Board Areas of Inquiry, which are reproduced in Appendix D hereto.[28]

## Opinion

### I. DESIGN APPROACH

Safety is a characteristic of paramount importance to nuclear electric plants, to which diligent attention must be paid from the beginning

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[27] NRDC filed an untimely motion for leave to reintervene in this proceeding on November 23, 1983, but such motion was not granted. *See* note 7, *supra.*

[28] In addition, the Board also considered: (a) the matter of evacuation of nearby DOE industrial facilities in the event of an accident at CRBRP (*see* Partial Initial Decision, Finding 52); and (b) the feasibility of implementing design and operational changes, if any, resulting from completion of Applicants' probabilistic risk assessment (PRA) (Tr. 7340-41).

drawing-board or design phase through to plant completion and operation. As a necessary backdrop, a competent design safety approach requires a design safety philosophy as guidance. The Clinch River Breeder Reactor Plant (CRBRP) design safety effort has followed a guiding philosophy that established early-on the importance of achieving a level of safety comparable to light water reactor (LWR) plants, by using a defense-in-depth approach[29] analogous to that used for LWRs. This defense-in-depth approach has been translated by the Applicants into three levels of design effort that are illustrative of the measures taken to prevent and mitigate accidents. In addition, an extra measure of protection has been provided by imposing structural and thermal margins beyond the design base.[30]

Because of the more limited experience with liquid metal fast breeder reactors (LMFBRs) relative to LWRs, the safety philosophy of the CRBRP has gone a significant step further to require the provision of additional features and capabilities to assure that there is a low likelihood of containment failure and other unacceptable consequences associated with disruptive core melt accidents beyond the design basis. Major design emphasis has been placed upon the prevention of accidents that could lead to core melt and disruption and the loss of containment integrity.[31]

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[29] The defense-in-depth design safety approach is a three-level approach defined as follows for the CRBRP:

> The first level of safety provides criteria for reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic features of the design, such as quality assurance, redundancy, diversity, independence, maintainability, testability, inspectability, and fail-safe characteristics. The plant design criteria must not only accommodate steady-state power conditions, but also have adequate tolerance for normal operating transients, such as start-up, shutdown, and load following.
>
> The second level of safety provides criteria for protection against anticipated and unlikely faults, such as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel-handling errors, that might occur in spite of the care taken in design, construction, and operation of the plant. This level of safety for the public is provided by redundancy of critical components as well as by protection devices and systems designed to ensure that such events will be arrested. The requirements for these protection systems must be based on a spectrum of occurrences that the plant design must safely accommodate. Conservative design practices, including the provision of redundant detecting and actuating equipment, must be incorporated in the protection systems to ensure both the effectiveness and reliability of this second level of design.
>
> The third level of safety establishes criteria that supplement the first two levels by providing acceptable plant response to extremely unlikely faults such as pipe leaks, sodium fires, or sodium-water reactions. Although occurrence of these faults is of low probability, appropriate engineered safety features must be incorporated into the CRBRP design to safely accommodate such events. Conservative assumptions and evaluation methods are used to develop adequate designs. In addition, conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant. These include such low-probability events as severe earthquakes, tornadoes, and floods. These faults and natural phenomena combine to define the design-basis envelope.

[30] Findings No. 23-26.

[31] Finding No. 1.

As a measure to control the implementation of this philosophy, principal design criteria (PDC) were developed by the NRC staff (Staff), using the LWR criteria of Appendix A to 10 C.F.R. Part 50 as guidance where appropriate. Sixty design criteria (several of which are unique to the CRBRP) have been defined; CRBRP conformance with these is required by the Staff.[32]

Simply stated, the ability to prevent accidents in the CRBRP is directly dependent upon the ability of the reactor system and its control features to avoid two accident initiating conditions:

- Impaired heat removal capability beyond that necessary to dissipate the normal (including shutdown) heat generation rate within the reactor core;
- Excessive heat generation rate within the reactor core beyond that which the properly functioning heat dissipative systems (cooling capability) can accommodate.

Should either of these conditions obtain, an accident is initiated, the severity of which will depend upon the efficacy of numerous design features in preventing or arresting the progress of the accident.[33] Thus, not surprisingly, these design features play a dominant role with respect to plant safety.

The principal design features of importance to safety have been described in detail and carefully reviewed by the Board. They need only be summarized here. The design safety philosophy underlying these features has two additional facets:

- Redundancy (two or more features available to accomplish the same purpose) accompanied by functional independence and diversity of design to minimize single cause failures;[34]
- Use of existing technology so that prior experience can provide guidance to design feasibility, and to practicality of implementation and application, including a basis for confidence with respect to reliability.[35]

The principal design features important to safety are: (a) the reactor shutdown systems (RSSs); (b) the shutdown heat removal systems (SHRSs); (c) features to prevent or minimize the chance of catastrophic rupture of large primary heat transport system (PHTS) pipes; (d) features to prevent local imbalances between heat generation and removal (within an individual fuel pin or bundle of pins) from propagating

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[32] Finding No. 2.
[33] Finding No. 3.
[34] Finding No. 2.
[35] 17 NRC 158, 169 (1983); Appl. Ex. 71, at 15.1-4.

throughout the core; and (e) containment and confinement structures that will, should core-melting and disruption occur, minimize any adverse impact upon public health and safety. The Staff requires that single failures within each of these systems not disable safety effectiveness.[36]

The RSSs consist of two fast-acting shutdown systems (rather than one as in LWRs), each of which is independently capable of shutting down the reactor, and both of which actuate automatically upon loss of power. The SHRS provides four heat removal paths, each independently capable of removing all decay heat. Station blackout (loss of offsite power and loss of operability of all onsite, standby diesel generators) does not disable the SHRS; natural convective circulation of the coolant, a steam-driven auxiliary feedwater system (AFWS) pump, and battery-powered instrumentation and controls are available to provide continued core heat removal. The RSSs and the heat removal paths of the SHRS function automatically without need for operator initiation; operator response consists merely of confirming and monitoring those functions.[37]

The features for prevention of large pipe ruptures rely upon four successive levels of protection, which are supported by extensive analytical and experimental evidence, and by domestic and foreign operating experience. The four levels are: (1) stringent quality standards limit the potential for crack initiation from preexisting material flaws; (2) even if flaws exist, the fracture toughness of the piping has been shown to limit growth of cracks to sizes well below that necessary to penetrate a pipe; (3) even if a crack should grow and penetrate the pipe, it would be detected by a sensitive leak detection system well before any rupture could occur; and (4) even if a crack should grow undetected, the crack would have dimensions well below those at which a pipe rupture would occur. In addition, the CRBRP can accommodate coolant pipe leaks substantially larger than that for which continued operation will be permitted without a significant reduction in heat removal.[38]

The features to prevent progression of local imbalances between heat generation and heat removal to core-wide involvement incorporate passive mechanical interlocks to assure proper fuel subassembly positioning, and a multiplicity of redundant inlet flow paths to prevent debris-induced blockage of any subassembly. Steel hexagonal subassembly ducts house each fuel rod bundle (subassembly) to limit inherently

[36] Findings No. 4-14 [12 responds to Board inquiry Item 2, Appendix D]. *See also* Staff Ex. 5, at 3; Staff Ex. 32, at 6; Tr. 8041.

[37] Findings No. 4-5 [5 responds to the Board's inquiry in Item 4 of Appendix D].

[38] Findings No. 6-7 [includes response to Board's inquiry in Item 3 of Appendix D].

the propagation of local imbalances between subassemblies. Extensive analyses, experimental data, and domestic and foreign operating experience all show that propagation beyond a single subassembly is highly unlikely. Any localized fuel failures can be detected by independent systems at levels well below those that could result in a significant local imbalance, and pending completion of testing at EBR-II, the Staff has identified operating restrictions precluding any real possibility of a local imbalance that could progress to core melt and disruption.[39]

Based upon its concerns about certain analytical methods and assumptions of Applicants pertaining to fuel design, the Staff has identified specific operational fallback positions that can be imposed to mitigate these concerns if future analytical and experimental data do not substantiate the Applicants' proposed design. The Staff has explained to the Board's satisfaction why it has concluded that it is unlikely that any of these fallback measures will need to be implemented; and if implemented, why it has concluded that the programmatic objectives of the CRBRP will not be compromised[40] (*see* Board's inquiry Item 13, Appendix D).

Since the CRBR is designed to continue operation with on the order of 1% of the fuel exhibiting gaseous fission product leakage, the Board was concerned lest the gaseous leak path also provide an opportunity for inleakage of sodium to the detriment of fuel pin performance (Board's inquiry Item 14, Appendix D). The Staff has addressed this concern to the Board's satisfaction. Not only are there independent methods to detect fission gas leakage and to detect sodium-fuel contacts, but additionally it has been agreed between Applicants and Staff that gas-leaking pins will be withdrawn from service sufficiently promptly to obviate concerns about adverse impacts of any sodium inleakage.[41]

The containment-confinement system (hereinafter referred to as the containment) comprises a welded steel containment shell around the reactor vessel, that is surrounded by a reinforced concrete confinement building, there being a 5-foot annulus in between. The intent of these structures is to control and limit radiological releases to the environment. The design approach for this system makes it the ultimate barrier for protection of the environment under the challenge of an accident that progresses to core meltdown and disruption. Structural and thermal margins have been incorporated into the design to permit meeting a short-term (minutes) mechanical challenge (should core disruption

[39] Findings No. 8-10.
[40] Finding No. 19.
[41] Finding No. 20.

be accompanied by a significant release of energy) as well as a long-term challenge (hours to months) from temperature and pressure increases that might result if the reactor vessel fails.[42]

It was determined during the LWA-1 phase of this proceeding that the Staff's final position on the adequacy of the containment design would be presented when its SER is published (17 NRC 158, 171 (1983)). In its SER, the Staff reserves, for FSAR and OL reviews, several aspects of the containment design for final confirmation and approval. Thus the SER (CP version) cannot literally represent the final position of the Staff at this time. Nevertheless, our review of the evidentiary record regarding containment design and expected performance in the face of accident threats (especially accidents involving core melting and disruption) leads us to the opinion that we have no substantive reservations about the adequacy of the design concept that would preclude the issuance of a construction permit, which opinion is consistent with that of the Staff.[43]

During the contested LWA hearing, Intervenors expressed concern about the design of heat exchangers proposed for the CRBRP. Although evidence at that time did not indicate a substantive problem, the Board indicated its intent to explore this matter further during the CP proceeding (17 NRC 158, 187 (1983); portion of Board's inquiry in Item 8, Appendix D).

The CRBRP HTS comprises three heat transport loops and each loop has three identical tube and shell heat exchanger modules; two of these modules, in parallel, serve as evaporators (steam generators), whereas the third module is used as a superheater. Elevated temperature design methods specified in ASME Code, Code Cases and RDT standards were used for these heat exchangers. The most severe thermal transient that could be postulated was analyzed and found not to provide an unacceptable challenge to the design. Rupture disks are provided to prevent damage from pressure transients. These design details and analyses will be augmented by a comprehensive component test program (to be completed prior to fabrication of the actual CRBRP units). The Board is satisfied with the adequacy of Applicants' efforts in these regards.[44]

The Board has reviewed the CRBRP design approach from the standpoint of its comprehensiveness with respect to a safety-inspired design philosophy. How well the detailed design fares in the face of specific accident threats comes later. At this point, we are satisfied that Applicants

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[42] Findings No. 13-15 [satisfies the containment portion of Board's inquiry in Item 8 of Appendix D].

[43] Finding No. 22.

[44] Finding No. 21.

have adopted a cautious and conservative approach that emphasizes plant safety without substantive sacrifice to operability and maintenance. We note that Applicants are undertaking two efforts that will critique and contribute to design adequacy and safety:

- A probabilistic risk assessment (PRA), previously scheduled for completion in December 1984, to obtain an improved understanding of the relative importance of systems and components to overall plant reliability and risk;[45] and
- A reliability assurance program (RAP) that was to address all important plant safety features and was to continue over the entire life of the plant, providing input into the design and operation of the CRBRP.[46]

The Board was concerned about the implication of the identification, in the Staff's SER, of numerous design items being left for Staff consideration at the OL review stage (Board's inquiry Item 12, Appendix D). The Staff has addressed this concern and we are satisfied with the basis for its conclusions that cost, schedule and safety will not be compromised.[47] [The Board observes that the design and fabrication of CRBRP components have progressed further than is frequently the case at the time of an LWR CP licensing hearing. This situation can offer the opportunity for a more rigorous CP phase assessment of the adequacy of designs and their supporting analyses, which opportunity we appreciate. However, it also holds the potential for premature finalization by Applicants of possibly flawed design concepts. On balance, the Board feels that the current CRBRP design status profits from the former more than it suffers from the latter of these countervailing viewpoints.]

The Board was also concerned about research and development still required to augment effective CRBR fuel safeguards measures (Board's inquiry Item 10, Appendix D). Staff testimony alleviated this concern.[48]

Based upon our review of the foregoing information, the Board is of the opinion that a credible and competent design safety approach is under way.

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[45] Finding No. 16.

[46] Finding No. 17 [including information dispositive of the Board's interest expressed in Item 15 of Appendix D].

[47] Finding No. 18.

[48] Finding No. 26.

## II. ACCIDENT ANALYSES — INTRODUCTION

In the foregoing section, two classes of accidents are mentioned in discussing the CRBRP design safety approach: one class comprises those accidents during which the core remains intact and coolable; the other comprises those accidents during which core coolability may be reduced to the extent that some amount of core melting occurs, which may be accompanied by a kinetically significant disassembly [disruption] of the core. In the language of the evidence before us, these two accident classes are termed, respectively, a design basis accident (DBA) and a hypothetical core disruptive accident (HCDA or CDA).[49] This terminology is adopted here as appropriate to accident analysis discussions that follow.

As indicated previously, the NRC Staff has mandated that LMFBR safety is to be comparable to LWR safety, offered as underlying guidance to the design, construction and operation to the CRBRP.[50] This mandate leads to extending the accident severity analog between LWRs and LMFBRs to the point of likening the first class of CRBRP accidents to DBAs in LWRs and the second to Class 9 accidents[51] in LWRs. During both the previous LWA 1 (contested) hearing and the recently held CP

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[49] *Core Disruptive Accidents* (CDAs) — sometimes referred to as hypothetical core disruptive accidents (HCDAs) — are those accidents in which the physical and/or mechanical integrity of the core has been altered to an extent that effective core cooling may not be maintained. The loss of effective core cooling geometry may result in the release of orginally clad or contained fuel into the reactor vessel in some combination of solid, liquid or vapor forms and may be accompanied by a mechanically damaging energy release. *Design Basis Accidents* (DBAs) are those accidents whose likelihood of occurrence is deemed to be credible and for which the engineered safety features of a specific facility assure that the health and safety of the general public will not be endangered. DBAs are considered to be of insufficient severity to cause a loss of coolable geometry within the core.

[50] Finding No. 1.

[51] The designation "Class 9 accident" now has no official regulatory standing. Historically, the Class 9 designation originated with the publication, on December 1, 1971, of the (former) Atomic Energy Commission's (AEC) proposed Annex to Appendix D of 10 C.F.R. Part 50. Said Annex characterized Class 9 accidents as involving sequences of postulated successive failures more severe than the eight classes postulated for design basis accidents and for the protective systems and engineered safety features provided to protect against DBAs. On June 13, 1980 the Nuclear Regulatory Commission (NRC), the successor organization to the AEC responsible for nuclear power plant regulation, withdrew the proposed Annex and thereby abolished the formal Class 9 designation [45 Fed. Reg. 40,101 (1980)]. In so doing, however, the NRC stated

> that its Environmental Impact Statements shall include considerations of the site specific environmental impacts attributable to accident sequences that lead to releases of radiation and/or radioactive materials, *including sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core.* In this regard, attention shall be given both to the probability of occurrence of such releases and to the environmental consequences of such releases . . .

[*Ibid.*, emphasis added]. In the interim time period, the NRC's Staff had determined that the CRBRP accident analysis warranted consideration of Class 9 events, a determination reflected in the scope of its February 1977 Final Environmental Statement (FES) for the Plant. Consistent with the above-quoted excerpt, the NRC Staff's October 1982 supplement to the FES and its March 1983 Safety Evaluation Report deal with beyond-design-basis accidents that involve some degree of core melting and possibly an energetic disruption of the core — hence the term "core disruptive accident" and its association with the now obsolete term "Class 9 accident."

(uncontested) hearing, Applicants and the Staff introduced evidence in support of the proposition that CDAs should not be considered as design basis accidents. Originally, it had been the Staff's position that CDAs should be included in the spectrum of design basis accidents, based upon limited information available at the time. However, in May 1976, the Staff advised the Applicants that

> [i]t is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum. We will therefore not consider CDAs as design basis accidents. Nevertheless, because of the difference in the state of technology and experience between LMFBRs and LWRs, the consequent inability to evaluate the safety of the CRBR design as precisely as can be done for LWRs, and the absence of a quantitative risk assessment based on experience and data such as the Reactor Safety Study for LWRs, prudence dictates that additional measures be taken to limit consequences and reduce residual risks from potential CRBR accidents having a lower probability than design basis accidents to ensure that the public health and safety is adequately protected.

[*See* Staff Ex. 5, at 5.] With this as background, the thrust of the evidence before us is in context. We now discuss each of these accident classes in Sections IIA and IIB, and their attendant dose consequences in Section IIC.

### A. Design Basis Accidents

Design basis accidents (DBAs — *see* note 49, above) may be considered as a collection of events each of which is considered to have some likelihood of occurrence during the lifetime of the plant under consideration. They provide an envelope of "what if" occurrences that permits the plant design to be critiqued or tested in the face of off-normal conditions that require mitigation by active and passive design features. This critique in turn leads to the assessment of the adequacy of accident accommodation by the plant design. When events have been judged to be so improbable that they are not credible as events against which the plant design should be tested, they have been excluded from the DBA collection of events. The potential radiological consequences of the DBAs must also be evaluated to determine whether predicted consequences fall within appropriate dose guidelines.[52] The result of this latter evaluation forms a portion of the discussion presented in Section IIC, below. In this section, we shall address two questions:

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[52] Finding No. 27.

— Have all credible DBA events been identified for the CRBR, and
— Is the design of the plant and its protective systems capable of preventing these events or mitigating their impact?

The Applicants have identified and analyzed seventy-one events in connection with their DBA review. Applicants' and Staff's discussions of these events have placed them in the following categories:

— Reactivity insertion events,
— Undercooling events,
— Local fuel failures,
— Fuel handling and storage events,
— Sodium fires, and
— Miscellaneous other events.

Within each of these categories, qualitative classifications as to frequency of occurrence were designated by the Applicants, based upon whether the events are deemed to be "anticipated," "unlikely," or "extremely unlikely." Applicants' analysis format progresses from anticipated through extremely unlikely events in each category and provides, for each category and classification, a discussion of:

— Identification of causes and accident description,
— Analyses of effects and consequences, and
— Conclusions.[53]

In these analyses, the Applicants determined that there are three worst-case or umbrella events whose potential consequences justify detailed presentation. The remaining less severe events bounded by these three are reported in a more summary fashion. The bounding or umbrella events are:

— Reactivity insertion events, involving a $0.60 step increase of reactivity [*see* note 60, below] accompanied by the occurrence of a safe shutdown earthquake (SSE);
— Undercooling events, involving loss of offsite power; and
— Fuel handling and storage events, in which radioactive sodium pool cover gas may be released.

The Staff has reviewed the set of DBA events identified and analyzed by the Applicants and has determined that there are no serious inadequacies in the completeness of the set of events or in the results of Applicants' analytical approach that could not await resolution at the FSAR-OL review phase, although certain concerns were identified as requiring

[53] Finding No. 28.

additional analyses.[54] These Staff concerns principally involve the following:

- It is not clear that all credible malfunctions have been considered that could permit overcooling of sodium in the intermediate heat transport system (*e.g.*, loss of feedwater heating); these must be analyzed in the FSAR;
- The Applicants' analyses of failure modes and effects of the heat transport system, the control system and the cover gas system need to be taken into account in the FSAR to demonstrate that the DBAs will bound all credible off-normal plant conditions;
- The analyses of some of the reactivity insertion events involving the current heterogeneous core design concept suffer from the fact that in several instances Applicants have evaluated the differential changes resulting from the abandonment of the original homogeneous core design rather than having fully reevaluated all events for the current heterogeneous core design. The Staff will require a full evaluation in the FSAR for the OL review.

The Staff has emphasized that it does not view these concerns to mean that there is an inherent inadequacy in the CRBRP design; but rather the concerns are of a nature amenable to straightforward design or operational modifications should further analyses confirm the existence of problems.[55]

As mentioned, Applicants' analyses of all DBA sequences identified three DBAs (listed above) that represent worst-case or bounding events with respect to challenging the facility design parameters. One of these — an SSE concurrent with core compaction reactivity insertion — could, under conservative (pessimistic) assumptions, lead to some amount of fuel pellet melting. This analysis showed, however, that there would be insufficient thermal energy released or pressure generated to result in any significant loss of fuel cladding strength or to raise the sodium coolant temperature to its boiling point. Gaseous fission product release from some of the fuel pins might occur. However, under the worst-case assumption that this might happen to all of the 217 fuel pins in a given fuel pin bundle (fuel subassembly), a margin of more than 100°F below coolant boiling was predicted, with no functional or behavioral acceptance criteria being violated.[56]

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[54] Findings No. 29, 30.
[55] Finding No. 30.
[56] Finding No. 31.

The other two bounding events have less significant consequences so far as challenges to the accident accommodation features of the plant design are concerned. However, the release of radioactive materials as the result of certain fuel handling accidents can result in calculable releases to the environment. The resulting doses, as will be seen in Section IIC, below, are within dose guideline values. In summary, the Applicants have concluded that all DBAs are limited, terminated or mitigated by specific plant features that assure the reestablishment and/or maintenance of a balance between heat removal and heat generation in the reactor core.[57]

We turn now to the pivotal questions identified at the beginning of this section. In so doing, we note that this Section IIA does not stand alone: the answers to the questions identified above must draw (at least in part) upon support from Sections I (Design Approach, pp. 298-304), and III (Quality Assurance, pp. 322-24). Based upon our review of the evidence supporting this section and of each of the other cited Opinion sections, the Board opines in the affirmative to both questions. Additionally, we are of the opinion that none of the identified DBAs will progress to such severity as to result in core disruption; nor will any of the identified DBAs require calling upon containment protective features (*e.g.*, activation of the containment vent/purge system) provided to mitigate beyond-design-basis accidents.

### B. Beyond-Design-Basis Accidents

Consistent with discussions elsewhere in this section, the term CDA (*see* note 49, above) is used to denote a beyond-design-basis accident in which there has occurred some degree of melting of fuel and cladding sufficient to allow enough relocation of core material to affect reactor behavior in an adverse or unwanted manner. Under circumstances to be discussed later, this relocation of core material can take place with substantial enough driving force (kinetic energy) to generate damage beyond the internals of the core. Thus, CDAs may be characterized as being either nonenergetic or energetic. In terms of core accident kinetics, there is no sharp demarcation between nonenergetic and energetic behavior. In the context of the instant discussion, the term "nonenergetic" signifies that there is insufficient energy released to cause damage to those physical structures within the reactor vessel that surround and support the core. [With respect to the foregoing, it should be recognized that at the start of accident initiation it is undeterminable

[57] Finding No. 32.

as to whether an accident will turn out to be a DBA or a CDA, since that outcome is determined by whether the RSSs and other safety systems function subsequently upon demand, and as intended.] The general point of departure for CDA analyses is the assumption that failure of the actuation of the RSSs occurs and delays reactor shutdown. Accident initiation events accompanied by failure of the RSSs to shut down the reactor are termed unprotected events.[58] The only noteworthy exception to the assumption of RSS failure as the premise for CDA initiation lies with the possibility that proper functioning of the RSSs may be followed by a failure of the SHRS to remove core (or fission product) decay heat rapidly enough to protect the core. This is termed a protected loss-of-heat-sink (LOHS) event. Thus, in addition to characterizing CDAs as either nonenergetic or energetic, they can further be characterized as protected or unprotected events.

The potential for core disruption arises from four postulated situations:

1. A decrease of core cooling without an appropriate reduction of core heat generation, termed an unprotected loss-of-flow (LOF) accident;
2. A core overpower condition resulting from an unprotected insertion of reactivity, termed an unprotected transient overpower (TOP) accident;
3. A protected LOHS accident, mentioned above; and
4. The propagation throughout a significant portion of the core of subassembly (fuel bundle) failures initiated by the failure of one subassembly or one fuel pin within a subassembly.

Based upon design details, analyses and experimental information, Applicants and Staff have concluded that core disruption through the mechanism of the propagation of individual fuel failures can be neglected, leaving three of the above four postulated situations for discussion. In addition to assessing each of the three above identified postulates and evaluating their energetic consequences through to accident termination, various combinations of the three accident sequences have been considered in combination with extreme external events such as an earthquake beyond the safe shutdown earthquake. The Staff's review of these indicates that those few cases for which severe energetic behavior cannot be precluded at this time are of sufficiently low probability to be neglected.[59]

The Staff had an independent assessment made of the energetics of CDA accident sequences corresponding to the first three of the above

[58] Finding No. 33.
[59] Finding No. 34.

postulated situations. That assessment considered in detail the accident behavior for these three classes of events and inquired into the capability of the CRBRP design to accommodate the energetic impacts of various core events in a realistic yet conservative fashion. This accommodation capability was found to be equivalent to the theoretical maximum energy release that could result from a rate of core reactivity increase (in CRBR) of about 200 $/s,[60] which translates to a maximum energy release (for CRBR) of 2550 MJ (million joules[61]). It is noted that this energy release corresponds to that which would be required to produce a slug impact kinetic energy of about 75 MJ, and which represents the impact resistance design capability of the reactor vessel closure head. This accommodation capability was compared with the consequences of energetic CDA behavior which, as a consequence of the CRBR heterogeneous core design, was shown to result only from advanced core disruption configurations that were subsequently subject to gravity-driven recriticality. Upon analysis, the accident behavior was found to be bounded by neutronic activity conservatively assessed to be associated with an equivalent reactivity ramp increase of 100 $/s, which corresponds to an energetic equivalence of 1130 MJ. Such an activity is self-terminating because it promotes the removal of enough fuel from the core to make the core subcritical and accomplishes this before complete core melting and whole core pool formation can occur. However, even the formation of a molten whole core pool was found to produce energetic releases below that which the CRBRP design can accommodate. The Staff's independent assessment concluded that failure of the closure head resulting in an early challenge to the containment is physically unreasonable.[62]

The reactor vessel closure head provides a barrier between the reactor core and the containment building. If this closure head remains intact

[60] The term "$/s" is read as "dollars per second." Its use to describe a rate of change of reactivity may be understood as follows: in the fission process, two types of neutrons are emitted that are characterized as being either "prompt" (*i.e.*, emitted at the moment of fissioning) or "delayed" (*i.e.*, emitted subsequent to the formation of the fission fragments and deriving from said fragments). A reactor operating at a constant rate of fission events is said to be "critical" if the total number of fission events per unit time is sustained at a constant value by the available supply of prompt and delayed neutrons. This condition is termed "delayed critical" since, were there no delayed neutrons, the chain reaction would die out; it is independent of the power level at which delayed critical occurs. Any change of reactor geometry that results in an increase in the number of fission events per unit time beyond that required to maintain the reactor at delayed critical is said to result from a "reactivity increase." A uniform rate of reactivity increase with time is termed a "ramp increase" of reactivity. As reactivity increases, a condition can be reached wherein the prompt neutrons alone could sustain the chain reaction and the reactor is then said to be "prompt critical." The amount of reactivity increase required to bring a reactor from delayed critical to prompt critical is termed a "dollar."

[61] The joule is a unit of energy or work equivalent to 1 watt-second or approximately equivalent to 0.738 foot-pound.

[62] Finding No. 35

after the onset of a CDA, then one of two events will occur: either the disrupted core material will remain in the primary system, or the reactor vessel will fail and the core debris will eventually penetrate into the reactor cavity (*see* Fig. 1 (Staff Ex. 41 at Tr. 8279) for a graphic representation of the geometry involved). In either event, the consequences have been analyzed to be acceptable and they do not present an early challenge to the containment. If the CDA results in closure head failure, radioactive materials and sodium would be released into the containment and be available for release to the environment, and/or missiles or sodium fires could present an early challenge to the integrity of the containment. Extensive analyses by the Staff and its consultants have shown that CDA energetics of sufficient magnitude to fail the closure head are highly unlikely to occur and are not expected to fail the balance of the reactor vessel.[63]

In evaluating the level of energetics required to produce significant damage, Staff consultants gave consideration to the fact that in between the core and the reactor vessel there exist structures that effectively form an inner protective containment: these structures are the core barrel, the upper internal structure and the core support structure, which, collectively, are referred to as the CB/UIS/CSS envelope. This envelope of structures is able to absorb energy from a disrupting core and lessen the challenge to the reactor vessel from a CDA. Analyses have shown that a level of energetics equivalent to about 1130 MJ would be required to breach this inner containment and thus no release could be expected to breach the reactor vessel for any energetic core disassembly below this level. At higher energetic levels, an upward displacement of the UIS would take place permitting a longer-term expansion against the sodium pool, and would provide the only mechanical opportunity for large-scale sodium-fuel contact. However, approximately twice this inner containment breaching energy release would be required (*i.e.*, about 2550 MJ) to produce the 75-MJ slug kinetic impact energy cited above to challenge the design capability of the reactor vessel closure head.[64]

Three areas of formal Board inquiry (Appendix D, Items 11, 16, and 17) dealt with specific aspects of CDA analyses made by the Staff. Item 11 (release energetics) and Item 16 (aerosol behavior) have been addressed dispositively to the Board's satisfaction. Item 17 dealt with eight areas of Staff concern regarding CDA analyses that were each reviewed as to the adequacy of Applicants' responses to Staff concerns. With one

[63] Findings No. 35, 36.
[64] Finding No. 37.

exception, the Staff explained the basis for its satisfaction with Applicants' responses. The one exception involves the consideration of whether fission product gas acting upon fuel in the core, early in the core disruption process, can significantly increase the energetics of a CDA. As the record now stands, Applicants' position on this matter has not been accepted by the Staff; the matter has been noted for resolution at the operating license review stage. Since the Applicants have agreed to a fuel design modification that can obviate this concern if it cannot be resolved by further analyses,[65] the Board is satisfied with the current status.

We have reviewed the Applicants' analyses of CDAs, the Staff's critique of same, and the Staff's detailed, independent analyses. As noted with respect to the previous section, this Section IIB does not stand alone: a proper consideration of CDAs must in part also draw upon support from Sections I (Design Approach, pp. 298-304), and III (Quality Assurance, pp. 322-24). Based upon our review of all relevant evidence, the Board has formed the opinion that failure of the reactor vessel along with an early (less than 24 hours) breach of the integrity of the containment is physically unreasonable as a consequence of a core disruptive accident. [The 24-hour time period has significance with respect to emergency response protective action: Applicants have testified that if evacuation of the 10-mile plume exposure pathway is decided upon, it can be achieved in about 9 hours.][66]

The Applicants and the Staff have each concluded that CDAs need not be considered within the spectrum of DBAs for the CRBRP. Their conclusions are based upon extensive evidentiary material that lends credence to the thesis that the Plant is being designed and can be constructed and operated in a manner so as to preclude CDAs as credible events.[67] These considerations are discussed in various parts of the instant memorandum and lead us to the opinion that, based upon the present status of the evidence before us, there is no substantive barrier to the achievement of such an objective, and that failure of the reactor vessel with a resultant early challenge to containment integrity is physically unreasonable.

Owing to popular interest in a hypothetical phenomenon called the China Syndrome,[68] the Board inquired of Staff witnesses during the hear-

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[65] Finding No. 38.

[66] Findings No. 39, 108.

[67] Finding No. 39.

[68] *China Syndrome* is a hypothetical phenomenon in which, following a whole-core-melt accident, molten fuel and core debris are postulated to melt through the reactor building foundation, and the earth beneath it and emerge, ultimately, into China (gravitational physics and geography notwithstanding).

ing whether a CDA at the CRBRP could generate such a phenomenon. The Staff responded that based upon their analysis, the China Syndrome would not be expected to occur. However, uncertainties in that analysis led the Staff to further analyses to assure itself that there is not an unacceptable groundwater pathway for the spread of contamination (Tr. 8493-99; *see also* Staff Ex. 41, at 15; Tr. 8286).

**C. Dose Consequences of Accidents — Introduction**

The principal thrust of this section is to examine, based upon the evidence before us, how the public health and safety are impacted radiologically by DBAs and CDAs at the CRBRP. As a logical prelude to this endeavor, however, we review briefly certain historical and background matters relevant to this proceeding and then summarize what the evidence shows about the impacts of normal operation.

Culminating the LWA phase of this proceeding, a determination was made that the proposed Clinch River site is suitable for siting a facility of the general type and size of the proposed CRBRP (17 NRC 158, at 256 (1983)). Quantitatively, that determination followed from a three-step process that first assumed a radiological site suitability source term (SSST) analogous to that used for LWR analyses; that, secondly, computed the resulting dose (from release of the SSST into the containment) to a member of the general public in the vicinity of the site based upon CRBRP design features and site characteristics; and that, thirdly, compared the resulting SSST doses with guideline doses that are either directly available from or derivable from Title 10 dose guidelines provided for the explicit purpose of evaluating site suitability. The results of these analyses confirm the suitability of the site for the proposed facility.

The radiological doses and dose commitments[69] resulting from operating nuclear power plants are well known and documented. Accurate measurements of radiation and radioactive contaminants can be made with very high sensitivity so that the existence of much smaller amounts of radioisotopes can be recorded than can be associated with any possible

[69] The terms "dose" and "dose commitment" are used to distinguish between two types of exposures: exposures due to radiation external to the body, the dose being based upon radiation level and duration of exposure, and exposures due to radiation internal to the body (inhaled or ingested) that commits the exposed person to a future dose due to the duration of body retention of the inhaled or ingested radioactive material. Standard NRC practice is to use a 50-year time period for dose commitment analyses (*see, e.g.*, Reg. Guide 1.109). In the language of this memorandum and consistent with the jargon of the trade and the evidentiary material before us, the term "dose" is taken to mean both the "now" dose from external radiation and the (50-year) committed dose from ingestion and/or inhalation, unless explicitly stated otherwise.

observable ill effects. Furthermore, the effects of radiation on living systems have for decades been subject to intensive investigation and consideration by individual scientists as well as by select committees, constituted periodically to assess radiation dose effects objectively and independently. [Two noteworthy and authoritative work products of such committees are:

- "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980," by the Committee on the Biological Effects of Ionizing Radiation (BEIR Committee) of the National Academy of Sciences; and
- "Ionizing Radiation: Sources and Biological Effects," United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), 1982 Report to the General Assembly.]

Although, as in the case of chemical contaminants, there is debate about the exact extent of the effects of very low levels of radiation that result from nuclear power plant effluents, upper bound limits of deleterious effects are well established and amenable to standard methods of risk analysis. Thus the risks to a maximally exposed member of the public outside of nuclear plant site boundaries can be readily determined. The results of the Staff's determinations for normal CRBR operation are now summarized.

The average annual radiation dose due to normal operation of the CRBRP to an individual living at the site boundary would be less than one mrem/yr,[70] and the cumulative dose to the estimated year 2010 U.S. population within 50 miles would be about 0.1 person-rem/yr. These doses include contributions from air, water and food, and are about 1% and about 0.001%, respectively, of those received from natural background radiation. The total dose to the general public from operation of supporting CRBR fuel cycle facilities and transportation of radioactive fuel and wastes from the CRBRP is estimated to be 170 person-rems/yr; this is insignificant when compared to the estimated 28 million person-rems/yr received by the same U.S. population from natural sources. As to cancer fatalities and genetic defects, the risks to the general public from exposure to radioactivity associated with the annual normal operation of the CRBRP are extremely small fractions (less than 10 per

[70] The term "1 mrem/yr" refers to a dose rate of one millirem (one thousandth of a rem) per year. The term "rem" is an acronym for a unit of radiation dose called "roentgen equivalent man." It is defined as that quantity of ionizing radiation that produces the same biological damage in a person as would occur from the liberation of 100 ergs (or $10^{-5}$ joules) of energy per gram of body material that has absorbed some portion of the radiation. The term "rad" (for "radiation absorbed dose") will appear subsequently. Unlike the rem, the rad is used to express a dose experienced by biological as well as non-biological materials. Like the rem, the rad is a unit of absorbed dose of radiation that results in the liberation of 100 ergs per gram of absorbing material (100 erg/gram = 0.01 joule/kilogram).

billion) of the estimated normal incidence of cancer fatalities in the year 2010 population and of the estimated genetic abnormalities for the first five generations subsequent to that year. [It is generally assumed that radiation-induced cancer incidence ultimately leading to fatality does not impact progeny whereas a similarly induced genetic defect does.] Thus the Staff concludes that the contribution of normal CRBRP operation (including related fuel cycle activities) to radiological public health and safety risks will be very small.[71]

Having reviewed the methodology, the inputs and portions of the pertinent literature used by the Staff, the Board is of the opinion that the excess (above background) radiation doses attributable to normal CRBRP operation are indeed quite small, and comparable to those associated with the normal operation of light water reactors.

*Accident Doses — DBAs*

We have seen previously that none of the identified DBAs will either progress to core disruption or will require calling upon containment protective features that have been provided to protect the public from beyond-design-basis accidents, or CDAs.

Applicants have alleged that the maximum dose resulting from the release to the containment of radioactive material from a DBA is at least a factor of 25 less than the corresponding dose resulting from an SSST release to the containment. This statement has not been controverted by the Staff nor has our examination of the evidence uncovered any contradictory information. In fact, Applicants' analysis of the offsite dose resulting from one of the three worst-case or bounding DBAs previously discussed (fuel handling) yielded a maximum 2-hour site boundary dose of about 3 millirem, significantly smaller than the "factor of 25 less than" cited above. Be that as it may, an SSST release to containment was the starting point, as noted previously, for assessing offsite doses that were compared with 10 C.F.R. Part 100 dose guidelines *for the explicit purpose of evaluating site suitability.* The Part 100 guidelines do not represent acceptable doses for the general public. To put the above 3-millirem DBA dose into acceptable dose perspective, we observe that Appendix I to 10 C.F.R. Part 50 permits annual whole-body doses to persons in unrestricted areas of up to 3 millirems from liquid effluents and up to 10 millirads from gaseous effluents.[72] This leads us to the opinion that the CRBRP as currently envisioned has achieved design com-

[71] Finding No. 41.
[72] Finding No. 42.

parability with LWRs insofar as DBA offsite dose implications are concerned.

[We note here that the Code of Federal Regulations, Title 10 (Energy), Chapter I (Nuclear Regulatory Commission) provides three sets of guidelines governing radioactive materials and radiation doses. Their scopes and purposes are summarized later.][73]

*Accident Doses — CDAs*

The Staff established a safety objective that there be no greater than one chance in a million ($10^{-6}$) per year of operation for an accident to occur having consequences greater than the 10 C.F.R. Part 100 dose guidelines, this being a design goal rather than a fixed number that must be demonstrated for a given plant. Accident consequence (dose) analyses and the dose guidelines with which they are compared are based upon the possibility of significant core melting and the release to the containment of a portion of radioactive core materials prescribed by the Staff (for CRBR) to be:

| | | |
|---|---|---|
| Noble gases | 100% | |
| Halogens | 50% | (25% airborne) |
| Balance of fission products | 1% | |
| Pu (plutonium from core) | 1% | |

This is the core inventory release that has been mentioned earlier as the site suitability source term (SSST). The ultimate dose to a person who remains at the site boundary for 2 hours or at the distance to the low population zone during the entire passage of any containment release (established by Part 100) will be determined by site meteorology and by containment accident response characteristics. The Applicants have summarized and compared these dose guidelines and SSST doses for CRBRP. We find the resultant doses to be adequately below Part 100 guideline values.[74]

Owing to the fact that the NRC has indicated its intent to review its radiological source term assumptions, the Board inquired of the Staff what its views are with respect to any impact upon the CRBRP (Appendix D, Item 1). The Staff currently expects that such an impact will be minimal but it will ensure that the results of the source term

[73] Finding No. 43.

[74] Staff Ex. 5, at 2, 5; 10 C.F.R. § 100.11; Finding No. 44.

review will be factored into CRBRP considerations during the OL phase of licensing review. This satisfies the Board's current interest.[75]

The Staff's analyses of CDAs have identified four categories of primary plant system behavior in order of increasing radioisotope releases to the containment and increasing potential threat to containment integrity:

I. Primary coolant system remains intact, with no significant release to the containment atmosphere.

II. Failure of long-term (several hours after accident initiation) heat removal causes primary system to fail, with eventual release through reactor cavity vents of radioactive materials and sodium into the containment atmosphere.

III. Primary system seals (at the reactor vessel closure head) partially fail due to excessive mechanical and thermal loads from energetic core disruption. A limited release of core inventory into the containment occurs immediately.

IV. Primary system seals fail completely due to energetic core disruption, allowing a large release of core inventory immediately into the containment followed by vaporized sodium and radionuclides.

Three modes of containment response to the above were in turn considered:

A. Containment leakage control and filtered venting operate in accordance with design intent;

B. Containment fails at about 24 hours after accident initiation due to overpressure primarily from sodium vapor and hydrogen; and

C. Containment fails to isolate [Containment isolation is an engineered safety feature provided to close valves in lines that penetrate the containment in order to prevent release from the containment of radioactive gas or particulate materials. The containment isolation system (CIS) is instrumented to function automatically if needed within 10 minutes following accident initiation, or may be otherwise activated manually (Appl. Ex. 67, at 6.2-10, 7.3-1).]

The following four successively more severe classes of CDAs were analyzed by the Staff based upon combinations of the above four primary system response categories and the above three containment response

[75] Finding No. 45.

modes. Bounding estimates of containment release frequencies were then assigned to each CDA Class, as follows:

| CDA Class | Primary System Response | Containment Mode | Release Frequency (Per reactor-year of operation) |
|---|---|---|---|
| 1 | I, II, III or IV | A | $10^{-4}$ |
| 2 | II, III or IV | B | $10^{-6}$ |
| 3 | II or III | C | $10^{-6}$ |
| 4 | IV | C | $10^{-7}$ |

This summary of Staff's appraisal and release frequency estimates indicates that the Staff has in effect found the proposed CRBRP to be in conformance with the Staff's previously established safety objective, stated at the beginning of this CDA discussion.[76]

Staff and Applicants submitted evidence for the CP hearing in support of the thesis that CDAs should not be included in the spectrum of DBAs. That evidence considered numerous matters, the principal ones being the design approach discussed in Section I, above, the testing of design concepts that has been and will be undertaken, the operational experience elsewhere with similar design concepts, the effectiveness of quality assurance (Section III, below), and the anticipated later resolution of generic safety issues.[77]

The Applicants have defined and analyzed four successively more severe CDA cases, identified as Case(s) 1 through 4. The computational methodologies and their validations were also described, along with each of the Case assumptions made. The results are summarized below:

[76] Finding No. 46.
[77] Finding No. 40.

DOSE SUMMARIES FOR THE FOUR CDA CASES
CONSIDERED BY APPLICANTS

| | | REM | | | |
|---|---|---|---|---|---|
| | Organ | Case 1 | Case 2 | Case 3 | Case 4 |
| Exclusion Boundary (2-hour) | Bone Surface | 0.027 | 0.19 | 6.47 | 27.0 |
| | Red Bone Marrow | 0.026 | 0.040 | 0.56 | 2.18 |
| | Liver | 0.052 | 0.060 | 0.44 | 1.21 |
| | Lung | 0.021 | 0.032 | 0.72 | 1.77 |
| | Thyroid | 0.014 | 0.020 | 23.4 | 19.6 |
| | W. Body | 0.81 | 0.82 | 1.09 | 1.21 |

| | | REM | | | |
|---|---|---|---|---|---|
| | Organ | Case 1 | Case 2 | Case 3 | Case 4 |
| Low Population Zone (30-day) | Bone Surface | 0.92 | 0.95 | 2.45 | 6.07 |
| | Red Bone Marrow | 0.19 | 0.19 | 0.27 | 0.56 |
| | Liver | 0.36 | 0.36 | 0.18 | 0.32 |
| | Lung | 1.54 | 1.55 | 0.82 | 1.00 |
| | Thyroid | 85.3 | 85.4 | 8.13 | 5.43 |
| | W. Body | 2.10 | 2.09 | 1.73 | 1.65 |

Applicants present their interpretation of these results and conclude that:

— The design features to mitigate CDA consequences provide an effective means to control the releases for a wide range of conditions; and

— The resulting dose consequences are acceptable.[78]

[78] Finding No. 47.

Our review of Staff's and Applicants' evidence leads us to the opinion that the CRBRP can be designed, constructed and operated in a manner that precludes including CDAs within the envelope of DBAs. Likewise, based upon our review of the Staff's appraisal of CDA frequencies and dose consequences discussed above, we are of the opinion that it is feasible to design, construct and operate the plant in a manner consistent with the safety objective cited at the outset.

These opinions fall short of representing a firm conclusion on our part. There has been a considerable advance in the level of Staff confidence with respect to the CRBRP's ability to meet safety objectives. The Board nevertheless cannot find that the "heavy burden" of technical persuasion we foresaw in our LWA opinion (17 NRC 158, 171 (1983)) has as yet been dispositively borne, owing to the numerous safety considerations not yet dealt with. We do not see this as a lack of diligence or lack of adequate CP hearing preparation. Rather it reflects the unavoidable fact that many safety-related matters would have been left to the operating license review phase for resolution, not unlike the situation facing licensing boards at the CP phase of LWR proceedings.

**D. Intervenors' Challenge to Accident Analyses**

On July 8, 1983 the Natural Resources Defense Council, Inc. and the Sierra Club, who had been joint intervenors during the contested LWA proceeding, filed a document entitled "Limited Appearance Statement of Dr. Thomas B. Cochran Regarding Issues Raised in the Construction Permit Proceeding." This Statement was admitted by the Board during the CP hearing session on August 8, 1983, identified as Board Exhibit 125 (there being no other Board Exhibits) and bound into the transcript of the proceeding (Tr. 7652). The Statement includes four attachments and appears at Tr. 7653-7714; the Statement is also reproduced in its entirety in Appendix E to this memorandum. [However, Attachments are not reproduced in this issuance.] Owing to the potential significance of the matters raised by the Statement, Applicants and Staff addressed the Statement in their prefiled testimony and were questioned by the Board concerning same during the CP hearing. The parties (Applicants and Staff) have concluded that the Statement contains no matters of sufficient import to justify a modification of the evidence they have submitted. We have carefully considered the Statement and the testimony of the parties regarding it and are of the opinion that nothing contained therein alters the opinions and findings of this Board.[79]

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[79] Findings No. 48-53.

## III. QUALITY ASSURANCE

To lend perspective to this discussion of quality assurance,[80] a brief organizational description of the CRBR program is useful. The United States government is the owner of the Clinch River Breeder Reactor Plant (CRBRP); the U.S. Department of Energy (DOE) is custodian and has lead management responsibility, exercised through its CRBRP Project Office (PO) located in Oak Ridge, Tennessee. The Applicants in this proceeding are the DOE, the Tennessee Valley Authority (TVA) and the Project Management Corporation (PMC). TVA will be responsible for plant operation and maintenance. PMC is responsible for administering utility industry interests, providing personnel and disbursing financial support. The PO is staffed by personnel from DOE, TVA and PMC, and is headed by a project director who is an employee of and reports to the DOE. In addition to the entities already mentioned, the following contractor-participants have responsibility under the PO for the design, manufacture and construction of the CRBRP:

- Westinghouse Electric Corporation, Advanced Energy Systems Division (AESD);
- General Electric Corporation, Advanced Reactor Systems Department (GE-ARSD);
- Rockwell International Energy Systems Group, Atomics International Division (ESG-AI);
- Burns and Roe, Incorporated (B&R);
- Stone and Webster Engineering Corporation (SWEC); and
- The DOE (fuel fabrication).[81]

The Applicants have developed a comprehensive quality assurance program that includes quality control and governs the QA activities of contractor-participants responsible for the satisfactory performance of the CRBRP.[82] The QA program was initiated in the early 1970s, and from there it has evolved into the current format. This has provided opportunities for participant familiarization, for learning from QA problems at other plants, and for profiting from the knowledge and experience of the TVA, a co-applicant.[83]

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[80] *Quality assurance* (QA) comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control (QC), which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system to assure adherence to predetermined requirements.

[81] Findings No. 54-57.

[82] Findings No. 54-55.

[83] Findings No. 60, 63.

The QA program has adopted a graded approach to all aspects of the plant's systems and components, irrespective of whether plant safety is involved.[84] A management policies and requirements (MPR) document, maintained by the CRBRP Project Office (PO), establishes the framework for the implementation and coordination of the QA programs of the various participants.[85] The MPR defines the different levels of QA endeavor needed to implement the graded QA approach,[86] and sets forth the administrative controls by which the Project Office manages its overall QA responsibilities.[87]

The Board has reviewed the Applicants' description of all of the QA programs of the various contractor-participants,[88] and finds them to be satisfactory, and dispositive of Board Interest No. 5 concerning the Applicants' implementation of its QA program. The NRC's review of these program descriptions and its audit and surveillance of prior and ongoing QA activities have uncovered no unrectified deficiencies or impediments to its implementation.[89] The NRC Staff has adequately responded to Board Interest No. 6 with respect to Staff surveillance of Applicants' QA program. The NRC's recognition of the importance and complexity of the CRBRP QA effort is exemplified by its assignment of a full-time person to the PO and its engagement of an outside organization to assist with the evaluation of the Applicants' overall design control efforts.[90]

Examples of the diligence of the Project Office include its attention paid to job-site safety, to the independence of all QA program components from adverse cost and schedule impacts, to the review and approval of engineering change proposals, and to the daily maintenance of computerized updating of drawing and specification information, thus assuring that up-to-date as-built drawings will be available.[91]

Based upon its review of the QA program organization and activities of the Applicants and the audit and surveillance of these matters by the Staff, the Board is of the opinion that a workable and working QA program is in place, and that it fulfills the requirements and the intent of Appendix B to 10 C.F.R. Part 50.

A potentially vulnerable feature of the CRBRP quality assurance program lies with the fact that, although it was an already ongoing effort at

[84] Finding No. 61.
[85] Findings No. 59-60.
[86] Finding No. 61.
[87] Findings No. 62-64.
[88] Finding No. 58.
[89] Findings No. 68-71.
[90] Finding No. 72.
[91] Findings No. 65-67. This information further responds to Board Interest No. 5 regarding Applicants' QA program approach.

the time of the CP hearing, it had so far not been required to function in the face of strict funding and/or schedule constraints. Now it is moot as to whether the future would have imposed fiscal and schedule stringencies that would compromise the effectiveness of quality assurance implementation. All that the Board can do now is to advise the Applicants and the NRC of our concern about what might have been a problem.

## IV. EARTH SCIENCES AND ENVIRONMENTAL MATTERS

### A. Geology and Seismology

The geological, seismological and subsurface settings are matters of prime importance to the ultimate safety of a nuclear facility such as the CRBR. The SER for the Clinch River Breeder Reactor includes extensive analyses of the geology and seismology of the CRBR setting and that data was reviewed by the USGS. Its review identified two items of major concern. These were the possibilities of a limestone cavern underlying some portion of the site and of active faulting in the area.[92] However, the Staff and the USGS agreed that the presence of a major undetected cavity beneath a site structure was unlikely based upon the site's geological setting and the extensive studies carried out by the Applicants.[93]

Though to date no active faults have been recognized in the Appalachian region, the USGS was more skeptical concerning active faulting in the area. The CRBR site is located between two thrust faults, these being the Copper Creek Fault and the Whiteoak Mountain Fault. It concluded that, although the Applicants had not mapped or trenched across these faults, it is reasonable to conclude that these faults, like others in the region, are noncapable.[94]

The USGS noted there may be a concentration of seismicity in eastern Tennessee based upon recent studies which have relocated earthquakes in the eastern U.S. Though there is insufficient evidence to identify a specific structure, they raise the possibility of a local seismic source. The USGS presented evidence both for and against the hypothetical structure but found that at present the data are insufficient to establish whether

[92] Findings No. 74-76.
[93] Finding No. 77.
[94] Findings No. 78-79.

the hypothetical local source exists. The USGS concluded that the CRBR SSE is reasonable based on present data.[95]

The Staff independently reviewed the geology and seismology relating to CRBR and took cognizance of the USGS concerns. It concludes that the faults at the site and in the region around the site are not capable, but certain confirmatory investigations should be carried out. These investigations should go forward but need not be completed before the issuance of the CP.[96]

In its consideration of the seismology of the area, the Staff indicated that it was well aware of the postulated seismogenic source zone. Moreover the Staff found that most of the evidence relating to the hypothetical structure is equivocal or negative, and it took the position that the evidence for the hypothetical local seismogenic source zone was so weak that it does not warrant consideration as a capable fault. The Board agrees with this position.[97]

The USGS had suggested that it would require a definitive seismological investigation to address the question of a possible concentrated seismic source in eastern Tennessee. Such an investigation would require a local network, velocity models and source mechanism determinations.[98] The Staff noted that there are additional studies under way and there is a well-distributed network of seismographic stations in the CRBR region and more stations are planned for the future. If, in the future, a sufficient number of well-located earthquakes occur in the area, a definitive study, such as suggested by USGS, would be possible. The Applicants would have been required to report any seismological developments in the area in the Final Safety Analysis Report. Nothing more is possible or required at this time.[99]

After the close of the CP hearing in Oak Ridge in August, the Board became aware of the discovery of certain faults on the site. The Board then issued an Order Requiring Disclosure of Seismic Information (September 15, 1983 (unpublished)). The Applicants made a full response shortly after the Order was issued. That response contained two affidavits.[100] The affidavits established firmly that the faults observed in the CRBR excavations were reported in accordance with the commitments stated in the PSAR and SER, were expected to be found based upon prior geological investigations at the site and in the region, and are

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[95] Findings No. 80-82.
[96] Findings No. 83-86.
[97] Findings No. 87-89.
[98] Finding No. 82.
[99] Finding No. 90.
[100] Appl. Ex. 98 and 99.

not capable within the meaning of 10 C.F.R. Part 100. The presence of the faults does not affect any geological or seismological conclusion in the SER.[101]

We have concluded that the investigation of the geology and seismology of the CRBR site and region by the Applicants and Staff has been thorough and that those studies and the data derived from them meet the applicable regulations. We know of no outstanding facts or questions concerning the geology and seismology of the site or region that would alter our Opinion and Findings.[102]

## B. Emergency Planning

We have studied the site of the CRBR and considered its location with reference to the surrounding communities, cities, counties and states,[103] and we had concluded earlier that an effectively coordinated site, state and local radiological emergency response plan can be achieved for the CRBRP.[104] We now have more information which includes the Staff's review of the Clinch River Breeder Reactor Radiological Emergency Plan. The Staff reviewed the Plan against the applicable regulations and concluded that the planning standards will be met in the final emergency plan. The Staff also concluded there were no special or unique circumstances that would preclude the development of an adequate preparedness plan at the operating license stage of review and that the plans are in conformance with the TMI Action Plan. We have found no bases upon which to disagree with these conclusions.[105]

We noted in our Partial Initial Decision (LWA) that we would explore the emergency responses of the three major DOE facilities at Oak Ridge in the event of an emergency at CRBR as part of the CP hearings.[106] The three facilities are the Oak Ridge Gaseous Diffusion Plant, the Oak Ridge National Laboratory and the Y-12 Plant. Each of these facilities has long-standing emergency plans of its own, and each has extensive emergency planning, preparedness and response experience. The response options of sheltering or evacuation in an emergency are available and practical and can be readily accomplished for each. There are no im-

101 Findings No. 91-99.
102 Findings No. 73-100.
103 Finding No. 101.
104 17 NRC 158, 243; Finding No. 6.
105 Findings No. 102-03.
106 17 NRC 158, 203; Finding No. 52.

pediments to assuring compatibility with CRBRP emergency planning needs.[107]

During the Hearing as a result of Board questions we learned that the emergency evacuation time estimates used in the CRBRP emergency plan were derived from site-specific considerations and are based upon standard and conservative procedures for deriving those estimates. We could identify no unusual condition, such as loss of a bridge or severe weather, which would so impede evacuation as to make it impractical.[108]

Before the CP hearing, the Board had posed inquiry Item No. 7 to the Applicants.[109] We have learned from the Applicants that commercial and recreational river traffic within the 10-mile EPZ will be controlled by the Tennessee Wildlife Resources Agency, assisted as necessary by the U.S. Coast Guard, during periods of off-normal plant operation. Passage of vessels through the upstream lock at Melton Hill Dam will be controlled by the U.S. Army Corps of Engineers. Appropriate arrangements will also be made to control traffic and provide prompt warning and removal of persons within the exclusion boundary on the Clinch River. Implementation criteria for these controls will be described in the CRBRP and State of Tennessee Radiological Emergency Plan.[110]

No hazardous materials have been transported by barge past the site in the past, and none are anticipated in the future. Should commodities being shipped on the river change in the future, new barge facilities would be required. Since TVA would be involved in the issuance of any new permit for barge facilities, it seems highly unlikely that any permit would be issued that would allow the shipment on the river past the site of materials hazardous to the well-being of the CRBR.[111]

In our Partial Initial Decision (LWA), we noted that, since the fuel to be used in the CRBR will have significantly different isotopic composition than other licensed reactors, accidental releases then will be made up of concentrations of isotopes which are unique to the CRBRP type of reactor. Of particular interest to the Board was the fact that both the Staff's and Applicants' witnesses had testified that there were no Protective Action Guides (PAGs) for bone surface dose, and the Staff's witnesses further testified that this dose could be controlling.[112] Such doses would originate primarily from alpha particle emitters, such as plutonium, which originate in the reactor fuel. We instructed the parties

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[107] Findings No. 104-07.
[108] Finding No. 108.
[109] *See* Appendix D, Item 7.
[110] Findings No. 110-11.
[111] Findings No. 112-13.
[112] 17 NRC 158, 174.

to this proceeding to address the question of whether the PAGs currently in use for evacuation planning purposes should be revised for use at CRBR to take account of those possible releases unique to CRBR.[113] Furthermore, we specifically requested the Staff to respond to this question.[114]

In response the Applicants filed Exhibit 94 and the Staff filed its Exhibit 43.[115] By the analyses of the radiological consequences of HCDAs and by deriving ranges of PAGs for other organs, including bone surface doses, from ICRP-26 tissue weighting factors, the Applicants found the whole-body and thyroid doses to be controlling. That is, the bone surface doses from plutonium and other actinide elements are not controlling.[116] The Staff, using similar but more conservative analyses, likewise concluded that the EPA PAGs were adequate for emergency planning purposes and that bone surface doses are not expected to be controlling for evacuation purposes in the event of an HCDA at CRBR.[117]

We have concluded that there are no insurmountable impediments to effective planning or the development of a fully satisfactory emergency plan for CRBRP.

### C. Environmental Matters

The Board considered the suitability of the proposed CRBRP site and environmental matters related to CRBRP in its Partial Initial Decision (LWA).[118] We found the CRBRP suitably sited and environmentally acceptable and we have no basis for changing that conclusion.[119]

## Findings of Fact

### I. DESIGN APPROACH

1. The NRC Staff (Staff) has established the objective that the CRBRP must achieve a level of safety comparable to that for light water reactors. Major emphasis must also be placed upon features and capabilities to assure that there is a low likelihood of containment failure asso-

[113] 17 NRC 158, 174-75.
[114] *See* Appendix D, Item 9.
[115] Finding No. 114.
[116] Finding No. 115.
[117] Finding No. 116.
[118] 17 NRC 158 (1983).
[119] Finding No. 117.

ciated with disruptive core melt accidents (Staff Ex. 5, at 1-2; Staff Ex. 32, at 14; Tr. 8049; Appl. Ex. 89, at 2-3; Tr. 7764-65).

2. The General Design Criteria (GDCs) for LWRs are considered to be applicable to other types of nuclear power plants; they serve as guidance in developing Principal Design Criteria (PDCs) for a new type of plant such as the CRBRP. Accordingly, the Staff developed as its general design requirements a set of sixty PDCs specific to the CRBRP with which compliance is required. The PDCs require sufficient redundancy, diversity and independence in safety systems so that the opportunity for the basic safety systems to permit an accident to progress to core melt and disruption is minimized. A detailed description of these PDCs along with their differences from and similarities to the LWR GDCs is presented in the SER § 3.1, "Principal Design Criteria" (Staff Ex. 32, at 16-18; Tr. 8051-54; Staff Ex. 26, at 3-7 through 3-34).

3. One or both of two basic conditions must exist in order for an accident to be initiated: reduced heat removal, and/or excessive heat generation. Absent the reestablishment of a balance between heat generation and heat removal, an accident can progress in severity to the point of melting and disruption of the core (Appl. Ex. 1, at 14-15; Tr. 2003-04; Appl. Ex. 87, at 4-5; Tr. 7381-82).

4. For the purpose of shutting down the fission chain reaction there are two independent and diverse reactor shutdown systems (RSSs), designated as primary and secondary systems. Each system detects a variety of plant operating parameters in order to determine the need for automatic insertion of control elements capable of terminating the fission reaction, designated as a reactor trip or scram. The primary RSS uses three redundant and physically separate instrument channels for each measured parameter. The three channels are used in a two-out-of-three coincidence logic to generate reactor trip signals. Three redundant logic trains are provided. There are five scram breakers of the primary RSS, arranged in a manner such that trip signals from two of three logic trains will open a sufficient number of the scram breakers to interrupt power to the primary RSS control rods. Interruption of power to the control rods causes the rods to be inserted into the core. The secondary RSS uses types of equipment different from that in the primary RSS and, in general, monitors a different set of parameters from those monitored by the primary RSS (neutron flux, however, is monitored by both RSSs). Neutron flux is sensed with compensated ionization chambers in the primary RSS, and with fission chambers in the secondary RSS. Three redundant and physically separate instrument channels are used to sense each measured parameter. Three redundant logic trains are used in the secondary RSS such that two out of three trip

demand signals will result in insertion of the secondary RSS control rods. The secondary RSS control rods are tripped by venting pneumatic pressure which releases a latch on each control rod. The pressure is vented by scram solenoid valves actuated by the secondary RSS in a two-out-of-three configuration. Scram breakers are not utilized. As with the primary RSS, the secondary RSS control rods will scram on loss of power. Since both of the RSSs consist of three redundant channels and three redundant sets of logic, each system, by itself, is capable of performing the safety function of shutting down the reactor even if a single failure has occurred within that system. The shutdown system designs include provisions such as the use of physical separation and isolation devices to ensure that malfunctions in a channel or set of logic of one shutdown system cannot propagate to a redundant channel or set of logic of the same shutdown system, or to any channel or set of logic of the second shutdown system. All RSS equipment required to shut down the reactor is designed to remain functional following either an operating basis earthquake or a safe shutdown earthquake. The Staff's evaluation of these systems is documented in Volume 1 of its SER (Staff Ex. 32, at 22-23; Tr. 8057-58; Staff Ex. 26, at 7-5 through 7-16).

5. Operational and shutdown heat removal modes are described. The CRBRP heat removal systems consist of the main heat transport system (HTS) and the shutdown heat removal system (SHRS). The main HTS is comprised of three identical heat transport loops used to carry heat from the reactor core through a primary loop, isolated from an intermediate loop by means of an intermediate heat exchanger (IHX). The heat transported by the intermediate loop generates steam in two identical evaporator modules. The generated steam is passed through and superheated in a third module. The superheated steam then passes to a turbine generator to generate electricity, and the waste heat is rejected to the atmosphere. A pump in each primary and intermediate loop provides motive power to circulate the coolant. These three main HTS loops are designed to remove the full-power heat generation of the core.

The SHRS consists of subsystems utilized for removing decay heat after the reactor has been shut down. The SHRS consists of the three main HTS loops, plus a diverse heat removal system called the direct heat removal service (DHRS). Decay heat is normally removed through the main heat transport system (HTS), steam, condenser, and feedwater systems. Each HTS loop is also provided with a safety-grade backup decay heat removal system called the steam generator auxiliary heat removal system (SGAHRS). The SGAHRS utilizes steam vent valves, and a steam-to-air heat exchanger to dump heat to the atmosphere.

Feedwater is supplied by a safety-grade auxiliary feedwater system similar to that utilized in LWRs. These systems normally use electric power supplied by offsite or safety-grade (1E) onsite power supplies. However, no offsite or onsite power (other than batteries) is required for decay heat removal through the SGAHRS. This can be accomplished via natural convection in the sodium loops, and via steam venting and the steam-to-air heat exchanger in the SGAHRS. The natural circulation capability has been verified by Applicants' and Staff's analyses and confirmed by tests at FFTF and EBR-II, thus satisfying the Board's inquiry in Item 4 of Appendix D. If for any reason, all three HTS loops are lost or unuseable beyond the IHX, operation of the DHRS can be initiated, utilizing the reactor overflow path through a heat exchanger to reject the decay heat through the air coolers used to cool the ex-vessel storage tank (EVST). The DHRS requires AC power (either from offsite or onsite sources) and can accomplish its function even with a single failure of any active component. Results of the Staff's review of these systems are presented in Volume 1 of its SER (Staff Ex. 32, at 25-27; Tr. 8060-62; Appl. Ex. 87, at 98; Tr. 7475; Tr. 7631; Staff Ex. 37, at 5; Tr. 8196; Staff Ex. 26, Sections 4, 5, 7 and 15 *passim*).

6. Rupture prevention of the primary heat transport system piping that delivers sodium coolant to the CRBR relies upon four operative considerations described below.

1. The piping is subjected to high-quality engineering standards specified for design, analysis, materials, fabrication, examination, and testing. The principal standard applied is the ASME Code, Section III, Class 1 which is the highest-quality national consensus standard for nuclear piping. The Code rules are supplemented by Code Cases and RDT standards* to account for elevated-temperature sodium service. The Nuclear Systems Materials Handbook provides material properties data not available from the Code. Rigorous quality requirements are specified for base and weld filler materials, finished pipe and piping subassemblies. Compliance with these requirements is verified in each step of fabrication and construction. A comprehensive quality assurance program ensures that the specified standards are met. There is little potential for initial flaws in the piping. A comprehensive inservice inspection program will assure that there is little potential for initiating flaws during Plant life. Implementation of these standards is discussed in

*RDT standards are those that have been developed by DOE and its predecessors to serve where the ASME Code does not include all considerations necessary for LMFBR design.

detail in WARD-D-0185 (Appl. Ex. 87, at 122-23; Tr. 7499-7500; Appl. Ex. 88, Sections 3.1, 4.1).

2. A detailed fracture mechanics evaluation has shown that, even if a large initial defect were to exist, the toughness of the piping material prevents significant growth of the defect. A survey of piping fabrication and installation experience established that a defect 1.5 inches long and 0.125 inch deep (one-quarter times the wall thickness) would be the largest defect to escape detection due to failure of the quality assurance program. Growth of the defect during Plant life was predicted using linear elastic fracture mechanics. The defect was assumed to be located in the most highly stressed elbow of the PHTS cold-leg piping. (Duty cycle and seismic events were considered. The duty cycle is a listing of normal and off-normal events postulated to occur over the 30-year Plant life. The expected number of occurrences of each event is specified. Equipment pressure and temperature loadings are conservatively calculated using an appropriate Plant system model for each event.) An extensive fracture mechanics data base was reviewed to assure consideration of the effects of temperature, internal environment, frequency, stress ratio, aging, cold work, crack orientation, heat-to-heat variations, grain size, irradiation, biaxial stress, loading waveform, weld material, static loading, and external environment. Using the fracture mechanics characteristics supported by the data base, crack growth of no more than 18 mils over the Plant life was predicted. Growth of the initial defect by 18 mils would not result in a crack approaching critical crack length** and would not result in a leak (over one-half of the pipe wall thickness would be intact). Thus, even if a significant flaw is undetected, the Plant duty cycle would not result in any leakage from the PHTS pipe. Details of this evaluation are reported in WARD-D-0185 (Appl. Ex. 87, at 124-26; Tr. 7501-03; Appl. Ex. 88, Section 4.3).
3. A comprehensive technology program has shown that even if a crack did grow significantly, it would penetrate the pipe and be detected as a small leak (by the leak detection system) prior to developing potential for a large pipe break. Tests and analytical studies have shown that crack growth will drive a defect

**That crack length above which concern about catastrophic rupture becomes important.

through the wall thickness without substantial growth in length (along the pipe wall). Test articles included piping elbows similar to those in the highest stressed regions of the PHTS cold leg. The tests show that even for very low cyclic stress ranges (where the ratio of through-wall growth to growth in length is smallest), wall penetration would occur when the crack length is very small compared to the critical crack length for the PHTS cold leg — 30 inches. Disregarding the evaluation which showed no significant crack growth, if a defect were to grow, it would penetrate the pipe wall and be detected as a small leak prior to approaching critical crack length. The leak detection system is capable of detecting a 100-gram-per-hour leak rate which is much smaller than the leak rate for which the Plant would be shut down. Details of this evaluation are reported in WARD-D-0185 (Appl. Ex. 87, at 126-27; Tr. 7503-04; Appl. Ex. 88, Section 4.3).

4. Analysis and testing have demonstrated that, even if a small leak is not detected and corrective action is not taken, toughness and ductility of the stainless steel pipe along with the low coolant operating pressure would limit the maximum crack length. The evaluation hypothesized the existence of an initial defect of 3 inches (twice as long as that established for the fracture mechanics evaluation discussed in ¶ 2, above). Contrary to the evidence discussed in ¶ 3, above, crack growth through the pipe was constrained so that penetration of the pipe wall did not occur. Crack growth was evaluated for the entire Plant life. The crack growth model used in this evaluation is supported by data from tests of plates with large initial defects subjected to bending stresses. The predicted length of the crack at end of Plant life is 5.4 inches. Abrupt penetration of the pipe wall is assumed to occur at end of Plant life. Due to the ductility of the stainless steel pipe material and the low internal pressure, the crack would not penetrate the wall over its entire length. Instead, the penetration would end where the remaining wall ligament is able to withstand the imposed stress. Penetration would occur only in the central 4-inch portion of the crack. This 4-inch crack is small compared with the critical crack length for the PHTS cold leg — 30 inches. Therefore, disregarding the first three levels of protection discussed above, no defect in the PHTS cold leg would grow to a length which

could cause a double-ended pipe rupture. Details of this evaluation are reported in WARD-D-0185 (Appl. Ex. 87, at 127-28; Tr. 7504-05; Appl. Ex. 88, Section 6).

7. The overall conclusion that the likelihood of a double-ended pipe rupture is low is supported by worldwide experience with sodium systems. Ninety-five percent of known leaks involved a total spill of 10 pounds or less. Almost half of these leaks occurred in valve bellows with no spill into the cell. Many of the remaining leakage events involved types of equipment and circumstances that are not relevant to CRBRP. Thirty leaks have occurred in piping. Most of these leaks occurred in small-diameter piping which was not designed and operated with the high standards applied to CRBRP. All leaks in sodium piping have developed as small cracks. Even without the sensitive leak detection capability provided in CRBRP, none of the leaks experienced in sodium piping have approached the magnitude of a double-ended pipe rupture (Appl. Ex. 87, at 129-30; Tr. 7506-07).

8. Accident initiation can result from local imbalances between heat generation and removal within an individual fuel pin or bundle of pins. Such imbalances could result from two causes: mispositioning of a fuel bundle (or core subassembly) in a location where it would receive inadequate coolant flow; or blockage of coolant flow to an individual subassembly. Means to prevent these occurrences are incorporated into the CRBR design.

A mechanical design integration approach has been adopted that coordinates each core subassembly structure with the lower coolant inlet module into which it uniquely fits, so that no subassembly will fit into a wrong inlet module. If an incorrect insertion is attempted, its misfit will prevent full insertion and also prevent release of the subassembly by the fuel-handling transfer mechanism. Manual and computerized inventory systems track each subassembly that is moved into and out of the reactor, to guard further against improper positioning. Source range flux monitors (SRFMs) are provided to monitor the subcritical state of the reactor while it is shut down. If, during the refueling, the change in count rate measured by any one of the three SRFMs exceeds the expected change, refueling action will be halted immediately and the anomalies resolved before proceeding. For example, the inadvertent removal of a control assembly (rather than a fuel subassembly) will be readily discernible by the SRFMs (Appl. Ex. 87, at 132-35; Tr. 7509-12).

9. Not only is the opportunity for flow blockage to any single fuel subassembly minimized, but if blockage does occur and causes a localized failure there are features to ensure that such failures do not propagate to other regions of the core. A blockage or reduction of flow to a

core subassembly is precluded by a multiplicity of redundant flow paths in the core support structure that supports the inlet modules, in the inlet modules that hold groups of subassemblies, and in each core subassembly inlet. These redundant and diverse flow paths to each core subassembly assure that no object or accumulation of foreign material could block enough passages to starve the flow to any subassembly. Extensive testing has been performed to confirm this concept. The tests were performed using water in a ¼ scale model of the inlet plenum. One test series considered the hypothetical blockage of all primary ports in seven modules. It was found that the flow was reduced by only 6%. This verifies the effectiveness of the debris barrier and auxiliary ports. Another test series determined the inlet plenum characteristics with regard to blockage by solid particles. The design was proven to be effective for eliminating blockage potential while maintaining adequate flow.

Although substantial quantities of particles are not anticipated in the primary sodium coolant, the flow paths are arranged so that smaller and smaller particles would be successively removed from the flow stream as it approaches the fuel rod bundles. Only very small particles (less than 0.25-inch diameter) could pass through the lower inlet module strainer and enter the subassembly inlet. Particles between 0.25 inch and about 0.10 inch in diameter would be trapped at the fuel rod bundle attachment rails. Particles between about 0.10-inch diameter and 0.06-inch diameter would be trapped in the rod bundle inlet region which is a negligible heat-generating region. Particles smaller than about 0.06-inch diameter that pass through the fuel rod bundle attachment rails and rod bundle inlet passages are able to pass through the coolant paths between the fuel rods.

Even if a major buildup of particles at the attachment rails or rod bundle inlet is assumed to block more than 50% of the flow area, the subassembly outlet temperature would not increase by more than 20°F. This modest temperature increase would not significantly reduce the large margin to coolant boiling. [Nominal full-power coolant outlet temperature is 995°F; at atmospheric pressure, the Na coolant boils at 1621°F.] Thus, the design has margin to accommodate a substantial blockage, in addition to the provisions to prevent such blockages (Appl. Ex. 87, at 135-39; Tr. 7512-16; Appl. Ex. 72, Section 15.4.1.3).

10. Extensive analyses supported by experimental data show that local fuel rod failures would not propagate beyond their immediate vicinity. Inherent protection to prevent such propagation from one subassembly to a second subassembly is provided by the steel hexagonal subassembly duct that encloses each fuel rod bundle and by the channel

of sodium between subassemblies. The fuel rod cladding and subassembly ducts act as barriers to prevent fuel failure propagation. The theoretical challenges, in addition to blockage previously discussed, to the cladding and ducts are fission gas release and molten fuel release. Operating experience with sodium-cooled reactors shows that fission gas release does not cause fuel rod failure propagation. Many thousands of mixed-fuel rods have been irradiated in power reactors and test facilities. Fuel rod failures resulting in fission gas release have occurred under conditions similar to those for CRBRP operation. No failure propagation beyond the immediate vicinity has occurred. The fuel failure data base originates primarily from testing and operation in EBR-II. That program has demonstrated that fuel failure is a benign event and additional failures do not occur as a consequence of fission gas release. Fission gas releases have occurred over a wide range of power, cladding temperature, and burnup. Fuels rods with internal gas pressures as high as 1700 psig have failed in testing. Corresponding CRBRP fuel rod pressures are less than 1000 psig. The fission gas release has been gradual, and no detrimental effects on neighboring fuel rods or the subassembly duct have been observed. Analyses are presented in the PSAR for limiting, conservative conditions for the CRBRP core. They include the effects of fission gas release and gas blanketing of adjacent rods due to jet impingement of the leaked gas, flow reduction and reversal due to fission gas release and the mechanical effects of pressure pulses associated with the postulated rapid release of the rod plenum gas. The calculations, which are supported by out-of-reactor proof tests discussed in the PSAR, show that even if all 217 rods in a limiting assembly were to fail, failure propagation due to fission gas release would not occur. These studies show that the subassembly ducts are not structurally challenged and act as an effective barrier to preclude propagation beyond the affected assembly.

Release of molten fuel from a fuel rod would require that the rod operate at a temperature significantly higher than planned and could only be postulated for fuel pellets containing higher-than-design enrichment. Since the CRBRP nuclear design is based on only one enrichment, molten fuel resulting from mixing enrichments will not occur. However, limiting analyses have been performed to demonstrate that molten fuel release will not result in local failure propagation. Data from numerous tests discussed in the PSAR show that molten fuel injected into sodium will fragment into small particles in the range of 0.004 inch to 0.040 inch in diameter. These particles are readily swept out of the subassembly by the sodium coolant. Test data also show that heat exchange between the molten fuel and sodium is very inefficient, limiting the vaporization of sodium and its effect on coolant flow rate. In tests simulating conditions more severe than any postulated to occur in CRBRP, extensive fuel melting was achieved in most of the fuel rods. Expulsion of some of the molten fuel from these rods resulted in a pressure of only 150 psig with a duration of a few milliseconds. These tests are discussed in the PSAR. Sodium voiding from events of this small magnitude would not have any significant effect on neighboring fuel rod cladding because the associated reactivity change is negligible and the wire-wrap design promotes inter-subchannel cross-flow. Even if it occurred, direct jetting of molten fuel onto cladding of an adjacent fuel rod could lead to its failure and release of fission gas, but further propagation would not occur. If it is postulated that the molten fuel does pass through the coolant and impinge upon the subassembly duct, local overheating and localized melting may occur. The high heat removal capacity of sodium flowing through the neighboring duct and sodium flow between ducts, however, would prevent any damage to the neighboring duct. (Appl. Ex. 87, at 143-47; Tr. 7520-24; Appl. Ex. 72, Section 15.4).

11. Applicants have imposed overall design requirements on the CRBRP fuel to deal with four levels of reactor conditions: (1) normal operation, (2) anticipated transients, (3) unlikely transients, and (4) extremely unlikely transients. Design limits and acceptance guidelines for the fuel for each level of reactor conditions were established to facilitate demonstration that the CRBRP fuel design requirements will be met. Because the structural capability of the cladding is a function of its ductility-limited strain and cumulative damage function, the fuel performance predictions took into account operating conditions such as temperature and pressure and damage mechanisms such as creep, irradiation effects and fatigue damage. The application of these design requirements, design limits and acceptance guidelines are intended to ensure that even in the case of extremely unlikely transients, the fuel will be maintained in a coolable configuration. Analyses of operating conditions and cladding properties as well as a large experimental data base from tests conducted at the EBR-II and TREAT reactors demonstrate that the overall design requirements will be met by the CRBRP fuel. Additional data and experience will be obtained from FFTF before operation of the CRBRP. Moreover, during operation of CRBRP, the reactor mixed mean inlet and outlet temperature, and reactor power, as well as outlet coolant temperature of most of the fuel and blanket assemblies, will be monitored to estimate cladding temperatures and to permit an early prediction of the capability of the fuel (Appl. Ex. 87, at 183-86; Tr. 7560-63).

12. The present CRBRP containment system concept involves a welded steel containment shell surrounded by a reinforced concrete confinement building. A 5-foot air-filled annulus separates the two structures. The annulus is maintained at slightly reduced pressure relative to the containment, so that out-leakage from the containment shell will be collected in the annulus. There it is circulated, filtered, and partially released to the atmosphere to maintain reduced pressure; the balance is returned to the annulus. The steel containment shell is designed for a leak rate of 0.1% (of volume) per day at a design pressure of 10 psi above atmosphere. Leakage that bypasses the annulus filtration system is to be held to no more than 0.001% of containment volume per day at design pressure. These specifications regarding pressures and leaktightness are within the feasibility of current practice. There is experience with other sodium-cooled reactors in building containments designed to withstand sodium fires (Appl. Ex. 1, at 50-51; Tr. 2039-40; Staff Ex. 3, at 22-25; Tr. 2505-08).

13. The potential challenges to containment integrity that might result from core melt accidents yielding an energetic release are of two basic types: (1) damage from internal missiles and (2) damage from pressure-temperature excursions. The internal missile challenge could result from excessive loadings on the reactor vessel head that exceed the structural capability of the head or the components mounted on the head. The pressure-temperature challenge could result from either excessive energy addition to the containment from sources such as burning of sodium and hydrogen and fission product decay heat, or from excessive formation of gases not readily condensable such as hydrogen. Two basic types of features and capabilities (design margins) are provided to address these challenges: (1) Structural Margin Beyond the Design Base (SMBDB) and (2) Thermal Margin Beyond the Design Base (TMBDB). The SMBDB addresses short-term (minutes or less) challenges to containment integrity, while the TMBDB addresses longer-term (hours to months) challenges to containment integrity.

The potential short-term challenges to containment integrity could derive from:

a) overpressurization of containment as a result of a very large prompt sodium release from the reactor coolant boundary into the containment (through the reactor vessel head) and the rapid burning of that sodium; or
b) internal missiles from the reactor vessel head area with sufficient energy to penetrate the containment.

Neither of these conditions could arise unless an accident occurred with sufficient release of mechanical work energy (energetics) to exceed the structural capability of the reactor coolant boundary. Specific requirements have been placed on the reactor coolant boundary to avoid such events. It should be noted that, to a large extent, the accommodation of these requirements to maintain short-term containment integrity does not require additional specific design features, but rather additional capabilities of the existing features. For example, the reactor vessel head and vessel support systems have been strengthened to accommodate the SMBDB dynamic loads. In certain cases, however, additional features have been added; for example, seals have been added to the reactor vessel head to meet the leakage requirements (Appl. Ex. 89, at 3-5, 9; Tr. 7765-67, 7771).

14. The potential long-term challenges to containment integrity would derive from:

a) overpressurization or overheating of the containment due to heat addition from the burning of sodium and hydrogen, fission product decay heat and the sensible heat from the sodium vapor; or

b) overpressurization by not readily condensable gases, principally hydrogen, if the hydrogen does not burn.

Such challenges to long-term containment integrity are avoided by design features that were referred to above as TMBDB features. These features provide capability to remove heat from the containment, and to vent and purge the containment, thereby avoiding challenges from overpressurization and excessive heat. Any vented products would be released through a cleanup system that would remove a large fraction of particulate materials. With these features, control would be maintained over the releases to the environment. Venting would not be required until about a day or more after an accident, thus allowing adequate time for interdictive measures to further reduce the accident consequences. Applicants have described the features and requirements necessary to maintain long-term containment integrity and to mitigate any radiological releases from the containment (Appl. Ex. 89, at 10-17; Tr. 7772-79).

15. The NRC requires the performance of a plant/site-specific probabilistic risk assessment (PRA) for certain specific LWRs and a manufacturing facility, as part of its TMI-related requirements. The NRC Staff has made the performance of a PRA a requirement for the CRBRP in order to provide added assurance that the risks from operation of the CRBRP will be equivalent to those from LWRs. Succinctly, the aim of the PRA is to seek such improvements in the reliability and safety of core and containment heat removals systems as are significant and practical, while not negatively impacting the plant excessively. The PRA can also aid in identifying specific preventive and mitigative actions that

might further reduce risks. Section 50.34 of 10 C.F.R. requires that the PRA be completed within 2 years following the issuance of a construction permit (CP), and that its results be factored into the final design of the plant. The instant PRA effort was initiated in mid-1981 and is scheduled for completion in December 1984. The Staff finds that the work completed to date and the scope of the Applicants' commitment for the remaining effort are adequate to meet the NRC's requirements (Staff Ex. 32, at 45-47; Tr. 8080-82. Staff Ex. 27, at D-1 through D-5).

16. The Staff has required the Applicants to undertake a Reliability Assurance Program (RAP) to provide additional conservatism in the face of limited LMFBR operating experience. The objective of the RAP is to provide additional assurance that the inherent reliability in the CRBRP design concept is achieved and that the likelihood of exceeding the offsite radiological dose guidelines of 10 C.F.R. Part 100 is acceptably low. The RAP is to be performed by the Applicants throughout the life of the Plant. Portions of it are currently under way. Although the RAP is not a formal part of the quality assurance (QA) program, the Applicants, in response to the Board's inquiry [Appendix D, Item 15] described the interactions that occur between the two programs (Staff Ex. 32, at 50-52; Tr. 8085-87; Staff Ex. 27, at C-1 through C-10; Appl. Ex. 87, at 168-69; Tr. 7545-46).

17. Several items have been identified in the Staff's SER that will require review at the OL stage. Those that have potential for impacting safety, cost or schedule have been described by the Staff as falling within the following five areas:

1) Fuel design limits, analysis methodologies and bases;
2) High-temperature design limits and analysis methodology;
3) Reactor vessel closure head structural capability;
4) PRA and RAP analyses; and
5) Natural circulation capability.

The Staff has offered the opinion (and its reasons for same) that there is a low likelihood for any of these items to result in a significant impact on cost or schedule. The Staff will not accept a confirmation or resolution of any item at the OL review stage that it considers will result in a compromise of safety. The Staff and the Applicants are developing a program and schedule for the review and resolution of each item in a manner that will minimize design and construction impacts (Staff Ex. 38, at 4-6; Tr. 8211-13).

18. The Staff has concerns about certain analytical methods and assumptions employed by Applicants in connection with the design of the fuel. Because of this, the Staff has identified four operational fallback positions that could be implemented if future analytical and experimental

data fail to substantiate (during the OL review phase) the Applicants' proposed design. These fallback measures are:

1) Reduction of exposure (or burnup) objective;
2) Reduction of peak power;
3) Reduced operating temperature; and
4) Adjustment of trip points on Plant protective systems.

(Staff Ex. 26, at 4-47, 4-48; Staff Ex. 39, at 3; Tr. 8225.)

Based upon the Applicants' commitment to address the Staff's concerns through the conduct of experimental and analytical programs, and based upon existing experimental results, the Staff concludes that it is unlikely that these fallback measures will need to be implemented. If there is need for implementation at the beginning of operation, it is unlikely that they will have to be imposed for the life of the Plant since design changes can be made on future reload fuel. The Staff has addressed the possible impacts upon design objectives that might derive from implementation of these fallback measures. No substantive adverse impacts were identified (Staff Ex. 39, at 2-7 plus enclosures; Tr. 8224-47).

19. The CRBR is designed with the intention of operating with failed fuel. Applicants anticipate operation with not more than 1% of the fuel being failed. [The term failed fuel in this context refers to fuel pins whose physical strength is not impaired but whose cladding permits the outleakage of gaseous fission products.] Because the same leakage path that permits outleakage of fission product gas may also permit inleakage of sodium coolant, there is potential concern that such a mechanism may adversely impact fuel pin performance and hence operational safety. Staff and Applicants have considered this and have concluded that two considerations will obviate this concern:

1) Two different types of leak detection systems are employed that will permit the detection of fission gas leakage and the detection of sodium that has made contact with fuel, thus identifying the existence of failed fuel;
2) Applicants have an experimental program under way at EBR-II to determine the consequences of operation with failed fuel; pending the results of these tests, Staff and Applicants have agreed to an operational restriction that will ensure the timely removal of such failed fuel.

(Staff Ex. 26, at 4-20, 4-21; Tr. 7637; Staff Ex. 40, at 2-3; Tr. 8249-50; Appl. Ex. 87, at 152-53; Tr. 7529-30; *id.* at 216-17; Tr. 7593-94.)

20. The heat transport system for the CRBRP is made up of three heat transport loops, each of which employs three identical tube and

shell heat exchanger modules. Two of these serve in parallel as evaporators (steam generators) and the third is used as a superheater (Appl. Ex. 87, at 14; Tr. 7391. Appl. Ex. 66, Fig. 5.1-1 at 5.1-16). The module design is configured in a manner that can accommodate relative movement between tubes and shell due to differential thermal forces without interference or excessive stress. This design also provides features to minimize flow-induced tube vibrations within the module. Both steady-state and transient conditions of temperature, pressure and mechanical forces have been anticipated in the design, which reflects ASME Code, Code Cases and RDT standards. The most severe thermal transient that could be mechanistically postulated was analyzed and found not to compromise design integrity. In addition, rupture disks will be provided to protect against overpressure. Finally, a comprehensive component test program will be undertaken to verify performance prior to the fabrication of the CRBRP modules. Additional design and analysis details are given in PSAR Section 5.5 (Appl. Ex. 87, at 187-95; Tr. 7564-72).

21. Applicants' containment design and its accident accommodation capabilities have been assessed and independently reviewed and critiqued by the Staff. Some aspects must await publication of Applicants' FSAR and Staff's OL review for final Staff approval (Staff Ex. 26, at 6-1 through 6-9; Staff Ex. 27, Appendix A at A.1-1 through A.6 *passim*).

Despite this lack of finality of the Staff's CP stage approval, all major components of the containment and their associated ESFs have been determined by the Staff to be appropriate and adequate for the CP phase of licensing review. This determination included radiological offsite dose consequences of severe accidents (*ibid.*; Tr. 8518-30).

Finally, ten Staff witnesses, all of whom had participated in various aspects of the Staff's containment adequacy determination, independently and affirmatively testified that they are satisfied, with respect to containment adequacy, that a construction permit should issue (Tr. 8529-30).

22. *First Level of Design — Inherent and Basic Design Characteristics.* An important safety consideration in any reactor is the ability to remove heat from the fuel sufficiently rapidly that the fuel elements do not overheat under any operating or accident conditions. Sodium is an excellent coolant because of its favorable combination of viscosity, conductivity, vapor pressure and specific heat. In addition, the CRBRP operates hundreds of degrees below the boiling point of the coolant. Therefore, the reactor coolant need not be pressurized; the sodium surface above the reactor is at essentially ambient pressure and the pressure exerted on the coolant system boundaries of the Plant is only that of the static sodium head plus the pump head required to force coolant through the reactor. For these reasons, the CRBR has very little stored energy in the

coolant; this is an outstanding advantage compared with systems that operate above the ambient vapor pressure of the core coolant at operating temperature (*e.g.*, LWRs). Small leaks, should they occur, have little likelihood of propagation into larger ones. Moreover, the low stored energy in the primary heat transport system does not of itself generate pressure within the secondary containment structure in case of leakage, thus greatly reducing containment structural requirements relative to those required for light water reactor plants.

In addition to the safety advantages inherent in the use of sodium as the coolant, a number of Plant design decisions were made to incorporate design features that avoid the occurrence of accidents or mitigate accident effects should they occur. Examples of these features are:

- A device in each control rod drive mechanism to prevent any rapid outward motion of rods.
- Provisions to prevent gas from entering the reactor core, including:
  - A vortex suppressor to prevent gas entrainment at the reactor vessel free surface, and
  - Continuous bleeding of small bubbles from the coolant.
- A thermal liner in the reactor vessel to maintain the upper vessel walls 100° to 150°F cooler than the reactor outlet temperature and to protect them from thermal transients associated with power level changes.
- Selection of core materials to give a negative Doppler coefficient of reactivity and thus provide a reliable feedback mechanism enhancing stability in normal operation and limiting reactivity excursions.
- Reactor fuel subassemblies with fuel pin spacing designed to reduce potential for reductions in coolant flow due to fuel swelling.
- Coordinated mechanical design of core assembly, core support, and fuel-handling machine control system to assure that a subassembly cannot be positioned by the fuel-handling machine in a location of increased reactivity or of reduced flow.
- Core support structure inlet modules and assembly inlet nozzles that provide multiple inlet passages and also prevent passage of foreign material which could cause flow blockage.

The project is using, to the maximum extent practicable, proven technology, including the incorporation of applicable FFTF, light water reactor, and other nuclear power plant experience. Where this technology and experience are not applicable or are only partially or indirectly applicable, an extensive program of development and proof tests is being implemented (Appl. Ex. 71, at 15.1-3, 15.1-4).

23. *Second Level of Design — Protection Against Anticipated and Unlikely Faults.* Recognizing that errors, or malfunctions can occur despite the care and attention given to the Plant design, construction, operation and maintenance, two avenues of second-level pursuit have been followed: (1) a number of protective systems and Plant features have been provided to protect against malfunctions, and to limit their consequences to definable and acceptable levels, and (2) a program of development and testing has been undertaken to define clearly the nature and consequences of accidents such as fuel failure, which might result from malfunctions. These features are:

- The reactor shutdown system (RSS) provides prompt automatic shutdown of the reactor when necessary to correct for off-normal conditions in the system. Two redundant, independent fast-acting systems are provided. Each system is complete with diverse sensors, logic, and circuitry, and each actuates separate, diverse sets of neutron absorber rods.
- All systems, components, and structures required for continued safe operation are designed to withstand or be protected from the effects of abnormal environmental conditions, such as earthquakes, floods or tornadoes.
- The three-loop heat transport system (HTS) design provides a redundant heat removal system such that core cooling is maintained even if, at the same time as a loss of normal power, an active component of one loop is disabled.
- Pony motors are provided for the primary and intermediate loop pumps of the HTS. They engage automatically upon reactor scram or shutdown to provide forced coolant circulation. The pony motors are capable of receiving power from the standby diesel generators.
- Natural circulation capability is provided in both primary and intermediate loops of the HTS.
- Extensive sodium leak detection capability is provided to assure that any failure of the primary boundary is detected promptly so that corrective action can be taken.
- A shutdown heat removal system having an independent flowpath exists; it uses the makeup and overflow system of the reactor vessel and rejects heat to the ex-vessel fuel storage system.

- The primary system components of each of the three independent heat transfer systems are installed in a massive reinforced concrete, steel-lined, inerted cell, capable of being isolated.
- A sensitive and redundant system is provided to detect the initiation of small leaks in the steam generator modules.
- A steam generator pressure relief system is provided to handle reaction products in the event of a large leak.
- The elevations of guard vessels and piping are configured to assure core coverage and continuity of core cooling even in the event of primary coolant system leaks.

The design emphasizes in this second level the need to ensure and confirm the high reliability of these protection systems and of any component or system whose failure could lead to severe core damage. An extensive program of qualitative and quantitative analysis and development testing is under way (*id.* at 15.1-4, 15.1-5).

24. *Third Level of Design.* The third level of design provides an extra measure of protection for the public health and safety, beyond that provided by the first and second levels, by imposing design requirements derived from low-probability events. Extremely unlikely faults are included as design basis events. The Plant design must include appropriate safeguard features to accommodate all of these events. Typical conservative assumptions, such as failure of a single component, are used in the analysis of these faults to demonstrate adequate design protection. Analytic evaluations of the capability of the Plant to withstand the identified extremely unlikely faults have been performed (*id.* at 15.1-6).

25. *Margins Beyond the Design Base.* In addition to the three levels of design derived from the defense-in-depth concept, a further extra measure of protection for the public health and safety has been provided by imposing structural and thermal margin requirements on the Plant design which are derived from a spectrum of events that lie beyond the Plant design base. The structural margins beyond the design base (SMBDB) impose additional structural loadings (based on CDA analyses) on the reactor vessel system and PHTS components and assure that extra margins exist to accommodate acceptably the additional requirements over and above those of the design basis accidents. The thermal margins beyond the design base (TMBDB) address the meltdown sequences that could follow a CDA and assure that the radiological consequences will be accommodated and/or mitigated to acceptable levels. Details and evaluations of the Plant capabilities in these regards have been provided (*ibid.*; Staff Ex. 27, at A.3-1 through A.3-16; *id.* at A.4-1 through A.4-27).

26. Additional efforts are under way to improve measurement capabilities for material control and accounting at the developmental reprocessing plant proposed by the DOE. This capability is not needed at the CRBRP nor intended for use there (Staff Ex. 36, at 2-4; Tr. 8176-78).

## II. ACCIDENT ANALYSES

### A. Design Basis Accidents

27. Design basis accidents are a set of events used to assess the way specific systems respond to abnormal conditions. As such, these events provide analytic tests of the design, selected to determine if installed or proposed safety features can cope adequately with the postulated event. For LWRs, plant response to these DBAs is assessed using the guidance from 10 C.F.R. Part 50, primarily the General Design Criteria, and the Standard Review Plan (NUREG-0800), primarily Chapter 15. It is normal Staff practice to require that conservative margins be demonstrated in analyses of the postulated events. Acceptance criteria applied in the tests range from mechanical stress limits to fuel cladding temperature limits. In addition, the postulated events must be acceptably mitigated, that is, meet all specified acceptance criteria, even if single failures are postulated to have also occurred in the safety systems under evaluation. Potential radiological consequences of DBAs are also assessed to determine whether predicted consequences fall within appropriate radiological dose guidelines. Dose guidelines for specific LWR accidents are specified in the Standard Review Plan, typically as fractions of the site suitability guidelines of 10 C.F.R. § 100.11. It is emphasized that these are dose guidelines for review rather than strict limits.

The design basis accidents were selected to represent a reasonable envelope of the credible events which might occur at a nuclear plant and which require mitigation by active systems or passive structures. The choice of the specific events typically depends on the type of reactor with different sets of events selected for BWRs, HTGRs (high-temperature gas reactors), PWRs, and LMFBRs (liquid metal fast breeder reactors). No regulatory criteria have been established for making these choices. Instead, engineering judgment regarding the kinds of faults or phenomena which might occur for a given kind of nuclear reactor is employed. The selected events may range from those which may occur once per year to those events which may never occur during the life of the plant.

When events have been judged to be so improbable that they are not "credible" as events against which the design should be tested, they have been excluded from the design basis envelope. For example, accidents involving an initiating event and simultaneous multiple failures of the mitigating safety systems have been judged so improbable that they have not been included as design basis accidents. Such accidents often have been designated as Class 9 accidents or "beyond-the-design-basis accidents." The term "Class 9 accident" has no official regulatory standing, but is used here because of its historical familiarity. Because Class 9 accidents typically involve some degradation of the reactor core, the term "core disruptive accidents" is also used to describe such severe accidents (Staff Ex. 26, at 15-4).

28. The Applicants and Staff have identified and analyzed seventy-one DBA events, organized into six categories as follows:

- Reactivity insertion events,
- Undercooling events,
- Local fuel failures,
- Fuel handling and storage events,
- Sodium fires, and
- Miscellaneous other events.

Within each of these categories, Applicants established qualitative event classifications based upon their assessment of whether the events are "anticipated," "unlikely," or "extremely unlikely." For each category and classification they provided a discussion of the:

- Identification of causes and accident description,
- Analyses of effects and consequences, and
- Conclusions.

(Appl. Ex. 71, Section 15.2; Appl. Ex. 72, Sections 15.3-15.7; Staff Ex. 26, Section 15, at 15-1 through 15-4; *id.* at 15-14 through 15-39.)

29. In analyzing the DBAs, Applicants identified bounding cases or umbrella events for which more detailed presentations were made. The remaining DBAs less severe than these bounding cases were treated more summarily. The bounding cases are:

- Reactivity insertion events involving a $0.60 instantaneous reactivity increase accompanied by the occurrence of a safe shutdown earthquake (SSE);
- Undercooling events involving loss of offsite power; and
- Fuel handling and fuel storage events in which some portion of the radioactive cover gas above the sodium pool is released.

The Staff found the identification of events and methods of analysis to be satisfactory (Staff Ex. 26, Section 15, at 15-5, 15-11, 15-12).

30. As a result of its review of the Applicants' DBA analyses, the Staff concluded that it has certain concerns relating to the adequacy of the analyses. These concerns involve principally the following considerations:

- It is not clear that all credible malfunctions have been considered that could permit overcooling of sodium in the intermediate heat transport system (*e.g.*, loss of feedwater heating);
- The Applicants' analyses of failure modes and effects of the heat transport system, the control system and the cover gas system need to be taken into account in the FSAR to demonstrate that the DBAs will bound all credible off-normal Plant conditions;
- The analyses of some of the reactivity insertion events involving the current heterogeneous core design concept suffers from the fact that in several instances Applicants have evaluated the differential changes resulting from the abandonment of the original homogeneous core design rather than having fully reevaluated all events for the current heterogeneous core design. The Staff will require a full evaluation in the FSAR for the OL review.

The Staff has emphasized that it does not view these concerns to mean that there is an inherent inadequacy in the CRBRP design; but rather the concerns are of a nature amenable to straightforward design or operational modifications should further analyses confirm the existence of problems (Staff Ex. 26, Section 15, at 15-10, 15-11, 15-17).

31. The Applicants have performed a conservative analysis of the response of the CRBR for conditions involving a reactivity insertion due to core compaction accompanied by an SSE. A maximum reactivity increase of $0.60 was used on the basis of core assembly design and anticipated manufacturing tolerance. The SSE occurrence was assumed to retard reactor shutdown by slowing down control rod insertion time. A small fraction of molten fuel was predicted for the hottest fuel subassembly, but the analysis indicated there would be no melting of cladding, no release of molten fuel, and no local sodium boiling. Some fuel rods may leak and release gaseous fission products that could increase sodium temperature locally. However, a limiting evaluation was performed that assumed that all 217 fuel rods in a subassembly released their contained fission gas into the coolant. This evaluation indicated that a margin to sodium boiling in excess of 100°F is maintained (Appl. Ex. 87, at 8; Tr. 7385; *id.* at 45-51; 7422-28; Appl. Ex. 71, at 15.2-4).

32. Applicants' analyses of the undercooling events and fuel-handling events indicated smaller challenges to the accident accommodation features of the Plant design than for the reactivity insertion-SSE events. Certain fuel-handling DBAs can result in calculable releases to the environment that are within dose guideline values. Applicants have concluded that all DBAs are limited, terminated or acceptably mitigated

by specific Plant safety features that assure that there is maintenance and/or reestablishment of a balance between heat generation and heat removal in the reactor core (Appl. Ex. 72, Sections 15.3, 15.5 and 15.7; Appl. Ex. 87, at 6; Tr. 7383).

### B. Beyond-Design-Basis Accidents

33. In LWR safety reviews, impacts of the failure to achieve reactor shutdown through proper functioning of the reactor control system have been considered and characterized by the term "anticipated transients without scram" (ATWS). For the CRBRP review, the term "unprotected transients" is used in an analogous manner except that for CDA initiation purposes the failure of the reactor to shut down is assumed — this despite the fact that an ATWS in the CRBR would require a failure of both of the RSSs. The Staff's review of reliability of these RSSs led to the conclusion that an ATWS event in the CRBR is substantially less likely to occur than in an LWR and thus it belongs more appropriately in the CDA analysis (Staff Ex. 26, at 15-9; Staff Ex. 27, at A.1-1, A.1-2; *id.*, Appendix B, at B-4).

34. Depending upon whether reactor shutdown has been achieved, core disruption may initiate at powers ranging from near normal to decay levels. The corresponding heating rates vary by two orders of magnitude and define a classification of CDAs into "unprotected" and "protected," respectively, depending upon whether the RSSs have failed, or have functioned properly. Mechanistically a protected CDA is the result of sustained failure to remove decay heat and is commonly referred to as a loss-of-heat-sink (LOHS) accident. In the unprotected CDA case, initial core disruption may occur due to either an undercooling or an overpower condition. Mechanically, the undercooling would be the result of loss of coolant flow, known as the loss-of-flow (LOF) accident, and the overpower would be due to an uncontrolled reactivity insertion, which is commonly referred to as a transient overpower (TOP) accident. In general terms, these three accidents exemplify the generic behavior over the whole range of the CDA spectra of circumstances, hence, they can be used to adequately characterize the spectra of energetic consequences. Another class of CDA initiators, that of fuel failure propagation, has also been identified and extensively studied. The evidence is conclusive that the attainment of whole-core disruption through such a mechanism can be neglected. Finally, various combinations of functional failure events (TOP/LOF, etc.) and/or structural failures (*i.e.*, due to extreme external events such as earthquakes beyond the SSE, yielding core support failures, loss of piping integrity,

etc.) have also been considered. The Staff's review of these areas indicates that those few cases for which severe energetics behavior cannot be precluded at this time are of sufficiently low probability to be neglected. The analytical approach thus consists of realistically following each one of the three generic CDA initiators through the core disruption phases and until accident termination. These so-called mechanistic CDA analyses provide an overall framework against which the potential for energetic phenomena is assessed with due regard for the controlling physical processes and for the accident accommodation design capability of the facility (Staff Ex. 27, at A.2-2).

35. An independent assessment of core-disruptive-accident energetics for the Clinch River Breeder Reactor has been performed for the Nuclear Regulatory Commission under the direction of the CRBR Program Office within the Office of Nuclear Reactor Regulation. It considered in detail the accident behavior for three accident initiators that are representative of three different classes of events: unprotected loss of flow, unprotected reactivity insertion, and protected loss of heat sink. The primary system's energetics accommodation capability was realistically, yet conservatively, determined in terms of core events. This accommodation capability was found to be equivalent to an isentropic work potential for expansion to 1 atmosphere of 2550 MJ or a ramp rate of about 200 $/s applied to a classical two-phase disassembly. This accommodation capability was contrasted to the potential for energetic behavior, which, due to the heterogeneous CRBR core design, was shown to arise only in the advanced core disruption states that lead to gravity-driven recriticalities. The core disruption behavior was assessed through integral analyses to establish an overall viewpoint, and separate, bounding evaluations of recriticality severity at various states of disruption; and separate, conservative estimates of fuel removal during disruption were also performed. The accident behavior was found to be dominated by neutronic activity that was bounded conservatively by 100-$/s events. This neutronic activity effectively terminated itself by promoting the necessary fuel removal from the active core, and it did so before a homogenized whole-core pool formed, thereby avoiding the regime of highest ramp rates. Even the whole-core pool was found to produce energetics levels within the system's accommodation capability. Based on a qualitative probabilistic approach, it was concluded that massive failure of the reactor head with associated early challenge to the containment building is physically unreasonable (Staff Ex. 42, at v).

36. If a CDA event is energetic enough, it could threaten the integrity of the upper reactor vessel (RV) closure head. This head provides

a barrier between the reactor vessel internals (reactor core) and the reactor containment building (RCB) environment. Figure 1 of Staff Ex. 41 (Tr. 8279) illustrates this point. Figure 2 of Staff Ex. 41 (Tr. 8280) provides some detailed perspective of the reactor vessel, head and cavity regions. The operating floor (which is illustrated in Figure 1), together with the head, isolates the regions containing primary sodium from the containment environment. If the head should fail, radioactive materials could be released directly from the disrupted core to the RCB environment. These materials would then be available to leak to the atmosphere early in the CDA sequence. In addition, such a failure could challenge the integrity of the containment by sodium fires or missiles. If the head remains intact the disrupted core will be retained within the reactor vessel or the debris will eventually be discharged to the reactor cavity where it will (at least initially) still be isolated from the containment environment (Staff Ex. 41, at 11; Tr. 8282).

37. The levels of energetics required to produce significant structural damage in the CRBR were evaluated taking into account an "inner containment" formed by the Core Barrel (CB)/Upper Internal Structure (UIS)/Core Support Structure (CSS) envelope. This configuration is illustrated schematically in Figure 4 of Staff Ex. 41 (Tr. 8305). In addition, the pressure transmission characteristics of the two-phase expanding core medium and other materials found within were also taken into account. These characteristics have important implications on the resulting short-term loading of the local CB and CSS structures. This mitigating behavior is the result of a compliant core state (distributed voids) and it must be taken into account particularly since such compliance is one of the crucial prerequisites for highly energetic behavior to start with.

The analysis of the energy level required to fail the head was conducted in two steps. The first step involved evaluation of the response of the "inner containment" (*i.e.*, the "cage" formed by the CB-UIS-CSS envelope) to the fuel vapor expansion process. If the "cage" boundary fails, the fuel vapor can then expand against the sodium pool above the upper cage boundary (*i.e.*, the UIS). The second step in the evaluation involves the analysis of the expansion into the sodium pool. To assure conservatism in the analysis, not all losses expected in a real expansion were included. The analyses of both steps are described in detail in Staff Ex. 42. These analyses indicate a level of energetics on the order of 1130 MJ (isentropic expansion yield to 1 atmosphere) would be required to breach the inner containment. That is, minimal energetic release against the boundary of the primary system can be expected for any energetics below this level. At still higher levels, an upward displacement of the UIS and a longer-term expansion against the sodium pool would take place. Evaluations of the long-term expansion phenomena indicate that an energetic event of nearly twice the above magnitude, approximately 2550 MJ, would be required to produce a slug impact kinetic energy close to the vessel head design capability of 75 MJ committed to by the Applicants (Staff Ex. 41, at 32-33; Tr. 8304-05; Staff Ex. 42, Section II.2).

38. In Items 11 and 16 of Appendix D, the Board requested that the Staff provide additional information regarding CDA release energetics and aerosol behavior, respectively. The Staff's testimony addressed these two matters and dispelled the Board's concerns about them (Staff Ex. 41, at 50-51; Tr. 8324-25; *id.* at 85-87; Tr. 8363-65). Staff consultants, in addressing the dynamic response of the CRBR to a CDA, identified eight areas of concern regarding Applicants' related analyses (Staff Ex. 42 (NUREG/CR-3224), Section I, Table II). In Item 17 of Appendix D, the Board requested the Staff to provide its position regarding these eight areas of concern. Staff testimony addressing this matter indicated that all the eight areas have been satisfactorily resolved. However, for one of these concerns (No. 3, involving the behavior of fission product gas during fuel pin disruption) there exists a difference of technical positions between Applicants and Staff as to the contribution of fission gas to fuel disruption. In this case, the satisfactory resolution reported by the Staff derived from Applicants' agreement to a fuel pin design modification if further analysis does not eliminate the differing positions (Staff Ex. 41, at 39-40; Tr. 8313-14; *id.* at 54-55; Tr. 8328-29; Tr. 8454-59).

39. LWR core melt accident analyses have generally indicated that containment failure can be expected at about 24 hours into the accident. Such a failure is assumed to be accompanied by uncontrolled and unfiltered releases to the environment. The CRBRP containment design objective is to prevent containment failure by means of controlled and filtered venting, if needed, subsequent to a CDA. Thus, rather than requiring no containment venting prior to 24 hours into an accident for the CRBRP the Staff has focused upon the following guidelines to assess containment adequacy:

- There must be adequate information upon which to base a decision of whether and when to vent;
- There must be adequate time between the decision that venting may be required and the time at which venting is initiated to implement protective action measures such as evacuation or sheltering;

- There must be adherence to 10 C.F.R. Part 100 dose guidelines (appropriate to the CRBR core inventory) as a consequence of venting; and
- Because the filtering and venting capability might conceivably fail to adequately protect the containment, there must be a high level of assurance that 10 C.F.R. Part 100 dose guidelines will not be significantly exceeded (Staff Ex. 27, at A.1-4, A.1-5).

40. Both Applicants and Staff have presented evidence to support their separate conclusions that CDAs should not be included within the spectrum of DBAs. Each body of evidence relies upon similar sets of considerations to support the indicated conclusion. These considerations primarily include the following:

- Redundancy, independence and diversity of Plant protection systems (PPSs) and their ability to function properly despite single failures postulated to occur;
- Comprehensive design approaches that anticipate all identified accident modes;
- Rigorous quality assurance to assure that materials and components conform to design intent;
- Selections of materials and of component designs for which prior operational and test experience is available or is being obtained; and
- Probabilistic risk assessment and reliability assurance programs to critique Plant and component performance.

(Appl. Ex. 87, at 1-217; Tr. 7378-7594; Staff Ex. 32, at 1-66; Tr. 8036-8101).

41. The risk to the maximally exposed individual is estimated by multiplying the risk estimators presented in Section 5.7.2.5 of Staff Ex. 8 by the estimated annual total body doses to the maximally exposed individual. This calculation results in a risk of potential premature death from cancer to that individual from exposure to radioactive effluents from 1 year of reactor operations of less than one chance in one million. The risk of potential premature death from cancer to the average individual within 50 miles of the reactor from exposure to radioactive effluents from the reactor is much less than the risk to the maximally exposed individual. These risks are very small in comparison to natural cancer incidence from causes unrelated to the operation of CRBRP. Multiplying the annual U.S. population dose from exposure to radioactivity attributable to the normal operation of CRBRP and its related fuel cycle (*i.e.*, 170 person-rems to the general public) by the preceding somatic risk estimator, the Staff estimates that about 0.023 potential

cancer death may occur in the exposed population. For the purposes of evaluating the potential genetic risks, the progeny of workers at CRBRP are considered members of the general public. Multiplying the sum of the U.S. population dose to the general public from exposure to radioactivity attributable to the normal annual operation of CRBRP and its related fuel cycle (*i.e.*, 170 person-rems), and a conservative estimate of the dose from occupational exposure (*i.e.*, 1000 person-rems) by the preceding genetic risk estimators, the Staff estimates that about 0.30 potential genetic disorder may occur in all future generations of the exposed population. The significance of these risk estimates can be determined by comparing them to the natural incidence of cancer death and genetic abnormalities in the U.S. population and in the first five generations of the U.S. population, respectively. Multiplying the estimated U.S. population for the year 2010 (~280 million persons) by the current incidence of actual cancer fatalities (~16%) and the current incidence of actual genetic ill health (~11%), about 45 million cancer deaths and about 150 million genetic abnormalities in the U.S. population and in the first five generations respectively are expected (HHS 1981, BEIR-III). The risks to the general public from exposure to radioactivity attributable to the annual operation of CRBRP are very small fractions (less than ten parts in a billion) of the estimated normal incidence of cancer fatalities and genetic abnormalities in the year 2010 population and in the first five generations of the year 2010 population, respectively.

On the basis of this comparison, the Staff concludes that the potential risk to the public health and safety from exposure to radioactivity attributable to normal operation of CRBRP and its related fuel cycle will be very small (Staff Ex. 8, Vol. 1, at 5-21, 5-22).

## C. Dose Consequences of Accidents

42. Appendix I to 10 C.F.R. Part 50, Section II states (in pertinent part) as follows:

> A. The calculated annual total quantity of all radioactive material above background[1] to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

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[1] Here and elsewhere in this appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors, of which specific account is required in determining design objectives.

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

43. Title 10 of the Code of Federal Regulations provides three different sets of guidelines that govern radioactive materials and radiation doses. Their respective scopes and purposes are briefly summarized:

*10 C.F.R. Part 20* (including Appendices) serves as a guide to the packaging and handling of radioactive materials and lists the allowable concentrations of radioactive materials that may be tolerated in air and in water in restricted access areas as well as in nonrestricted access areas.

*10 C.F.R. Part 50, Appendix I* provides, for LWRs, guidance regarding plant design and operation such that the resultant radiological doses from plant effluents are acceptably low in restricted and unrestricted access areas. Numerical dose guideline values are given for normal operation and for expected operational occurrences such as DBAs.

*10 C.F.R. Part 100* deals with reactor site criteria. The dose guidelines provided therein are *not* offered as biologically acceptable values. Rather, they deal with severe accident doses that must not be exceeded for any specific proposed reactor (not limited to LWRs) at any specific proposed site. As such, they offer guidance only with respect to site suitability.

It is important to note that Parts 20 and 50 offer guidance in the form of *upper limits* on radioactivity and doses. Both Parts impose an obligation on Applicants to maintain values as low as reasonably achievable (for which the acronym ALARA is frequently used) and stress the importance of doing better than guideline values.

44. Applicants have summarized the dose results from an SSST release from the core of the CRBR into the reactor containment building and have compared these doses with guideline values. The results are as follows:

GUIDELINES AND DOSES FOR SSST RELEASE

| Organ | 10 C.F.R. 100 or Equivalent Guideline Doses (REM) | Construction Permit Guideline Doses (REM) | Exclusion Area Boundary (2-Hour Doses) (REM) | Low Population Zone (30-Day Doses) (REM) |
|---|---|---|---|---|
| Whole Body | 25 | 20 | 1.3 | 0.9 |
| Thyroid | 300 | 150 | 8.2 | 6.8 |
| Lung | 75 | 37.5 | 0.6 | 0.5 |
| Liver | 150 | 75 | 0.4 | 0.4 |
| Bone Surface | 300 | 150 | 9.0 | 8.9 |
| Red Bone Marrow | 75 | 37.5 | 0.7 | 0.7 |

The bases and assumptions underlying these results were also presented (Appl. Ex. 87, at 205-09; Tr. 7582-86).

45. The NRC has established an Accident Source Term Program Office to address severe accident source terms for LWRs. If the efforts of that office do indicate a need to change the source term assumptions in a manner that might impact the CRBRP (not considered likely by the Staff), straightforward design modifications can accommodate such changes. These will be dealt with by the Staff during the OL stage of review (Staff Ex. 41, at 115-18; Tr. 8393-96).

46. In its consideration of the dose consequences of CDAs, the Staff identified four categories (I through IV) of primary system responses to accident initiation and three modes (A through C) of containment responses. It then defined four classes of CDAs (Class 1 through Class 4) of increasing severity, based upon various combinations of primary system and containment responses. For these four CDA classes, the Staff then estimated radioactivity release frequencies and stated which classes of CDAs would lead to offsite doses that might exceed 10 C.F.R. Part 100 dose guidelines. The results are summarized here.

The Staff characterizes CDA Class 1 as the most probable CDA, with an estimated likelihood of less than (<) $10^{-4}$ (*i.e.*, <1 chance in 10,000) per reactor-year of operation, and having offsite doses below the 10 C.F.R. Part 100 guidelines. The two most probable CDAs for which doses could exceed 10 C.F.R. Part 100 guidelines are CDA Class

2 and Class 3. Since CDA initiation is itself estimated to be $<10^{-4}$ per reactor-year and containment Modes B and C are each estimated to have a likelihood of $<10^{-2}$ per demand, Class 2 and Class 3 CDAs are each estimated to have a frequency (or likelihood) of occurrence of $<10^{-6}$ per reactor-year of operation. The least likely CDA to occur for which doses could exceed dose guidelines is CDA Class 4. In this instance, the Staff combined the frequencies of CDA initiation of $<10^{-4}$, of primary system response IV of 0.1 per demand, and of containment Mode C of $<10^{-2}$ per demand to estimate a combined Class 4 event frequency of $<10^{-7}$ per reactor-year of operation. The percent of core inventory released to the environment by each CDA class was also analyzed and reported by the Staff (Staff Ex. 8, Vol. 2, at J-5 through J-12).

47. To assess a spectrum of HCDA consequences, Applicants analyzed four cases in detail using successively more pessimistic assumptions regarding releases to the Reactor Containment Building during the initial release phase.

1. Case 1 is based on realistic evaluation of the HCDA sequence — a nonenergetic accident. Consequently, no significant immediate release of sodium or fission products through the reactor vessel closure head seals is considered. Penetration of the reactor vessel and guard vessel is assumed to occur at 1,000 seconds. At that time, all of the noble gases and the most volatile fission products (Cs and Rb) were assumed to be vented from the reactor cavity to the reactor containment building. Containment venting and purging through filters is assumed to begin at 36 hours.
2. Case 2 is similar to Case 1, except that an energetic CDA is assumed, such that the available work energy from fuel expanded to the free volume of the reactor vessel would be approximately 100 MJ. As noted earlier in this Exhibit (Tr. 7766-67), this provides a conservative representation of CDA energetics potential. Since the reactor vessel, head and primary system are designed to retain their structural integrity for the dynamic loadings derived from the 100-MJ condition, the immediate releases would still be limited. To represent this condition, an immediate release of 1,000 pounds of sodium and a gas leak rate of 1,000 standard cubic centimeters per second for the first 1,000 seconds were used. Following meltthrough, the releases to the reactor containment building were similar to those in Case 1.
3. Case 3 is similar to Case 2, except that a large immediate release of fuel, fission products, and sodium to the reactor con-

tainment is assumed in order to examine the sensitivity of the consequences to assumed releases that are much larger than expected. An immediate release of 1,000 pounds of sodium, 1% of the fuel and solid fission products and 100% of the noble gases, halogens, and volatile fission products was assumed.
4. Case 4 is similar to Case 3, except that the amount of sodium immediately released was increased to 3,300 pounds and the amount of fuel and solid fission products was increased to 5%.

The results show that:

1. Based on the best estimate of the energetics consequences of a CDA (Case 1), the doses are acceptably low. The dose for whole body and all organs are even below the 10 C.F.R. Part 100 guidelines or equivalent values.
2. Even assuming an energetic CDA (as in Case 2), the doses are very similar to Case 1 because the design prevents the short-term release of significant quantities of materials from the reactor coolant boundary.
3. The doses are not very sensitive to even much higher short-term releases of materials into the containment (Cases 3 and 4). This result is due to the aerosol fallout and plateout, which increase with increasing quantities of materials in the containment atmosphere. With a higher rate of radioactive material depletion by aerosol formation, less material would remain in the containment atmosphere to be available for leakage.

These dose calculations are based on the initiation of venting at 36 hours, which is the nominal predicted time for venting. Additional sensitivity studies have shown that the doses are not very sensitive to vent times over a range of times between about 10 to 36 hours because of the effectiveness of the cleanup system (CRBRP-3, Vol. 2, Appendix K.2). Even for these earlier vent times, the predicted doses remain below the 10 C.F.R. Part 100 or equivalent guideline values. It is concluded that the design features to mitigate CDA consequences provide an effective means to control the releases for a wide range of conditions. The resulting radiological consequences would be acceptable.

Applicants' overall conclusions from their analyses are that:

1. Adequate analyses of CDAs have been performed.
2. Although the analyses of CDA sequences predict a nonenergetic outcome, the design provides capability to accommodate an energetic CDA, and thus prevent a short-term challenge to the containment integrity.
3. Sodium-concrete reactions following loss of core geometry and penetration of the reactor vessel and guard vessel have been

adequately analyzed. The analyses show that the Plant design features can accommodate the full range of sodium-concrete reactions observed experimentally.

4. Sensitivity studies have been performed to assess a wide spectrum of whole-core-melt sequences. The studies show that the design features would effectively mitigate these sequences and that long-term integrity of the containment structure above the basemat would be maintained.
5. The design features would provide effective control of radiological releases for whole-core-melt sequences.
6. The consequences of a CDA in CRBRP are acceptably low.

The various computational methodologies, codes, and code validation efforts to support these analyses by Applicants are reviewed (App. Ex. 89, at 50-60; Tr. 7812-22).

## D. Intervenors' Challenge to Accident Analyses

48. Board Exhibit 125 (limited appearance statement of NRDC and the Sierra Club) alleges certain deficiencies in the CDA analyses of the Staff and Applicants that would, if considered in the manner prescribed by their statement, lead to the conclusion that CDAs should be included within the envelope of DBAs. These alleged deficiencies fall into three categories:

- Improper assessment of thyroid doses;
- Improper assessment of likelihood of CDA occurrence; and
- Improper assessment of site suitability

(Board Ex. 125, at 1-17; Tr. 7653-69). Each of these subject areas is addressed in findings that follow.

49. The limited appearance statement (Statement) offered two bases to support the allegation regarding improper thyroid dose assessment:

- That dose calculations made for comparison with 10 C.F.R. Part 100 guidelines should have considered infants rather than adults because of higher infant susceptibility and higher infant respiration rate; and
- That data from findings based on the accidental exposures of Marshall Islands residents support the use of a higher guideline value than given by 10 C.F.R. Part 100.

The parties (Applicants and Staff) rebutted these claims on the grounds that TID-14844 prescribes adult thyroid dose calculations for the purposes of 10 C.F.R. Part 100 guideline comparisons, and that the statistical reliability of the Marshall Islands data is highly uncertain and the data base is unreliable. The Marshall Islands report, cited in very limited part by the Statement, itself discourages use of those data for quantitative comparisons. [The Board notes further that, for the purpose of evaluating compliance with Appendix I of 10 C.F.R. Part 50, Reg. Guide 1.109 (Rev. 1, October 1977) does indeed take account of higher dose response characteristics of infants.] (Board Ex. 125, at 1-17, Tr. 7653-69; Tr. 7717-19; Tr. 8503-04; Tr. 8527; Appl. Ex. 96).

50. The Statement's dissatisfaction with the Staff's estimates of the likelihood of CDA occurrences is based upon Staff Exhibit 24 (Vol. 2) [note that Staff Ex. 24 comprises the identical documents that are also identified as LWA Staff Ex. 8]. Appendix J thereto estimates the accident initiation occurrence frequency to be $10^{-4}$ per reactor-year of operation. The Statement interprets this as being impermissibly larger than the $10^{-6}$ number prescribed in Staff Exhibit 5. What is apparently overlooked is the fact that accident initiation, per se, does not lead to a CDA unless there is a subsequent failure of safety features provided to mitigate accident consequences. Appendix J estimates the conditional probabilities that accident initiation will be accompanied by subsequent failures of mitigating safety features that then result in offsite doses, and concludes that for those CDAs for which offsite doses exceed guidelines, the likelihood of occurrence is $10^{-6}$ or less (Board Ex. 125, at 1-17; Tr. 7653-69; Staff Ex. 24, at J-8 through J-11).

51. The Statement challenges the favorable site suitability determination in part upon considerations disposed of in the two immediately preceeding findings. In addition, the Statement faults the Staff's use of meteorological parameters that are alleged to be unconservative, and uncertain. The Staff has rebutted both the unconservative and uncertainty allegations and explained the bases for the meteorological parameters it has used. There is more detailed meteorological information available for the CRBRP site than is usually the case at the CP stage of review for most LWRs, eliminating the need for the uncertainty factor frequently applied to LWR dose calculations because of less complete meteorology (Board Ex. 125, at 1-17; Tr. 7653-69; Tr. 8500-10; Staff Ex. 49).

52. The Statement raises a further objection alleging improper offsite dose calculations that do not take account of releases arising from the operation of the CRBRP containment vent/purge, thus seemingly invalidating the determination of site suitability. The Statement bases its allegation upon the assertion that such an accounting was taken for two specific older reactor systems. Applicants testified that the CRBRP vent/purge system operation plays no role in the context of evaluating CRBRP design basis accident consequences and site suitability evaluation, and that the two reactor systems alluded to do not contain

the functional analog of the vent/purge system, but more nearly the analog of the annulus filtration system. Releases from the CRBRP annulus filtration system are included in site suitability analyses (Board Ex. 125, at 9-17; Tr. 7661-69; Tr. 7722-25).

53. Other deficiencies alleged by the Statement (*e.g.*, conclusions drawn from preliminary and incomplete PRA analyses) have been reviewed and are judged to be insufficiently persuasive to warrant consideration.

## III. QUALITY ASSURANCE

54. Within the context of nuclear industry usage, the term "quality assurance" (QA) includes the functional activity known as quality control (QC) (Appl. Ex. 95, at 4; Tr. 8628; Tr. 8669, 8671).

55. Appendix B to 10 C.F.R. Part 50 defines these terms as follows:

> "quality assurance" comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

(Staff Ex. 44, at 2; Tr. 8761).

56. The United States government is the owner of the Clinch River Breeder Reactor Plant (CRBRP); the U.S. Department of Energy (DOE) is custodian and has lead management responsibility. The CRBRP Project Office (PO) is the owner's management organization. This Office is staffed by personnel from DOE, the Tennessee Valley Authority (TVA) and Project Management Corporation (PMC). TVA will be responsible for Plant operation and maintenance. PMC is responsible for administering the interest of the utility industry with respect to the CRBRP; it provides personnel and financial support (Appl. Ex. 95, at 1; Tr. 8625).

57. The PO has contracted with the following organizations that are contractually joined to manage and complete the design and construction of the CRBRP; major areas of responsibility are given for each:

Westinghouse Electric Corporation, Advanced Energy Systems Division (AESD), is the Nuclear Steam Supply System Supplier (NSSS/S) and is responsible for the overall design and manufacture of the Nuclear Steam Supply System (NSSS). AESD is also specifically responsible for the design and manufacture of reactor and reactor enclosure systems, primary sodium heat transport system, and related components and controls;

General Electric Corporation, Advanced Reactor Systems Department (GE-ARSD), is a Reactor Manufacturer (RM). In this capacity it is a major contractor for the NSSS/S and is responsible for the design and manufacture of the intermediate heat transport system, and related systems and controls;

Rockwell International Energy Systems Group, Atomics International Division (ESG-AI), is a Reactor Manufacturer (RM). In this capacity it is a major contractor for the NSSS/S and is responsible for the design and manufacture of fuel-handling systems, auxiliary sodium systems, reactor plant maintenance system, and related systems and controls;

Burns and Roe, Incorporated (B&R) is the Architect-Engineer (AE) for the overall Plant, including Balance of Plant (BOP) and portions of the NSSS; and

Stone and Webster Engineering Corporation (SWEC) is the Plant Constructor and will function as both a construction manager and the Plant construction contractor.

In addition, the DOE, through its Hanford operation, is the nuclear fuel supplier (Appl. Ex. 95, at 1-3; Tr. 8625-27).

58. The QA programs of the Applicants and of each of the major participants have been described in detail by the Applicants in the PSAR (Appl. Ex. 73, Appendices 17A-17F, 17H-17J).

59. The CRBR PO maintains a management policies and requirements document (MPR), signed by the director, that governs and is followed by all program participants in implementing their individual QA programs. Noncompliance with the MPR results in a formal inquiry as to why it happened and what will be done to prevent a recurrence (Tr. 8737).

60. The MPR is an evolving document that represents a way of life in the CRBR project. It goes back to the early 1970s and has involved each of the project participants, whose individual management systems have been structured to conform thereto, and whose personnel have implemented same in accordance with specific procedures prepared by these participant organizations (Tr. 8720-21; 8711-19). The QA management experience and knowledge of the TVA has been factored into the CRBR QA program development (Tr. 8723).

61. The CRBR QA program is applied in a graded manner to all systems, structures, components and activities of the project, not just those designated as safety-related and important to safety. To implement this graded approach, the MPR defines nine levels of QA programs that

have been developed for application based upon the importance of the items and activities to the Plant functions (Appl. Ex. 95, at 20-23; Tr. 8644-47; Tr. 8727, 8731).

62. The MPR establishes communications channels amongst the program participants (Tr. 8677-78), design responsibilities for the participants and approval requirements for their activities (Tr. 8680-81), and a configuration management plan that controls the identification of project requirements and changes thereto (Tr. 8697).

63. QA problems at other plants have been reviewed in depth by all participants to assure that the CRBR QA program is properly structured to avoid such problems (Tr. 8725).

64. Activities within the CRBR QA program are coordinated and integrated by means of three levels of controls, as indicated below. The first level of control includes the system, component, material and service suppliers. Their quality assurance programs are primarily quality control programs concerned with direct control and verification through analysis, review, inspection, examination and testing. This level requires the performer of an activity to implement a system of checks and balances that provides direct control over his work process.

The second level of control includes the program participants that have direct and indirect interfaces with each other and the PO. The NSSS Supplier and the Constructor are examples of this level of participation. These portions of the overall program are management-type programs with responsibilities for the quality assurance functions such as surveillance, audit, interface coordination, and lower-tier program integration functions including overview of the lower-tier quality control processes.

The PO portion of the program is the third level of control. The PO is responsible to the DOE for the overall program and its adequacy. The PO program is a management-type program with audit and surveillance activities for verification of participant performance, interface coordination and program integration including the coordination of fabrication and construction efforts for the project.

This system is designed to provide the inspections and review functions, the verification and overview of those functions, and the checks and rechecks necessary to assure the quality required for this Plant and to minimize QA oversights. The entire quality assurance program is a major part of the management control systems that cut across all levels of project activities. A strong PO organization is in place to coordinate and integrate the design, fabrication and construction effort. It serves to minimize problems with quality, especially where interfacing is involved. The coordination of interfacing systems is controlled through a formal review and approval cycle that provides the necessary safeguards for proper system integration. Prespecified contractual provisions have established the mechanisms for surveillance and audit among participants (Appl. Ex. 95, at 11-12; Tr. 8635-36; Tr. 8745-46).

65. Specific attention to job-site safety is focused upon within the PO organization and the constructor's organization. These organizations have personnel with explicit responsibility for this activity. The job-site safety program complies with OSHA and DOE regulations (Tr. 8742-44).

66. The Applicants' PO has reviewed and accepted each participant's QA program and assured itself of the independence of said programs from undue influence due to cost and schedule considerations. Management dedication to QA within all participant organizations, and especially within the DOE's PO was attested to (Tr. 8750-53).

67. Within each participant's organization, a configuration management system operates to assure that engineering change proposals, however they may originate, are processed through appropriate, prescribed approval channels including the ultimate approval authority of the Project Office. A computerized project documentation and control system tracks and records all changes. This system is updated daily and is used to ensure that up-to-date information on drawings and specifications is timely available (Tr. 8682-92; 8732-35).

68. The Staff has performed a review of the QA programs of each of the major participants and concluded as follows:

> On the basis of its detailed review and evaluation of the QA programs of the owner (DOE's CRBRP Project Office), the NSSS supplier (Westinghouse Electric Corporation, Advanced Reactors Division), the architect-engineer (Burns and Roe, Inc., Breeder Reactor Division), the constructor (Stone & Webster Engineering Corporation), and the two reactor manufacturers (General Electric Company, Fast Breeder Reactor Department and Rockwell International Corporation, Energy Systems Group), the Staff concludes that the QA program for design, procurement, and construction of the CRBRP meets the requirements of 10 C.F.R. Part 50, Appendix B, and is acceptable, except that certain additional information and clarifications are necessary regarding the items that are under the control of the QA program. These items have been identified in the PSAR and are being reviewed by the Staff. At the completion of this review, the Staff will require additional information from the Applicant. This item is considered open until satisfactory resolution is obtained.

(Staff Ex. 26, at 17-1 through 17-5, 17-8.)

69. The Staff's review of the open items of Applicants' QA program was completed, and the results were found to be satisfactory, as noted:

> The Staff's evaluation of the Applicants' QA program is provided in Section 17.3 of the Clinch River Breeder Reactor Plant (CRBRP) SER (NUREG-0968, dated

March 1983). The program was reviewed against the applicable QA criteria of 10 C.F.R. Part 50, Appendix B (as reflected in NUREG-0800, "Standard Review Plan") and TMI Action Plan (NUREG-0660) Item I.F. In the SER, the Staff indicated that it was still reviewing the list of structures, systems, and components controlled by the CRBR QA program. The Staff has completed its review and asked several questions in this regard. The Applicants have provided a response (Longenecker to Grace letter dated May 5, 1983) which acceptably addressed the Staff questions. Thus, the Staff has found the description of the Applicants' QA program and the list of items to which it applies acceptable and now has no open items in this regard.

(Staff Ex. 30, at 17-1.)

70. Staff field and headquarters personnel responsible for QA are aware of no deficiencies or needed changes to Applicants' QA program that would prevent or would be needed to permit its implementation (Tr. 8785-86).

71. Implementation of the NRC's inspection and audit of Applicants' QA program began several years ago. Inspections started in April 1975 were conducted at the site Project Office, the architect-engineer's (Burns and Roe, Inc.) corporate offices, and various manufacturing facilities where CRBRP equipment was being fabricated. Areas examined included: (1) program organization, (2) QA program for design and procurement, (3) implementation of QA program for design and procurement, (4) audit reports, (5) manufacturing process control, and (6) manufacture of selected equipment. Major components inspected during manufacture included, but were not limited to, the reactor vessel, vessel closure head, core support structures, sodium pumps, and the core restraint systems. These inspection activities were reviewed against the design guidelines that comply with the DOE standard RDT F2-2, "Quality Assurance Program Requirements," and the 10 C.F.R. Part 50, Appendix B, criteria.

The inspection findings revealed some deviations and unresolved items with respect to the CRBRP PSAR commitments and standard RDT F2-2. However, upon termination of licensing and inspection efforts in 1977, the CRBRP Project Office had completed all corrective actions and resolved all open items identified by the inspectors. The CRBRP Project Office actions were documented in an amendment to the PSAR that was issued in October 1977.

Since licensing review of CRBRP has begun again, Region II has reinitiated the inspection program. A site inspection was conducted in October 1982 to observe site-clearing activities and to examine stored equipment. An overall inspection program is being developed that will include extensive examination of stored equipment and fabrication records. Review of updated QA programs began in the spring of 1983 (Staff Ex. 26, at 17-8, 17-9; Tr. 8794-95, Tr. 8796).

72. NRC's Office of Nuclear Reactor Regulation has assigned a full-time person to the CRBR PO. NRC has recently contracted with an outside organization to assist with its evaluation of the Applicants' overall design control efforts. The decrease of nuclear construction activities in general assures the availability of more than the usual number of inspection personnel to assist the resident, onsite inspector and to participate in the overall inspection and audit activities. The NRC's light water reactor construction surveillance program is being reviewed in depth against the CRBR QA program to critique and determine the adequacy of Applicants' program plan (Tr. 8790-99 *passim*).

## IV. EARTH SCIENCES AND ENVIRONMENTAL MATTERS

### A. Geology and Seismology

73. We reviewed the geological, seismological and subsurface setting of CRBR in our Partial Initial Decision (17 NRC 158, 244-46, Findings 9-11) and concluded that the site was suitable based upon the evidence then before us.

74. We now have before us Staff Ex. 26 (NUREG-0968, Vol. 1, Safety Evaluation Report related to the construction of the Clinch River Breeder Reactor Plant). Section 2.5 of the Exhibit includes extensive analyses of the Geology and Seismology of the CRBR setting and includes new information (at 2-18 to 2-40).

75. The USGS acted as advisor to the Staff in the review of the geology and seismology, and its findings are included as Appendix H of Vol. 2 of the SER (Staff Ex. 27).

76. The USGS review, however, conservatively and properly identifies some uncertainties relating to the CRBR site. From their review of the geology as presented in the PSAR, the USGS identified two items of major concern. These were the possibilities of a limestone cavern underlying some portion of the site and of active faulting in the area (Staff Ex. 27, Appendix H, at 3).

77. Concerning limestone caverns at the site, the USGS concluded,

> [e]xamination of the drill-core and the geologic cross-sections drawn by the Applicants, limitation of known caverns to the Knox Group (PSAR, p. 2.5-7), and the concept of "continuous rock" based on the core-hole data and seismic refraction

work, makes reasonable the Applicants' contention that the presence of a major undetected cavity beneath a site structure is unlikely (PSAR, p. 2.5-15a).

The Staff agrees (Staff Ex. 26, at 2-22 to 2-23).

78. Concerning active faulting in the area, the USGS was more skeptical. They noted that the Applicants had not carried out mapping and/or trenching across two critical faults in the area. The CRBR site is located between these thrust faults, which are the Copper Creek and Whiteoak Mountain Faults. The Copper Creek Fault at its closest point to the site is about 3000 feet to the south. The Whiteoak Mountain Fault system consists of a main thrust fault with several subsidiary branch faults, the nearest trace being 1.7 miles northwest of the site (Staff Ex. 27, Appendix H, at 2-3).

79. The USGS noted that mapping and/or trenching across these faults could have demonstrated conclusively that the Copper Creek Fault and the Whiteoak Mountain Fault are not capable. They concluded that, although there had not been as definitive a demonstration as possible of the noncapability of faults in the area, the analysis of site geology by the Applicants resulted in reasonable conclusions based upon current theories of Appalachian tectonics and upon the data available. They also noted that to date no active faults have been recognized throughout the Appalachian region (Staff Ex. 27, Appendix H, at 4).

80. The USGS reviewed the seismological analyses presented by the Applicants in the PSAR. They compared that with the seismological literature and with some results of ongoing research by the USGS. Recently a large number of eastern U.S. earthquakes have been relocated. Nine of these make up a zone 15 km wide and 180 km long that runs through Knoxville and forms an azimuth nearly 20 degrees more northerly than the surface trend of the Appalachians. The USGS notes that this may represent a concentration of seismicity in eastern Tennessee and that it is possible this alignment represents a basement seismic source zone or fault. They add, however, that there is insufficient evidence to identify a specific structure, but they raise clearly the possibility of a local seismic source (Staff Ex. 27, Appendix H, at 5).

81. The USGS presents evidence both for and against the hypothetical structure. They also present the results of calculations based on numerous assumptions which, treating the hypothetical structure as a fault, show that the CRBR SSE has an exceedance probability notably higher than $1 \times 10^{-4}$ (Staff Ex. 27, Appendix H, at 5-7).

82. In summarizing their review, the USGS states that the selection of the Giles County earthquake by the Applicants was reasonable. They also concurred with the assessments of the maximum intensity and SSE,

and the anchoring of a Regulatory Guide 1.60 response spectrum to this 0.25*g* SSE. Moreover, they believe the CRBR SSE has a conservative exceedance probability if one can confidently adopt a diffuse seismicity model to an Appalachian province. They found that at the present time the data are insufficient to establish whether or not the hypothetical local source exists. They concluded that the CRBR SSE is reasonable based on present data and that it would take a definitive seismological investigation to address the question of a possible concentrated seismic source in eastern Tennessee. Such an investigation would require a local network, velocity models and source mechanism determinations (Staff Ex. 27, Appendix H, at 8).

83. The Staff independently reviewed the geology and seismology relating to CRBRP. It took cognizance of the USGS concerns regarding both (Staff Ex. 26, Section 2.5).

84. Regarding the capability of faults in the area, the Staff made the following statement:

> The staff concludes that the faults at the site and in the region around the site are not capable. There are, however, additional data which might, if appropriate exposures are available, be utilized to confirm that conclusion. High terraces of probable Pleistocene age are relatively common in the site region. These terraces were used by the Tennessee Valley Authority (TVA) in the Phipps Bend (1975) and Watts Bar (Apr. 1974) geologic investigations to demonstrate, along with other data, that local Valley and Ridge Faults are not capable. It is the staff's opinion that it would be prudent for the CRBR applicants to investigate similar terraces in the vicinity of the site. This should be done by locating terraces in the region of the site where there is a high likelihood that they overlie faults. These terraces should be mapped and the cross-cutting relationships between them and the faults should be determined. Additionally, the applicants should map in cross-section the large terrace in the southeast section of the peninsula on which the site is located. Although no faults are recognized there, it is likely that minor tectonic structures will be found because of the proximity of the Copper Creek Fault. The staff regards this investigation as confirmatory and recommends that it not delay issuance of the construction permit.

(Staff Ex. 26, at 2-19).

85. Further on in the SER the Staff made specific recommendations for investigations to confirm that there are no active faults near or at the CRBR site. These were:

> (1) Investigate the high terrace in the southeast portion of the site peninsula to determine whether or not those deposits have been tectonically deformed. Because of the proximity of this terrace to the Copper Creek Fault, it is likely that structures are present there that are generically related to the Copper Creek Fault; and/or

(2) Locate sites in the subregion around the site where "datable" horizons appear to overlie mapped faults, and investigate those areas to determine whether or not the capping material is offset. Whether the applicants find an appropriately located terrace or not, the study should be documented in a manner similar to that described in Supplement 2 to PSAR Section 2.5 regarding sites where residual soil colluvium were photographed at projected outcrops of the Copper Creek and White Oak Mountain Faults.

(Staff Ex. 26, at 2-31 to 2-32)

86. The Staff has concluded on the basis of the information available that the faults at the site and in the region around the site are not capable as defined in Appendix A, 10 C.F.R. Part 100 (Staff Ex. 26, at 2-32). The Board agrees. As confirmation, the investigations the Staff has proposed should go forward, but they need not be completed before the issuance of the CP.

87. The Staff has carried out a detailed analysis of the seismology of the CRBR area (Staff Ex. 26, Section 2.5.2). It was well aware of the postulated seismogenic source zone capable of a large earthquake closer to the CRBR site than Giles County. This is the hypothetical source discussed in Appendix H of the SER (the USGS "Review"). The Staff discusses the SSE and the hypothetical local source in Section 2.5.2.3 of the SER (Staff Ex. 26).

88. The Staff's position is that the main evidence for the existence of the hypothetical structure is the apparent alignment of the relocated epicenters. Most of the other evidence relating to that hypothetical structure is equivocal or negative. The Staff also notes that several other alignments of the earthquake epicenters could be assumed and that large error ellipses are associated with several of the epicenters (Staff Ex. 26, at 2-29).

89. The Staff takes the position that the evidence for the hypothetical local seismogenic source is so weak that it does not warrant consideration as a capable fault under the meaning of Appendix A to 10 C.F.R. Part 100 (Staff Ex. 26, at 2-30), and the Board agrees.

90. The Staff also notes that additional studies are under way and that there is a well-distributed network of seismographic stations in the CRBR region and that more stations are planned for the future. This network of stations will allow epicentral locations to be made should earthquakes occur in the area. Given a sufficient number of well-located earthquakes, a definitive study of the hypothetical source, such as the study suggested by the USGS, would be possible. The Staff recommended that the Applicants keep informed on all seismological developments in the site region, since that information will have to be provided in the Final Safety Analysis Report (Staff Ex. 26, at 2-30).

91. After the close of the CP hearings in August, the Board issued an Order Requiring Disclosure of Seismic Information on September 15, 1983 (unpublished). That Order made reference to a September 7, 1983 letter from G.L. Chipman, Jr. (DOE) to J.N. Grace (NRC) [hereinafter, the September 7 letter], which stated that three faults had been discovered on the site during foundation excavation and concluded that none is capable within the meaning of 10 C.F.R. Part 100. Citing the parties' affirmative obligation to keep the Board fully and currently informed as to matters material and relevant to the adjudication, the Board ordered the Applicants and Staff to submit information of a kind and form sufficient to assure that the Board is fully informed of the details and analyses made by the Applicants and the subsequent review and conclusions of the Staff.

92. Subsequently, on September 21, 1983, the Applicants responded to the Board's Order and offered proposed findings of facts concerning the newly discovered onsite faults and two affidavits concerning those faults. These affidavits were from Peter J. Gross and Andrew P. Avel and are identified as Applicants' Exhibits 98 and 99, respectively.

93. The NRC Staff's Standard Review Plan (SRP), NUREG-0800, Section 2.5.3, Surface Faulting, defines the guidance for NRC Staff review of information in an Applicant's safety analysis report (SAR) related to the existence of a potential for surface faulting affecting a site. In accordance with 10 C.F.R. Part 50, Appendix A, General Design Criterion 2, and 10 C.F.R. Part 100 and Appendix A to that Part, SRP, Section 2.5.3, Subsection III contemplates that where a fault, the existence of which was previously unknown, is revealed in excavations during construction, the NRC Staff is to be notified by the Applicants as to when the excavations for critical structures are available for NRC inspection and when the detailed geologic maps to be used by the Staff while examining the excavations will be available for use.

94. In response to the SRP, the Applicants' Preliminary Safety Analysis Report (PSAR) stated in pertinent part:

> During geologic surface mapping for the CRBRP site investigation, a small tight fold and three minor shear dislocations were observed. Minor shear dislocations or offsets are interbed adjustments which formed contemporaneously with the regional thrust faults and represent displacements of traceable beds measured in terms of inches or at most a few feet. (Appl. Ex. 61, at 2.5-15)

> The minor structures observed at this site, including the bed slippage noted in the core, are common to the region and represent ancient adjustments. (Appl. Ex. 61, at 2.5-15a)

> There is no evidence for any capable faulting within 200 miles of the CRBRP site which may be of significance in establishing the Safe Shutdown Earthquake. (Appl. Ex. 61, at 2.5-25)
>
> No capable faults have been identified within five miles of the CRBRP site. (Appl. Ex. 61, at 2.5-27)
>
> As discussed in Section 2.5.1.2.4.3, small folds and minor dislocations are common in the region and are present at the site. However, the minor structures observed represent ancient adjustments. Results of laboratory and in-situ tests indicate that the rocks which occur within such zones are similar in character and competency to other sound rocks at the site. (Appl. Ex. 61, at 2.5-33)

An extensive inspection verification program will be established and implemented during construction, and will consist essentially of the following:

a. A qualified and experienced geologist will be on site immediately prior to the start of excavation and will monitor progress of the work until the base of the excavation has been prepared for the initial mat pour. He will report directly to the engineering and design organization and will be charged with the responsibility in the field of reviewing and commenting on the adequacy of the construction procedures proposed by the excavating contractor for ripping, blasting and removal of rock, inspecting exposed rock strata including side slopes and base of excavation and preparing a detailed geological map of the area. In addition to bedrock features, the map will include the relationship between overburden soils encountered in the excavation to structures in the rock. The map will be included in the FSAR.
b. A progress report will be submitted to the engineering and design organization on a weekly basis including photographs and detailed mapping of any significant geological features.
c. A consulting geotechnical review group consisting of specialists in rock mechanics and geology will inspect the excavation and report to the engineering and design organization on their findings at regular intervals, not exceeding 1 month.
d. If a geological discontinuity is noted, the engineering and design organization will be notified immediately and an inspection will be made by qualified personnel including members of the review board if considered necessary.

* * *

g. Formal approval of the prepared base of the excavation will be required by the review board prior to proceeding with the pouring of the mat.
h. The NRC will be kept fully informed of the progress of the excavation. In addition, they will be notified at least 1 week in advance of placing gunite, backfill or concrete on the exposed rock surface to permit a trip to be made to the site by a staff geologist if considered necessary.
i. The Nuclear Regulatory Commission will be notified if a geological discontinuity is noted.

(Appl. Ex. 61, at 2.5-40a-c.)

95. The Staff's Safety Evaluation Report (SER) stated in pertinent part:

> It is likely that many minor structures, including small faults, will be encountered during excavation at the site. The Applicants have committed to map the excavations and promptly notify the Staff of any faults discovered there so that field inspections can be made if necessary.
>
> Although the Staff expects additional small faults to be found, there is no reason to expect these faults to be younger than Late Paleozoic (more than 240 mybp).

(Staff Ex. 26, at 2-32.)

96. Pursuant to the commitments reflected in the PSAR and SER, the Applicants are in the process of mapping the CRBRP Category I excavations (Appl. Ex. 98, at 2). Consistent with the expectation expressly stated in the PSAR and SER, seven small faults or fault zones were found (*id.*; Appl. Ex. 99, at 3). Pursuant to Applicants' commitments, the Staff was promptly notified, and inspections were conducted by the NRC Staff (Appl. Ex. 98, at 3; Appl. Ex. 99, at 1).

97. Upon review of the relevant geological characteristics of the faults, the Applicants concluded that none is capable within the meaning of 10 C.F.R. Part 100 (Appl. Ex. 98, at 2-3; Appl. Ex. 99, at 5). The faults are not capable of producing differential ground displacements or generating earthquakes within the meaning of 10 C.F.R. Part 100, Appendix A, § IV(a) and (b) (Appl. Ex. 98, at 2-3; Appl. Ex. 99, at 5). The presence of such small faults was anticipated by the Applicants and Staff, and is consistent with previous CRBRP site investigations and observations for other nuclear power plant excavations in this region (Appl. Ex. 99, at 2-5; *see* Staff Ex. 26, at 2-31 to 2-32; Appl. Ex. 61, at 2.5-15). The geological evidence of faults in the excavation provides no basis for changing any geological or seismological conclusions in the PSAR or SER, including those relating to the seismicity model for the Appalachian province (Appl. Ex. 98, at 3; Appl. Ex. 99, at 2).

98. Pursuant to the commitments reflected in the PSAR and SER, the September 7, 1983 letter was submitted to the NRC Staff to provide the Staff with a written report of the geological findings from the excavation for the Staff's review (Appl. Ex. 98, at 3). The letter did not make references to the PSAR and SER discussion, which had anticipated the faults and established the procedure for mapping and reporting (Appl. Ex. 98, at 3; *compare* September 7 letter, with Appl. Ex. 61, at 2.5-15 to 2.15-40c; Staff Ex. 26, at 2-32).

99. On the basis of the foregoing, the faults observed in the CRBRP excavation were reported in accordance with the commitments stated in the PSAR and SER, were expected to be found, based upon prior geological investigations at the site and in the region, are not capable within the meaning of 10 C.F.R. Part 100, and do not affect any geological or seismological conclusions in the SER.

100. The Board concludes that the investigation of the geology and seismology of the site and the area by the Applicants and Staff has been thorough and that those studies and the data derived from them meet the applicable regulations. There are no outstanding facts or questions regarding the geology and seismology which should impede the issuance of a Construction Permit.

## B. Emergency Planning

101. The plume exposure pathway emergency planning zone (EPZ) established for the CRBRP site is about 10 miles in radius. This 10-mile radius encompasses portions of five counties: Roane, Anderson, Morgan, Loudon and Knox. The 10-mile EPZ is shown in SER Figure 13.2 (SER at 13-22). The ingestion pathway EPZ is an area of about 50 miles in radius and encompasses east-central Tennessee and a small portion of western North Carolina. The location of the CRBRP site in relation to counties and states is shown in SER Figure 13.3 (SER at 13-23). The site is located in Roane County in eastern Tennessee, approximately 25 miles west of Knoxville. It is bounded on the north by DOE's Oak Ridge Reservation (Staff Ex. 26, at 13-4).

102. We concluded earlier that an effectively coordinated site, state and local radiological emergency response plan can be achieved for the Clinch River site (17 NRC 158, 243; Finding 6), and now have the CRBR SER before us which includes a discussion of the Staff's review of emergency planning for CRBR (Staff Ex. 26, Section 13.3).

103. The Staff has reviewed the preliminary Clinch River Breeder Reactor Radiological Emergency Plan against the applicable regulations (10 C.F.R. Part 50, Appendix E, § II and 10 C.F.R. § 50.47(b)) and

concluded that the information was of sufficient depth and scope for the construction permit stage to indicate that the planning standards will be met in the final emergency plan. They also concluded that no special or unique circumstances had been identified which would preclude the development of adequate preparedness plans at the operating license stage of review. Moreover they found the plans to be in conformance with TMI Action Plan Item III.A.1.2 (Staff Ex. 26, Section 13.3.5). The Board has found no basis upon which to disagree with the Staff's conclusions.

104. In our Partial Initial Decision (LWA), we indicated our intention to explore in greater depth the emergency responses of the three major DOE facilities at Oak Ridge in the event of an emergency at CRBRP (17 NRC 158, 203; Finding 52). The three facilities of concern were the Oak Ridge Gaseous Diffusion Plant (ORGDP), the Oak Ridge National Laboratory (ORNL), and the Y-12 Plant.

105. The Applicants filed Exhibit 94 (Tr. 7979-8007) which responded to our intention, among other matters. This Exhibit describes the pertinent characteristics of each of the three DOE facilities and outlines the elements of their long-standing emergency plans (Tr. 7990-93).

106. The emergency response needs for CRBRP accident response of each of the three facilities differ somewhat. For each, however, sheltering or evacuation of nonessential personnel can be accomplished readily. The Y-12 Plant, which is the most sensitive facility of the three, is 9-11 miles distant from CRBRP, and that distance makes it highly unlikely that emergency evacuation would be needed. Nonetheless, should evacuation be called for, this could be accomplished promptly, as at the other DOE facilities, and a small security staff would be maintained there. This should not present a significant impediment to effective contingency planning since the lower doses at the more remote Y-12 Plant would allow for implementation of suitable protective measures (Tr. 7993-96).

107. Each of the DOE plants has extensive emergency planning, preparedness and response experience which provides an excellent basis for assuring compatibility with CRBRP emergency planning (Tr. 7996).

108. During the CP hearing, the Board explored some matters relating to evacuation during an emergency at CRBR. We learned that evacuation time estimates were based upon standard procedures, that the estimates were based upon the site-specific details of CRBR and its location; and that the loss of a bridge during or before evacuation would not change the time needed for evacuation. From the responses to our questions we conclude that the time estimate for evacuation of the EPZ — up to 9 hours — is reasonable and conservative, and we foresee no

combination of weather and road conditions so severe that evacuation would not be practical (Tr. 8008-18).

109. Among the concerns we identified before the CP hearing was Board inquiry Item 7 (*see* Appendix D) which requested the Applicants to discuss commercial and recreational river traffic (if any) from two points of interest as follows:

a) Practical methods of controlling same during off-normal plant conditions, and

b) The potential for hazardous cargo posing a threat to the CRBR.

110. The Applicants responded to this inquiry in their Exhibit 94 at 19-22 (Tr. 7997-8000). During periods of off-normal plant operations, commercial and recreational river traffic within the 10-mile EPZ will be controlled by the Tennessee Wildlife Resources Agency (TWRA), assisted as necessary by the U.S. Coast Guard. Upstream lockage through Melton Hill Dam will be controlled by U.S. Army Corps of Engineers (COE). Implementation criteria for this control will be described in the CRBRP and State of Tennessee Radiological Emergency Plan (Tr. 7997).

111. For the portion of the Clinch River adjacent to CRBR within the exclusion boundary, appropriate and effective arrangements will be made with TWRA and the U.S. Coast Guard to control traffic and provide for prompt warning and removal of persons present in the area. Implementation criteria for this control will be described in the CRBRP and State of Tennessee Radiological Emergency Plan (Tr. 7998).

112. No hazardous materials have been transported by barge past the site in the past, and none are anticipated in the future. According to records, steel is essentially the only commodity that has been shipped through the Melton Hill Lock since it was opened. The COE maintains two reporting systems which document the kinds of commodities shipped on the waterway: a vessel operations report identifying the commodities shipped which is required monthly from carriers, and a vessel log report (which also identifies commodities) that is submitted by the towboat captain to the lockmaster as each tow goes through the lock. In the foreseeable future, coal is the commodity having the greatest potential for increased movement through the Melton Hill Lock (Tr. 7998-99).

113. There are few potential industrial sites in the area large enough to accommodate an industry which might either use or produce commodities in large enough quantities to take advantage of water transportation. In the event that a new industry which would ship material by water develops, a barge terminal would have to be constructed. Plans for any such terminal must be submitted to the Corps of Engineers and TVA for review and approval. The information requirements for a

permit to construct a terminal include identification of materials to be shipped if they are known. If the materials are unknown, as in the case of a public terminal, the permit would be issued for the handling of non-hazardous materials only (Tr. 7999-8000).

114. In response to our question concerning whether the PAGs currently in use for evacuation planning purposes should be revised for use at CRBR to take account of those possible radioactive releases unique to CRBR, especially the actinide elements including plutonium (17 NRC 158, 174-75; *also see* Appendix D, Item 9), the Applicants filed Exhibit 94 (Tr. 7979) and the Staff filed its Exhibit 43 (Tr. 8575).

115. The Applicants concluded that the controlling doses for HCDAs are whole body and thyroid, and that other organs are less limiting. Therefore, PAGs for other organs are not required for emergency planning at CRBRP (Tr. 7984). This conclusion was reached by considering the radiological consequences of HCDAs which revealed that plutonium releases are not controlling (Tr. 7985-86), and that whole-body and thyroid doses are the limiting doses (Tr. 7986). They also derived ranges of PAGs for other organs by applying the ICRP-26 tissue weighting factors. This approach, like the foregoing, showed that the whole-body and thyroid doses would be controlling when compared to the PAGs for whole body and thyroid and the derived PAGs for other organs (Tr. 7986-89).

116. The Staff developed what it referred to as "analog PAGs and analog nonstochastic limits" to determine whether additional PAGs would be required for CRBRP (Tr. 8580-82). They concluded, based upon seemingly conservative assumptions, that the EPA PAGs were adequate for emergency planning purposes (Tr. 8585). They went on to examine the possible doses from four classes of HCDAs in three ways (Tr. 8585-87). This, as above, led the Staff to the conclusion that bone surface doses are not expected to be controlling for evacuation purposes in the event of an HCDA at CRBR (Tr. 8588). The Staff also noted that, in the event EPA's PAGs are revised or design modifications are made by the Applicants, the PAGs can be examined effectively at a later date (Tr. 8589-90).

## C. Environmental Matters

117. In our Partial Initial Decision (LWA), as required by 10 C.F.R. § 50.10(e)(2)(ii), we found reasonable assurance that the proposed site for the CRBRP is a suitable location for a reactor of the general size and type proposed (17 NRC 158, 256), and we made all of the environmental findings required by 10 C.F.R. § 51.52(b) and (c) which are needed

prior to the issuance of a construction permit (17 NRC 158, 242-54). Before the LWA evidentiary hearings were held, however, we indicated that, although Intervenors' environmental contentions would be fully resolved at the LWA stage, their finality would have to await the conclusion of the CP stage, since information received at the CP stage might affect the findings (Transcript of April 20, 1982 Conference with Counsel at 510-15), but no such information was received.

THE ATOMIC SAFETY AND LICENSING BOARD

Dr. Cadet H. Hand, Jr.
ADMINISTRATIVE JUDGE

Gustave A. Linenberger, Jr.
ADMINISTRATIVE JUDGE

Marshall E. Miller, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 20th day of January 1984.

[Appendices A, B, and C have been omitted from this publication, but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

## APPENDIX D – BOARD AREAS OF INTEREST*

1. In its Safety Goal Development Program announcement (48 Fed. Reg. 10,772 (1983)) the Commission stated that during the 90-day period (ending June 8, 1983) for public comment on the proposed evaluation plan "it is expected that preliminary information on new radiologi-

*Contained in the Board's Notice of Construction Permit Evidentiary Hearing, dated May 24, 1983.

cal source terms will become available . . ." (*id.* at 10,778). The Staff is requested to advise whether that information will be evaluated for any impact on this proceeding, and the reason for its answer.

2. As regards fuel performance, to date the use of the term "failed fuel" has not consistently permitted delineation of the various failure modes that might have been alluded to (*e.g.*, clad perforation, fission product leakage, clad bulging or rupture, melting of fuel pellets, etc.). The Applicants are requested to summarize the anticipated performance of the CRBR fuel associated with normal operation and accidental transients, describe various failure modes that must be dealt with, identify any operational limits (*e.g.*, maximum linear heat generation rates, maximum cladding hot spot temperature, etc.) to be imposed, and to review the basis for confidence (*e.g.*, supportive evidence) that the proposed fuel behavior characteristics will be realized.

3. Avoidance of primary coolant pipe rupture seems to depend in part upon the fact that coolant temperature is well below its boiling temperature and that coolant pressure is near atmospheric pressure (<10 atm.). Applicants are requested to present a technical summary of how these coolant characteristics will result in a reduced likelihood of pipe rupture in piping designed for CRBR use.

4. Applicants are requested to explain how the CRBR will be configured to assure that convective circulation of the sodium coolant will be available to prevent fuel damage, if needed. This explanation should reference any supportive experimental or operational evidence. The Staff is requested to advise the Board whether it accepts convective circulation as a viable mechanism for fuel protection, and the reason for its answer.

5. In the area of quality, the Applicants are requested to explain whether (and/or how) differing functional levels of effort will be applied, depending upon whether a component or system is necessary for safety, important to safety, or not safety-related. The divisions of authority and functional responsibilities for quality assurance and quality control amongst the various contractors and the Applicants should be discussed with emphasis on how the management of the various CRBR contractor fabrication and construction efforts will be coordinated to assure the minimizing of QA and QC oversights, especially where interfacing is involved. Applicants are also requested to describe what efforts will be undertaken to ensure that accurate as-built plans and specifications will be available when needed, if the CRBR is constructed.

6. The SER discussion of quality seems to emphasize quality assurance and the various separate contractor organizations that will implement it. Does the Staff consider that QC responsibilities and activities

are separate from QA or an integral part thereof? The Staff is requested to discuss its answer to this question and to explain briefly how it will monitor QA and QC efforts for adequacy.

7. Applicants are requested to discuss commercial and recreational river traffic (if any) from two points of interest:

(a) Practical methods of controlling same during off-normal plant conditions, and

(b) The potential for hazardous cargo posing a threat to the CRBR.

8. Applicants are requested to discuss the design characteristics of the containment/confinement structures and the steam generator, with respect to challenges to those structures arising from transient (or accident) induced overpressure and overtemperature conditions. This discussion should address any engineered safety systems or components that will be relied upon for protection (*e.g.*, containment shell cooling), and should reference supportive test or operational experience.

9. The Staff's attention is directed to the discussion of protective action guidelines (PAGs) at 17 NRC 174-75 of the Partial Initial Decision of February 28, 1983. The Staff is requested to address the question of whether a PAG revision for the CRBR should be made, and to explain its answer.

10. The Staff's testimony at Tr. 3694 anticipates the need for further research and development on measurement capabilities to achieve DOE's goals for material control and accountability at the DRP. The Staff is requested to explain whether this additional effort is currently under way or definitively planned for the future, and the extent to which it is critical to the effectiveness of CRBR fuel safeguards measures.

11. In discussing the energetics of accidents beyond design basis, the Staff offers the statement that there will be an "isentropic expansion yield to one atmosphere" (NUREG-0968, Vol. 2, at A.2-5). The Staff is requested to discuss briefly what is the physical significance of this statement and the extent to which it contributes to any conservatism in the analyses of energy releases. Phenomenologically, how has the Staff satisfied itself that "approximately 2550 MJ would be required to produce a slug impact kinetic energy close to the head design capability of 75 MJ" (*ibid.*).

12. NUREG-0968 contains many references to items that are to be resolved at the OL review stage. In view of the apparently advanced stages of hardware design and procurement currently in being, the Board is concerned that said OL review (assuming a CP issues) may require substantive changes of a costly and time-consuming nature, or in the alternative, result in a compromise of performance safety. The Staff

is requested to offer comments upon this situation and to provide whatever insights it can now offer for avoiding such problems.

13. With respect to the fuel system, the Staff has identified certain operational fallback positions potentially available to mitigate unresolved problems (NUREG-0968, Vol. 1, at 4-47, 4-48). The Staff is requested to discuss briefly the extent, if any, to which invoking such operational fallbacks might compromise the achievement of CRBR programmatic objectives.

14. Operation with leaking fuel pins could conceivably offer the opportunity for these pins to "inhale" some amount of sodium whenever the reactor is shut down. Should this occur, subsequent return to operation at power might then result in a significant increase in pellet-to-cladding gap conductance with an attendant off-normal performance of the fuel. The Staff is requested to comment upon whether it sees this as a problem requiring resolution and the reasons for its answer.

15. The Applicants have proposed a reliability assurance program that focuses primarily on plant protective systems. The Board requests Applicants to address the question of whether said program will (or ought to) take account of findings derived from the CRBR quality assurance program, and if so, describe the administrative mechanism envisaged to accomplish this.

16. The SER discusses the impact of aerosol behavior on containment shell cooling. The Staff is requested to comment on whether changing concrete aggregate from calcitic to dolomitic limestone could significantly alter the behavior of the aerosols, and explain the basis for the answer.

17. What is the status of the Staff's review of, and what is the Staff's position with respect to, "The Eight Areas of Concern" listed in Section I, Table II of NUREG/CR-3224?

## APPENDIX E

## LIMITED APPEARANCE STATEMENT OF DR. THOMAS B. COCHRAN REGARDING ISSUES RAISED IN THE CONSTRUCTION PERMIT PROCEEDING

Pursuant to 10 C.F.R. § 2.715, and in accordance with the Board's order of June 29, 1983, Dr. Thomas B. Cochran hereby submits a limited appearance statement on behalf of the Natural Resources Defense Council, Inc., and the Sierra Club, regarding several issues raised by the

Board for resolution in the upcoming CRBR construction permit hearings.

## I. The Radiological Consequences of a CRBR Core Disruptive Accident

Staff has evaluated the radiological consequences of Applicants' postulated CRBR core disruptive accident (CDA) scenarios and reported the results in Appendix A.5 of the March 1983 Safety Evaluation Report (SER) and the May 20, 1983, SER Supplement No. 2. According to Staff, the evaluation used "realistic (albeit conservative) assumptions" (SER Suppl. No. 2 at A.5-1), including 50% X/Q meteorology (SER Suppl. No. 2 at A.5-3). The low population zone (LPZ) thyroid dose was reported to be 192 rem (SER Suppl. No. 2 at A.5-4) using thyroid dose conversion factors taken from TID 14844 (NRC Staff's Response to Intervenors' Third Set of Construction Permit Interrogatories and Request to Produce to Staff, Response to Interrogatory 1(d), p. 2, May 20, 1983).

Staff claims the 192 rem thyroid dose at the LPZ

> gives the staff confidence in the applicants' claim that the critical organ dose for a CDA would be within the 10 C.F.R. Part 100 dose guidelines

and that

> the comparison to 10 C.F.R. dose guidelines is made here to provide perspective regarding the relative severity of the CDA consequences and to provide assurance that if such an event were to occur that adequate accommodation has been provided to limit the consequences of such an event, so that doses would not exceed dose *guidelines in 10 C.F.R. 100.*

(SER Suppl. No. 2 at A.5-1)

I dispute these claims on several counts:

First, Staff has calculated the thyroid dose for an adult, but the infant thyroid should be considered the critical organ of interest. Infants can be expected to receive a thyroid dose twice that of an adult or, in this case, approximately 400 rem — some 100 rem (or one-third) higher than the 300 rem guideline value for thyroid used by Staff. Evidence for this is as follows:

> The thyroid dose conversion factor for inhalation of I-131 given in TID 14844 at p. 25 is $1.48 \times 10^6$ rad/Ci inhaled (= $1.48 \times 10^{-3}$ mrad/pCi inhaled), the same value as that given for an adult in NUREG-0172 at Table 8, p. 2 of 4. Likewise, the breathing rate used in TID 14844 at p. 23 is the value for adults, 20

$m^3/day$ (= 7300 $m^3/yr$), as indicated in NUREG-0172, Table 8-4, at p. B-4.

The ratio (infant dose/adult dose) for inhalation of I-131 can be calculated from data in NUREG-0172 (at Table 5, p. 2 of 4; Table 8, p. 2 of 4; and Table B-4, p. B-4)) as follows:

$$\frac{D_{131}\text{ (infant)} = 1.06 \times 10^{-2}\text{ mrad/pCi} \times 2045\text{ m}^3\text{/yr}}{D_{131}\text{ (adult)} = 1.49 \times 10^{-3}\text{ mrad/pCi} \times 7300\text{ m}^3\text{/yr}} \simeq 2$$

Similar calculations can be made for other halogen isotopes.

Second, for purposes of judging the adequacy of CRBR containment to mitigate CDAs, Staff uses as a benchmark the 10 C.F.R. 100 dose guideline values developed for siting analysis (SER Suppl. No. 2 at A.5-1). In the 10 C.F.R. 100 site suitability analysis at the CP licensing stage, Staff requirements are to reduce the guideline values by approximately a factor of two to account for

> uncertainties in final design detail and meteorology and new data and calculational techniques that might influence the final design of engineered safety features or the dose reduction factors allowed for those features.

NRC Staff's Supplemental Answers to Intervenors' Twenty-Sixth Set of Interrogatories to Staff, at pp. 19-20. Staff, for example, uses a thyroid dose guideline of 150 rem at the CP stage, rather than 300 rem used at the OL stage. (1982 Site Suitability Report (SSR) at p. III-9.) Staff fails to apply the same logic — although it applies equally — to the CDA analysis, realizing of course that, if they did so, the calculated "realistic" CDA adult thyroid dose of 192 rem would exceed the 150 rem guideline value. Staff's failure to apply the same logic in the two cases is arbitrary and serves only to ensure licensability of the current CRBR design rather than to protect the public health.

Third, I believe the estimated severity of a CDA at CRBR, assuming "realistic (albeit conservative)" conditions, namely, thyroid doses of 192 rem to adults and 400 rem to infants at the LPZ boundary, is excessive and should not be tolerated for CRBR, or for any reactor. In effect, this is also a challenge to Staff's use of 300 rem (to the adult thyroid) as a benchmark to judge the adequacy of CDA mitigation based on a *realistic* CDA scenario (*cf.*, SER Suppl. No. 2 at A.5-1).

The basis for this view is a direct comparison of these thyroid doses against the observed medical findings in a Marshall Islands population accidentally exposed to fallout from the 15-megaton thermonuclear device (code named Bravo) tested at Bikini Atoll on March 1, 1954. A summary of the thyroid abnormalities that have appeared as of 1981 are reported in the attached table (Attachment 1) taken from Robert A. Conard, M.D., *et al.*, "Review of Medical Findings in a Marshallese Population Twenty-Six Years After Accidental Exposure to Radioactive Fallout," Brookhaven National Laboratory, BNL 51261, Jan. 1980, Table 1 of Chapter IX, p. 59. These data speak for themselves. I note only that exposure occurred only 26 years ago, that many of the victims are still in their early years, and that additional thyroid abnormalities can be expected as the survivors grow older.

## II. Combined Probability and Consequences of CRBR Core Disruptive Accidents

In the LWA proceeding, Intervenors Natural Resources Defense Council, Inc., and the Sierra Club presented an affirmative case regarding their Contentions 1 and 3, namely that Staff's and Applicants' analysis of the consequences of CDAs, coupled with Staff's analysis of the probability of a CDA (Appendix J of the CRBR FSFES; Staff Ex. 8), demonstrate that the Commission Standard Review Plan criterion for identification of design basis events is not met and consequently the CDA should be a containment DBA. (*See* Intervenors' Proposed Findings of Fact for the LWA-1 Proceeding, January 24, 1983, at ¶¶ 1-23). I hereby reaffirm and incorporate that testimony in my statement today and request that the Board take that evidence into account in the current proceeding.

Using CRBR design-specific information generated by Applicants and Staff, I am able to provide additional evidence in support of our earlier claim. Any of the documents cited below will be made available to the Board upon request.

Pursuant to Intervenors' CP discovery request, Applicants made available the bulk of Applicants' CRBR probabilistic risk assessment (PRA) analyses that were by Board Order ruled beyond the scope of the LWA-1 proceeding (*see* Natural Resources Defense Council, Inc., and the Sierra Club First Set of Construction Permit Interrogatories and Request to Produce to Applicants, April 7, 1983; Letter from Thomas A. Schmutz to Dr. Thomas B. Cochran, June 20, 1983, with enclosure). Among the documents produced was EG&G Idaho, Inc., Wood-Leaver and Associates, Inc., and Fauske & Associates, Inc. "Clinch River Breeder Reactor Plant Probabilistic Risk Assessment — Phase I Main Report," EGG-EA-6162, January 1983. (Selected pages of this voluminous work are attached as Attachment 2.) I wish to call attention to two aspects of this work. First, as evidenced by the Abstract (reproduced in

Attachment 2), this PRA has as its overall objective a "realistic evaluation of the risk" associated with CRBR, with the caveat that, since the entire PRA must await Phase II, the results of Phase I must be interpreted cautiously.

The second aspect of this work that I call to your attention is its estimate of the cumulative probability of dominant Core Damage Sequences (*i.e.*, CDAs), of 1.1 x $10^{4}$/yr [sic] (*see* Attachment 2, p. 8-11), which is dominated by loss of offsite power scenarios (Attachment 2, p. 11-2).

In sum, whereas Staff in Appendix J of the Final Supplement to the CRBR Final Environmental Statement (FSFES) estimated that a "conservative," or upper bound, estimate of a CDA at CRBR was $10^{-4}$/yr, Applicants' consultants calculate the same frequency based on "realistic," as opposed to Staff's conservative, assumptions. The sensitivity analysis performed by Applicants' consultants suggests the upper limit in the probability of a CDA at CRBR may be even higher than $10^{-4}$/yr (Attachment 2, pp. 11-8 and 11-9).

One can combine the Applicants' consultants' best estimate of CDA frequency of $10^{-4}$/yr with Staff's "realistic (albeit conservative)" estimate of the thyroid dose at the LPZ boundary of 192 rem to adults (400 rem to infants) in order to compare the results against the Commission's Standard Review Plan guidance for identifying DBAs.[1] Certainly by this test the CDA ought to be a containment DBA; the probability of exceeding 10 C.F.R. 100 guidelines is approximately three orders of magnitude too high to exclude the CDA from the containment DBA envelope.

I wish to anticipate several responses to this observation.

First, if the Board were to reject my view that the appropriate CP thyroid dose guideline is 150 rem rather than 300 rem, and were also to reject the argument that the infant thyroid dose should be examined as the critical organ dose, the Board might conclude that the calculated 192 rem to the adult thyroid is well within the 300 rem guideline. The response to this is straightforward. The 192 rem estimate is based on the median of some 8600 X/Q values in all cardinal directions (*i.e.*, 50% X/Q). At a somewhat higher X/Q, the adult thyroid dose at the LPZ boundary would exceed 300 rem. I do not know at what X/Q percentage this would occur since the X/Q spectrum is not reported. However, Staff

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[1] The Commission's Standard Review Plan for light water reactors (Staff Ex. 6, at 2.2.3-2) states:

> [T]he identification of design basis events resulting from the presence of hazardous materials or activities in the vicinity of the plant is acceptable *if the design basis events include each postulated type of accident for which the expected rate of occurrence of potential exposure in excess of the 10 C.F.R. Part 100 guidelines is estimated to exceed the NRC staff objective of approximately $10^{-7}$ per year*. . . . [T]he expected rate of occurrence of potential exposures in excess of the 10 C.F.R. Part 100 guidelines of approximately $10^{-6}$ per year is acceptable if, when combined with reasonable qualitative argument, the realistic probability can be shown to be lower. (Emphasis added.)

could readily produce this figure. For purposes of argument, I will assume that, at the 80% X/Q level, the adult thyroid dose would in fact exceed the 300 rem guideline. If so, then the dose guideline value would be exceeded for 20% of all CDAs, or 0.2 x $10^{-4}$/yr, still well above the $10^{-7}$/yr requirement for excluding CDAs from the DBA envelope.

A second response is likely to be that the quantitative probabilities, or the PRA results, are highly uncertain and therefore should not be used as a basis for determining the CDA envelope in lieu of the "judgmental" approach taken by Staff. This is a correct response to the wrong question. PRAs are indeed highly uncertain, and their primary function should be to identify previously unrecognized risks to health and safety. As a prudent health and safety practice, one should use great caution in applying highly uncertain PRA results to argue *against* the application of additional safety equipment or procedures, such as excluding CDAs from the DBA. Prudence dictates, however, that, if the PRA results support the application of additional equipment or procedures, such as including CDAs within the DBA envelope, then one should be extremely apprehensive about rejecting the results in favor of higher public health risks. In other words, prudence dictates an asymmetry in the application of PRA.

Staff is taking just the reverse approach to public safety in the case of the CRBR. Staff has applied their PRA results in the LWA-1 proceeding to "demonstrate" that CRBR risks are comparable to LWR risks and to eliminate alternative sites, but apparently Staff does not want to apply the PRA results to test whether the CDA should be included in the DBA envelope, realizing that to do so would force a safer design or a rejection of the CRBR site.

PRA should be used as a check on the "judgmental" approach taken by Staff. In this case, Staff's conclusions have been checked and found not to wash.

## III. CRBR Site Suitability

The Board resolved Intervenors' Contention 2 for purposes of the LWA-1 by finding that

> The containment/confinement design of the CRBRP has been shown capable of performing its intended function to accommodate all credible design base threats and hold doses to the general public below guideline values, without requiring any technological innovations. . . . The Staff's final position on the adequacy of the containment/confinement design will be presented when its SER is published. . . .

> The Board is not persuaded by the evidence of record to date . . . that the CRBR will be built and operated in a manner that precludes the necessity for considering CDAs within the design basis. . . . [W]e foresee a heavy burden upon these parties at the construction permit phase of evidentiary hearings to provide sufficient evidence to permit a resolution of this question.

ASLB Partial Initial Decision (Limited Work Authorization), February 28, 1983, at p. 22 [17 NRC 171]. At the CP stage, the Board must resolve these open issues and make a finding, based on reasonable assurance, that *the proposed facility* can be constructed and operated without undue risk to the health and safety of the public taking into consideration 10 C.F.R. Part 100. 10 C.F.R. § 50.35(a). I therefore offer the following new information for further consideration by the Board of one of the issues raised under Contention 2.

In the LWA-1 proceeding, Intervenors argued that, in assessing the suitability of the CRBR site, the effects of the containment vent/purge system on offsite doses must be considered. Had the effects of the vent/purge system been incorporated into Staff's and Applicants' calculation of offsite doses in the site suitability analysis, Intervenors demonstrated that the bone surface doses would have exceeded the 10 C.F.R. 100 dose guideline values.

Through discovery in separate litigation, it has come to my attention that there is a precedent for incorporating the effect of the containment vent system in 10 C.F.R. 100 analyses. This precedent, which lends further weight to the arguments made by Intervenors in the LWA-1 proceeding, is found in analyses DOE has performed for the airborne activity confinement systems used by DOE production reactors at the Savannah River Plant (SRP).

There are both differences and similarities between the confinement system of SRP production reactors and the containment/confinement system of CRBR. I will not elaborate on the containment/confinement, annulus filtration, and vent/purge systems of CRBR, since these systems are known to the Board. The SRP reactors are somewhat different in that they do not have a dual containment/confinement building and consequently do not have an annulus filtration system. The SRP reactors do not have a containment building capable of withstanding 10 psi (design pressure); rather the reactor is housed in a reactor building that can be sealed to withstand a 2 psi differential. The two systems are similar in that they both utilize a filtered vent system to mitigate offsite doses in the event of CDAs (core melting). Both filter systems filter halogens and particulates, but are ineffective with regard to noble gases. The SRP reactor Airborne Activity Confinement Systems are described further in J.A. Smith, *et al.*, "Safety Analysis of Savannah River Production Reactor Operation," DuPont Savannah River Laboratory, DPSTSA-100-1, Rev. 12/76, pp. IV-43 to IV-49 (Attachment 3); and Memorandum from F.F. Merz to S.P. Tinnes, "Airborne Activity Confinement System Base Case Design Basis Accident," July 19, 1979 (Attachment 4).

As can be seen from the Merz Memorandum (Attachment 4), DOE has selected as a DBA for the Airborne Confinement System a fission product release consistent with the 10 C.F.R. 100 site suitability source term for LWRs, namely, a full core meltdown with release of 100% noble gases and 50% of the halogens to the reactor building.[2]

Since DOE reactors are not licensed by the Commission, there is no requirement that they meet the requirements of 10 C.F.R. 100, and in fact, as evidenced by the Safety Analysis Report (DPSTSA-100-1, Rev. 12/76), they do not; but that is an issue for another proceeding. Nevertheless, it is clear from several SRP documents made available to me, including DPSTSA-100-1 and the Merz Memorandum, that DOE conducts 10 C.F.R. 100 analyses to assess the adequacy of the SRP reactor confinement system and alternative containment concepts. In each offsite dose analysis of the production reactor airborne confinement system, *including the design basis accident based on use of the 10 C.F.R. 100 site suitability source term,* the effect of the filtered vent system is treated in the offsite dose calculation.

It would be interesting to know, and the Board might wish to determine, whether a second precedent for inclusion of the filtered vent system in the 10 C.F.R. 100 dose calculations is found in the site suitability analysis of the Ft. St. Vrain reactor, which, I am told, uses a filtered vent/confinement approach.

In the LWA-1 proceeding, Intervenors indicated that the record was inadequate to determine the effect of including the vent/purge system in the 10 C.F.R. 100 CRBR site suitability source term analysis on organ doses such as bone surface, lung, thyroid, and liver. With Staff publication of its "realistic" CDA dose results in the SER Suppl. No. 2, additional evidence can now be provided relative to the effect of inclusion of the vent/purge system on the SSST thyroid and bone surface dose. I will examine the effect on the thyroid dose first, followed by the bone surface.

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[2] It is perhaps worth noting that for SRP production reactors the DBA for the emergency core cooling system is different from the DBA for the Airborne Activity Confinement System. For purpose of analysis of the operation of the emergency core cooling system for SRP reactors, the DBA is a double-ended pipe break in one of the six primary lines supplying heavy water to the reactor plenum with the simultaneous failure of a single active component, a second emergency cooling water addition system. Under these conditions, core damage is limited to 1%. (*See* J.W. Joseph and R.C. Thornberry, "Analysis of the Savannah River Reactor Emergency Core Cooling System," DuPont Savannah River Laboratory, DPST-70-463, October 1970).

In the 1982 SSR, the LPZ thyroid dose was estimated to be 7 rem, with no consideration given to the effect of containment venting, but with other parameters conservatively chosen, including the following 95% X/Q values:

| | 95% X/Q (sec/m$^3$) |
|---|---|
| 0-8 hours | $1.2 \times 10^{-4}$ |
| 8-24 hours | $8.4 \times 10^{-5}$ |
| 24-96 hours | $3.9 \times 10^{-5}$ |
| 96-720 hours | $1.4 \times 10^{-5}$ |

(Staff Ex. 1, p. III-11.)

In the SER Suppl. No. 2, the adult LPZ thyroid dose was estimated to be 192 rem, *with* the vent/purge system modeled, and with other parameters "realistically (albeit conservatively)" chosen, including the following 50% X/Q values:

| | 50% X/Q (sec/m$^3$) |
|---|---|
| 0-8 hours | $1.1 \times 10^{-5}$ |
| 8-24 hours | $1.0 \times 10^{-5}$ |
| 24-96 hours | $8.0 \times 10^{-6}$ |
| 96-720 hours | $5.7 \times 10^{-6}$ |

(SER Suppl. No. 2, p. A.1-3.)

The ratios of the 95% X/Q values (used in the SSST analysis) to the 50% X/Q values (used in the "realistic" analysis) are:

| | Ratio $\frac{95\% \; X/Q}{50\% \; X/Q}$ |
|---|---|
| 0-8 hours | 11 |
| 8-24 hours | 8.4 |
| 24-96 hours | 4.9 |
| 96-720 hours | 2.5 |

Staff's computer modeling output indicates that 153 rem of the 192 rem total LPZ thyroid dose occurs between 24 hours and 96 hours, where 24 hours is the time venting commences; the additional 39 rem occurs between 96 hours and 130 hours, where 130 hours is the time of sodium pool dryout. (NRC Staff's Response to Intervenors' Third Set of Construction Permit Interrogatories and Request to Produce to Staff (May 20, 1983); Staff's Computer Run for Halogens, Noble Gases, and Sodium, dated 3/19/83.) Thus, if the "realistic" assumptions were selected but 95% X/Q values were used instead of 50% X/Q values, the adult thyroid dose would be determined by multiplying the thyroid dose for each of the two time periods of interest (24-96 hours and 96-720 hours) by the 95%/50% X/Q ratio for that time period, and adding the two doses together. The result would be

$$153 \times 4.9 + 39 \times 2.5 = 850 \text{ rem},$$

over five times the 10 C.F.R. 100 CP guideline value for thyroid. The infant thyroid dose would be 1700 rem, or over 11 times the guideline values.

Turning now to the bone surface dose, in the 1982 SSR the Staff estimated the LPZ bone dose at 9 rem, with no consideration given to the effect of containment venting, but with other parameters conservatively chosen. In the SER Suppl. No. 2 (p. A.5-4), the bone dose was estimated to be 8 rem, with the vent/purge system modeled and other parameters "realistically (albeit conservatively)" chosen.

Staff's computer modeling output indicates that 5.7 rem of the 9 rem total LPZ bone dose occurs between 24 and 96 hours, with the additional 1.9 rem between 96 and 130 hours (Staff Computer Run for Solids Only, dated 3/11/83). Thus, if the "realistic" assumption were used, but 95% X/Q values were used instead of 50% X/Q values, the LPZ bone dose would be calculated using the same technique as the thyroid dose, thus yielding

$$(5.7)(4.9) + (1.9)(2.5) = 33 \text{ rem}.$$

The bone surface dose is three times this value, or 100 rem.

In their "realistic" CDA analysis, Staff assumed 0.16% of the plutonium is initially available to the sodium pool in the reactor cavity, whereas in the 10 C.F.R. 100 site suitability analysis Staff makes the more conservative assumption that 1% of the plutonium is available to the containment building. If the plutonium available in the pool is increased from 0.16% to 1%, the bone surface dose is increased from 100 rem to 625 rem, well above the 10 C.F.R. 100 bone surface guideline value of 150 rem used by Staff at the CP (1982 SSR, Staff Ex. 1, p. III-9).

In the LWA-1 proceeding, Intervenors noted that other corrective factors should be applied as well, including:

| Factor | Basis |
|---|---|
| 4.3 | to correct for potential use of plutonium from high burnup spent fuel. |
| 1.5[3] | to convert from a 50-year dose commitment to an 80-year dose commitment. |

Staff and Applicants argue that because in their judgment the CDA is not a DBA they are free to ignore the vent/purge system in the 10 C.F.R. 100 site suitability analysis, since no "credible" accident would ever challenge the containment and require activation of the vent system. If the CDA is a DBA, then of course this argument has no merit, and the CRBR site is not suitable for the CRBR containment design.

Even if the Board concludes the CDA is outside the DBA envelope, we believe Staff's and Applicants' argument is still incorrect. In the 20+ year history of 10 C.F.R. 100, it has always been assumed that, for purposes of assessing whether 10 C.F.R. 100 requirements are met, one should assume a gross fission product release following full meltdown (*cf.*, TID 14844, p. 10) and the use of substantial conservatisms in the analytical methodology for estimating offsite doses. Staff's and Applicants' approach — to ignore the concomitant effects of the core melting — is simply ludicrous. Ignoring the implications of fuel melting (*i.e.*, failure to model the vent/purge system), rather than conservatively treating them, results in site suitability source term thyroid and bone surface doses that are some two orders of magnitude less than the dose associated with the most benign full core melt event "realistically" calculated.

When the site suitability source term analysis is performed properly and 10 C.F.R. 100 requirements are not met, there is simply no basis for granting a CP for this reactor design at the CRBR site.

[Affidavit of Dr. Thomas B. Cochran and Attachments 1, 2, 3, and 4 have been omitted from this publication but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

3 If the Board chooses to follow the EPA, NRC, and DOE precedent of using 70-year rather than 80-year dose commitment, this factor would be 1.35.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**THE ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Marshall E. Miller,** Chairman
**Dr. Emmeth A. Luebke**
**Dr. Jerry Harbour**

| | |
|---|---|
| **In the Matter of** | **Docket No. 50-272-OLA** |
| **PUBLIC SERVICE ELECTRIC & GAS COMPANY (Salem Nuclear Generating Station, Unit 1)** | **January 25, 1984** |

# ORDER DISMISSING PROCEEDING

The Commission published a notice of consideration of issuance of an amendment to the Salem facility operating license and a proposed no significant hazards consideration determination, and an opportunity for a hearing, on September 21, 1983 (48 Fed. Reg. 43,113-14). This notice concerned a requested amendment of the technical specifications to permit a 7-month extension of time for the performance of a containment integrated leak rate type A Test (Technical Specification 4.6.1.2a).

The State of Delaware filed a timely petition for leave to intervene and request for hearing on October 21, 1983. The NRC Staff issued the requested amendment on October 31, 1983. By an Order entered November 17, 1983 (unpublished), the Board granted the intervention petition, and directed the State of Delaware to file a supplement to its petition setting forth at least one contention cognizable under 10 C.F.R. § 2.714(b), by January 4, 1984. That date was extended by two weeks upon motion.

On January 20, 1984, the State of Delaware filed a motion to withdraw its petition, no supplement having been filed. That motion is granted and the intervention petition is withdrawn. Inasmuch as there are no other intervention petitions or requests for hearing in accordance with the Commission's notice, the matter is uncontested, and the adjudicatory proceeding is therefore DISMISSED.

It is so ORDERED.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Marshall E. Miller, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 25th day of January 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Morton B. Margulies,** Chairman
**Ernest E. Hill**
**Dr. Paul W. Purdom**

**In the Matter of** **Docket No. 50-412**
**(ASLBP No. 83-490-04-OL)**

**DUQUESNE LIGHT COMPANY,** ***et al.***
**(Beaver Valley Power Station, Unit 2)** **January 27, 1984**

In this Report and Order the Licensing Board concludes that a hearing is not required and dismisses the proceeding.

## FLOODPLAIN MANAGEMENT: EFFECT OF EXECUTIVE ORDER

As an independent regulatory agency, the Nuclear Regulatory Commission is not subject to the requirements of Exec. Order No. 11,988, Floodplain Management, 42 Fed. Reg. 26,591 (1977).

## CONTENTIONS: PREMATURITY

The Licensing Board cannot decide the validity of actions that are yet to happen. Speculation concerning what the NRC Staff may do in an environmental impact statement that has not been issued does not provide an adequately specific basis for an admissible contention.

STANDING: REPRESENTATIONAL

In order for an organization to obtain representational standing on the basis of the interests of a member, it must be established that the member has authorized the organization to represent his interests in the proceeding. It is unwarranted for the Licensing Board to infer such authorization when the affidavit of the member is devoid of any statement that he wants the organization to represent him.

INTERVENTION: INTERESTED STATE

The filing and acceptance of the petition of the State of Pennsylvania pursuant to 10 C.F.R. § 2.715(c) permits it to participate in the adjudicatory hearing only if one is held. When no petitioner has submitted a litigable contention so as to necessitate the holding of a hearing, the filing and acceptance of the Pennsylvania petition to participate under the provisions of § 2.715(c) does not trigger a hearing.

INTERVENTION: ADMISSIBLE CONTENTION

When none of the concerns sought to be litigated by a petitioner for intervention are within the scope of an operating license proceeding, the petitioner has failed to submit an admissible contention, and his petition for intervention will be denied.

# REPORT AND ORDER
## ON SPECIAL PREHEARING CONFERENCE HELD PURSUANT TO 10 C.F.R. § 2.751a

### DEVELOPMENT OF THE PROCEEDING

Following the publication of a Notice of Opportunity for Hearing on June 1, 1983, for the captioned operating license application proceeding, timely petitions to intervene were filed by the Commonwealth of Pennsylvania (Pennsylvania) seeking to participate as an interested State under 10 C.F.R. § 2.715(c) and by Ohio Citizens for Responsible Energy (OCRE), Environmental Coalition on Nuclear Power (ECNP), George S. White and Ralph P. Walker, pursuant to 10 C.F.R. § 2.714.

Applicants, represented by Duquesne Light Company (Duquesne or Applicant) and Nuclear Regulatory Commission Staff (Staff) filed responses alleging omissions and deficiencies in the petitions (except that of Pennsylvania) including the need to present allowable contentions as prescribed in 10 C.F.R. § 2.714(b).

On August 4, 1983, we ordered the holding of a special prehearing conference to resolve, *inter alia,* contested issues of standing and to pass upon proposed contentions, with representations of petitioners on those matters to be submitted by supplemental petition by September 9, 1983. None were submitted by Walker and Pennsylvania. There had been no objection to the latter's participation. Also, a late-filed petition to intervene was received from William A. Lochstet.

Prior to convening the special prehearing conference at Pittsburgh, Pennsylvania on October 12, 1983, White and Lochstet formally withdrew from further participation. Appearing at the conference as ordered were Applicant, Staff, Pennsylvania, OCRE and ECNP. Walker did not attend. A review was conducted of the petitions to intervene of OCRE and ECNP.

An analysis of the petitions filed and their disposition follow.

## DETERMINATION OF THE OCRE PETITION

Petitioner OCRE is an unincorporated association, composed of approximately 50 Ohioans concerned with the health, safety, environmental, social and economic aspects of the generation of electricity using nuclear energy. It has at least one member residing within 40 miles of the subject facility, upon whom it rests its claim to representational standing. Neither Applicant nor Staff contest its claim to standing and interest as prescribed by 10 C.F.R. § 2.714. We also find it has made the necessary showings as to standing and interest, which are required for participation in the proceeding.

To qualify as a party intervenor, it is not enough to satisfy standing and interest requirements. Section 2.714(b) provides that one will not be permitted to participate as a party unless it submits an admissible contention. OCRE has submitted two proposed contentions that it would litigate.

### Proposed Contention One

As its first proposed contention OCRE asserts that there is no need for the generating capacity of Beaver Valley Unit 2. It contends that the National Environmental Policy Act of 1969, and the Atomic Energy Act of 1954, as amended, require determining whether there is a need for the facility, and the absence of such need should result in its abandonment. It alleges that demand projections have changed drastically since the construction permit stage, with the result that the four utilities that own the facility have excess capacity, making the operation of Beaver Valley Unit 2 unnecessary. The four owners are Duquesne Light Company, the Cleveland Electric Illuminating Company, Ohio Edison Company, and the Toledo Edison Company. Collectively they are known as the Central Area Power Coordinating Group or CAPCO for short. Petitioner would have the Licensing Board deny the operating license application, terminate the proceeding and recommend that the Commission revoke the construction permit.

OCRE recognizes that the proposed contention is a challenge to regulation, *i.e.*, 10 C.F.R. § 51.53(c). It petitions to waive the regulation, in accordance with 10 C.F.R. § 2.758 because of alleged special circumstances in the case. OCRE contends the application of the rule would not serve the purposes for which it was adopted.

Petitioner further requests that, should the Licensing Board choose not to hear the issue, OCRE be permitted to refer the need for Beaver Valley Unit 2 to the Public Utilities Commission of Ohio for determination while this proceeding be held in abeyance pending the State agency's decision. The Public Utilities Commission of Ohio is stated to have jurisdiction of the Pennsylvania facility because of its partial ownership by Ohio utilities.

The challenged regulation, Section 51.53(c), provides:

> Presiding officers shall not admit contentions proffered by any party concerning need for power or alternative energy sources for the proposed plant in operating license hearings.

Section 2.758(c) of Title 10 of the Code of Federal Regulations permits the waiver of a regulation in an adjudicatory proceeding on the sole ground that there are special circumstances with respect to the subject matter of the particular proceeding which are such that application of the regulation would not serve the purposes for which the regulation was adopted. The waiver petition must be supported by an affidavit that identifies the specific aspect of the subject matter of the proceeding as to which application of the regulation would not serve the purposes for which it was adopted. It shall set forth with particularity the special circumstances alleged to justify the waiver requested. Any other party may file a response, by counter affidavit or otherwise.

The Commission, in formulating the final regulation 10 C.F.R. § 51.53(c), succinctly set forth its reasons at 47 Fed. Reg. 12,940 (1982). It stated:

> The purpose of these amendments is to avoid unnecessary consideration of issues that are not likely to tilt the cost-benefit balance by effectively eliminating need for power and alternative energy source issues from consideration at the operating license stage. In accordance with the Commission's NEPA responsibilities, the need for power and alternative energy sources are resolved in the construction permit proceeding. The Commission stated its tentative conclusion that while there is no diminution of the importance of these issues at the construction permit stage, the situation is such that at the time of the operating license proceeding the plant would be needed to either meet increased energy needs or replace older less economical generating capacity and that no viable alternatives to the completed nuclear plant are likely to exist which could tip the NEPA cost-benefit balance against issuance of the operating license. Past experience has shown this to be the case. In addition, this conclusion is unlikely to change even if an alternative is shown to be marginally environmentally superior in comparison to operation of a nuclear facility because of the economic advantage which operation of nuclear power plants has over available fossil generating plants. An exception to the rule would be made if, in a particular case, special circumstances are shown in accordance with 10 C.F.R. 2.758 of the Commission's regulations.

At page 12,942, the Commission commented that there had never been a finding in a Commission operating license proceeding that a viable, environmentally superior alternative to operation of the nuclear facility exists and that the Commission expects this to be true for the foreseeable future.

The agency reached its conclusion in part on the basis of findings that nuclear plants are lower in cost to operate than fossil plants and if conservation lowers demand, then utility companies take the most expensive operating plants off-line first; as a result, a completed nuclear plant is used as a substitute for the less economical generating capacity. Examples given of special circumstances that were acceptable for waiver of the regulation were a showing "that nuclear plant operations would entail unexpected and significant environmental impacts or that an environmentally and economically superior alternative existed," 46 Fed. Reg. 39,441 (1981).

In support of the waiver petition, Susan L. Hiatt, Petitioner's representative, submitted an affidavit. No information on her experience in the subject area was offered. The first point she presented is that the operation of Beaver Valley Unit 2 will result in increasing already existing excess capacity for no purpose. She supports her position by data taken from filings made by CAPCO utilities with this agency, with an agency of the State of Ohio and from annual reports. They are used to support OCRE projections and to show: at the construction permit stage for Beaver Valley Unit 2, the growth rate in electrical demand for CAPCO service was overstated; peak demand and energy supplied have been much lower than projected; that CAPCO's more recent projections for peak demand are unrealistic; that there already is and will continue to be an ample reserve margin; and that there is no need for Beaver Valley Unit 2's production. She presents data to show no additional capacity will be needed for CAPCO until the year 2026. Hiatt offers that if there is a moderate annual growth rate in demand it can be reduced to zero by conservation and also it would be far cheaper for CAPCO to purchase power from other utilities than to continue building Beaver Valley Unit 2.

The matter of conservation is treated in the affidavit in a sentence. It is to be accomplished by providing incentives such as time-of-day rates. At the conference, weatherization and unspecified kinds of load management techniques were added as proposed incentives. Support for the thesis of purchasing power comes from an article in the December 15, 1982 issue of the *Wall Street Journal.* It describes a utility in the CAPCO area that markets electricity to other utilities "more cheaply than they could produce the power themselves." As a general news article it contains no information on such things as charges for the electricity and how they would relate to production costs of Beaver Valley Unit 2.

On the issue of whether the subject facility will be needed to replace older, less economical coal capacity, her conclusion is it will not. She bases the determination on the testimony of Dr. Richard Rosen, of Energy Systems Research Group, given in 1980 before the Pennsylvania Public Utility Commission on behalf of the Pennsylvania Office of Consumer Advocate. It was to the effect it will always be cheaper for CAPCO to operate its older coal plants, which were built at lower capital costs and lower interest rates. A copy of his testimony was not furnished.

She also relies on the thesis of David Dinsmore Comey, expressed in an article attached to the affidavit, entitled *Nuclear Power Plant Reliability: The 1973-74 Record.* It holds that the operating capacity factor of plants is a crucial determinant of the comparative costs of coal and nuclear power and that a 55% capacity factor is the break-even point between coal and nuclear; if both coal and nuclear plants exceed the 55% capacity factor, nuclear will be cheaper, and if both operated at less than 55%, coal will be cheaper. Hiatt forecasts that there is little likelihood that Beaver Valley Unit 2 will operate at or above 55% capacity. She concludes this from a review of the "Gray Book," NUREG-0020, Vol. 7, No. 3, "Licensed Operating Reactors Status Summary Report," March 1983, list of seventy-two plants, of which one-third had not achieved a capacity factor of 55%.

Another reason OCRE provides as to why the 55% factor will not be reached is based on a 35.1 cumulative capacity factor given for Beaver Valley Unit 1, coupled with data that for seventeen of twenty multi-unit nuclear generating facilities, the cumulative capacity factor of each unit is about the same as others at the same site, with a variance of plus or minus 10%. With the 35.1% cumulative capacity of Unit 1, Petitioner gives Unit 2 a 3/20 or a 15% chance of attaining a cumulative capacity factor much greater than 35.1%.

The CAPCO system is reported to be totally free of the use of oil, with 95% of the electricity being generated from coal. Despite the claim the nuclear facility will not substitute for coal generating plants because of the assertion the latter are more economical to use, Hiatt cites testimony that the effect of nuclear capacity in the system reduces the efficiency of coal units and that the Mansfield-2 coal plant, within the CAPCO system, is operated as a peaking plant rather than for baseload. (*See* ¶ 25 of the Affidavit.)

Much of the affidavit is devoted to contesting the rationale that went into the promulgation of the regulation, 10 C.F.R. § 51.53(c) and its validity. Some of the material relied upon was presented prior to publishing the regulation. When the regulation is attacked it is often referred to as the "proposed rule." (*See* ¶ 22 of the Affidavit.)

Generalities were offered on the issue of the superiority of alternatives. Paragraph 21 provides, "[t]he nuclear plant has definite environmental disadvantages in comparison to fossil generation, notable catastrophic accidents. Environmental disadvantages of fossil plants (air pollution, sulfur dioxide and particulates) can be removed by installing scrubbers and precipitators."

At the special prehearing conference, Hiatt offered that allowable effluents of sulfur oxides from a uranium enrichment plant operating in support of a nuclear power plant would be about the same as the emissions from a coal plant which could replace Beaver Valley Unit 2. To buttress her position on the need for nuclear plants to operate at high capacity factors to be competitive with coal, she cited from DOE/EIA-0356-2, "Projected Costs of Electricity from Coal and Nuclear Fired Power Plants," Executive Summary, Vol. I, prepared by the Energy and Information Administration, August 1982, which stated that at plant capacity factors above 65% nuclear plants could remain competitive with coal-fired plants in most regions of the country. However, should average lifetime capacity factors fall below 65%, total costs for nuclear power would suffer more severely than those of coal-fired plants.

The Department of Energy document addresses the economic merit of nuclear versus coal-fired electricity for new plants beginning baseload service in 1995. Hiatt also noted that in the "Gray Book" only nineteen of the seventy-two plants listed achieved a cumulative capacity factor of 65%.

Applicant contends the "special circumstances" which petitioner alleges are not sufficiently particularized to warrant a waiver or exception to Part 51. Assertedly they constitute a general attack on the economics of nuclear power and on the findings of the Commission underlying the amendments. Duquesne claims the proper response is to seek an amendment or recision of the rule, not a waiver. It further contends even if, as a matter of argument, OCRE's allegations are sufficiently particularized, the thrust of the presentation is that the facility would not be needed to meet increased energy needs. Applicant asserts OCRE does not adequately address the other premise behind the amendment — that the plant would be used to replace older generating capacity.

The utility dismisses the Department of Energy Report as support for the OCRE position. Duquesne stated it relates to the matter of a new nuclear plant versus a new coal plant. It is not applicable to the subject situation of operating a plant which is completed and in which the capital costs are sunk. It proposes that the only relevant analysis is whether or not the operating costs of the nuclear plant will be greater or less than those of the coal plants.

Duquesne points out that the Rosen testimony given in 1980 before the Pennsylvania Public Utility Commission was wholly rejected in an October 1982 decision by an Administrative Law Judge. The Pennsylvania proceedings, Docket Nos. I-79070315 and I-79070317, were terminated by an order of January 7, 1983.

Duquesne is of the position that OCRE has not presented the necessary proof to make a *prima facie* case.

Staff starts with the position that the supporting affidavit is not from one who claims to be an expert in "need for power" or the economics of electric generating stations and is entitled to little weight. It contends the affidavit to a large measure has nothing specific to do with Beaver Valley and is instead a generic attack on the rule prohibiting need-for-power consideration. It considers the Hiatt argument, that nuclear plants must operate at greater than 55% of capacity to be cheaper than coal plants and that Beaver Valley Unit 2 will operate at the 35% capacity of Unit 1, to combine a generic challenge to the economics of nuclear power with a specific assertion that Beaver Valley Unit 2 will operate at a much lower capacity than will the average plant. Staff contends the thesis fails for a number of reasons, including: Beaver Valley Unit No. 1 has operated at 70% capacity during the first 7 months of 1983; four multi-unit stations had operated with a greater difference than 10% among units; using the Hiatt "Gray Book" figures, Beaver Valley Unit 2 has a 67% chance of exceeding a capacity of 55%. She is criticized for giving no factual explanation of Beaver Valley Unit 1's low capacity and why she believes Unit 2 will operate similarly. Staff contends that there is only a bare assertion the subject facility will operate at too low a capacity to justify an operating license. It is claimed that such an unsupported assertion is inadequate to warrant the waiver of a Commission regulation.

As to the matter of delaying the proceeding until a case is presented to a State authority and decided, Staff cites *Wisconsin Electric Power Co.* (Koshkonong Nuclear Plant, Units 1 and 2), CLI-74-45, 8 AEC 928, 930 (1974), for the proposition that it would not be an efficient, economical and expeditious course and should not be followed.

Based upon the record submitted, we find OCRE has not made a *prima facie* showing that should result in a certification to the Commission of the issue of whether the regulation should be waived.

As the regulation was formulated, to make the *prima facie* showing of special circumstances a petitioner would have to establish that Beaver Valley Unit 2 would not be needed: (1) to meet increased energy needs; (2) to replace older, less economical generating capacity; and (3) that there are viable alternatives to the completed nuclear plant likely to exist which could tip the NEPA cost-benefit balance against issuance of the operating license. It envisions showing that the nuclear plant operations would entail unexpected and significant environmental impacts or that environmentally and economically superior alternatives exist. The Commission had predicated the regulation on a finding that nuclear plants are environmentally superior and lower in cost to operate than fossil plants. It places a formidable burden on one seeking waiver.

OCRE passed the first hurdle by establishing through data on the CAPCO system that the subject plant will not be needed to meet increased energy needs. The system was shown to have excess capacity that would continue well into the future. Beaver Valley Unit 2 would be unneeded to meet increased energy needs. Neither Applicant nor Staff provided anything to counter this conclusion.

Petitioner established only one of the elements it needs for prevailing on the waiver issue. It failed to show Beaver Valley Unit 2 would not be used to replace older, less economical operating capacity, which is fatal to its case.

The Rosen testimony, which Hiatt cited for the proposition it will always be cheaper for CAPCO to operate its older coal plants, could not be relied upon. It was not produced for the record and it was discredited by the Pennsylvania Public Utility Commission Administrative Law Judge who heard it. It also proved contradictory in that CAPCO was stated to be using the Mansfield-2 coal plant as a peaking plant while using nuclear for baseloading generation.

At no time did OCRE produce comparative cost figures to establish it would be less economical to employ Beaver Valley Unit 2 compared to the coal plants in the CAPCO system. In fact no figures were given for the CAPCO coal plants so that any comparison could be made.

The thesis that nuclear plants must operate at upwards of 55% capacity to be cheaper than coal plants is based on industry-wide data that are not specific to CAPCO facilities. The projection that Beaver Valley Unit 2 should operate at about 35% capacity, with a 10% variation in either direction is a mere percentage calculation without probative support to establish the validity of the projection. There is no comprehensive explanation as to why such low capacity should result, which is needed to make the projection convincing. Petitioner also submitted conflicting data which would support a projection that there is a 67% chance Beaver Valley Unit 2 will operate at upwards of 55% capacity and under the Comey theory be cheaper to operate than the coal plants.

The results of the OCRE presentation can only be viewed as conflicting, speculative and not determinative of relative costs of operating nuclear and coal facilities in the CAPCO system. OCRE has not sustained its burden on the issue that Beaver Valley Unit 2 will not replace older, less economical coal generating capacity, which is fatal to its case for a waiver of the regulation.

Without belaboring the matter of the remainder of its submission, there was no showing that there are viable alternatives to the completed nuclear plant likely to exist which could tip the NEPA cost-benefit balance against issuance of the operating license. At most there were a few superficial comments on conservation and purchasing electricity from another utility. They were made in the most cursory and general terms and cannot be considered as a serious attempt at establishing those methods as viable alternatives to the completed plant and do not do so.

Many of the remaining assertions in the affidavit were matters already considered by the Commission in formulating the regulation or were general attacks on the assumptions relied upon by the Commission in promulgating the regulation and on its soundness. They are more appropriate to a request to amend or rescind a regulation pursuant to 10 C.F.R. §§ 2.758(e) and 2.802(a) rather than to a petition for waiver of a regulation under 10 C.F.R. § 2.758(b).

Other allegations made but not discussed were found to be either irrelevant or otherwise immaterial to the waiver petition and proposed contention.

The Licensing Board finds OCRE has not made a *prima facie* showing for waiver of 10 C.F.R. § 51.53(c) and that its petition to permit the admission of a contention concerning need for power or alternative energy sources in an operating license hearing is without merit. Under 10 C.F.R. § 2.758(c) we are barred from further consideration of the matter. In conformity with that section the Licensing Board finds that OCRE's Proposed Contention One is not litigable and is denied.

As to the request of OCRE that it be permitted to refer the need-for-power issue to the Public Utilities Commission of Ohio and that this proceeding be stayed pending a decision by the State body, it is denied. The procedure sought would cause unnecessary delay when we are charged with the timely deciding of applications. The outcome of the State proceeding may have no relevance to this operating license application. The suit has yet to be filed and there is no time frame for its disposition. To approve Petitioner's request would be to place an unjustified impediment in the working of this Commission's administrative process.

**Proposed Contention Two**

OCRE contends that the operation of Beaver Valley Unit 2 will be an impermissible activity violating the requirements of Exec. Order No. 11,988, Floodplain Management, 42 Fed. Reg. 26,951 (1977).

It alleges the Nuclear Regulatory Commission (NRC) has responsibilities under that Order and its implementing regulations, Guidelines for Implementing Exec. Order 11,988, prepared by the Water Resources Council, 43 Fed. Reg. 6030 (1978), because they apply to all agencies that "conduct activities and programs affecting land use, including planning, regulating and licensing."

It charges that the Commission, contrary to the Order and Guidelines, failed to promulgate implementing regulations concerning floodplain siting or management, as required.

OCRE asserts as requirements of the Order and Guidelines that a hazardous facility, such as a nuclear power plant, not be permitted to be sited and operated within the 100- and 500-year floodplains (43 Fed. Reg. 6043), and alleges it is something NRC will allow. OCRE states that the Guidelines require early (as early as it is known that an action affects the floodplain) public notice and review of the proposed action (43 Fed. Reg. 6044), which requirement the Commission ignored. It noted that it was first in the NRC Notice of Opportunity of Hearing, of June 1, 1983, that the Commission stated that Beaver Valley Unit 2 will have structures and construction activities located on the floodplain and that the subject of floodplain management will be discussed in the Commission's environmental statement. OCRE contends that the NRC has failed to meet its responsibilities as required by not providing for an early public review of the action.

Petitioner charges that the Commission appears willing to support the licensing of a hazardous facility in a floodplain, contrary to the Executive Order,, by explaining away deleterious effects in a forthcoming environmental impact statement. It concludes that the operation of a nuclear power plant in a floodplain, with adverse environmental effects, is contrary to the Order and therefore the application for an operating license must be denied. OCRE would not consider alternative sites because of its determination in Proposed Contention One that there is no need for the facility.

Applicant raises the questions of whether OCRE as a private party has the authority to enforce the Executive Order and whether it is applicable to the NRC. As to the charge NRC has failed to adopt implementing regulations, Duquesne contends it is an attack on Commission regulation and not a permissible contention.

Applicant is of the position that NRC is not in violation of Exec. Order 11,988. The Order provides that the agency's evaluation "will be included in any statement prepared under Section 102(2)(c) of the National Environmental Policy Act" and this is what the Commission is committed to do. Duquesne contends OCRE does not recognize that the nuclear reactor and virtually all of its supporting facilities, except for such items as intake and discharge structures shared with Unit 1, are located outside of the floodplain and are not subject to the Executive Order. (Tr. 143). It charges no specific allegations are made that floodplain activities in connection with Unit 2 pose any risk of flood hazard or are in any way inconsistent with floodplain management.

Duquesne further charges that OCRE provides no basis for contending there are no practicable alternatives to the floodplain siting. It finds without merit Petitioner's contention that operation of a nuclear power plant in a floodplain is contrary to the Executive Order. Applicant sees nothing in Exec. Order 11,988 that would bar all federal actions in the floodplain. It cites *Cape May Green, Inc. v. Warren*, 698 F.2d 179, at 191-93 for the proposition that actions are expressly permitted if they are reasonable.

Staff asserts the contention is fatally vague and lacks both legal and factual bases. It claims no showing was made that operation of the plant will threaten the floodplain or that the effect of flooding was not fully considered in the FSAR and the construction permit FES.

We find that Exec. Order 11,988 and the underlying Guidelines for Implementing Exec. Order 11,988 are not binding on the NRC. The Order is directed to an agency as defined in its Section 6. That section defines "agency" as the term "Executive agency" as used in Section 105 of Title 5 of the United States Code. That section of the Code states:

> For the purpose of this title, "Executive agency" means an executive department, a Government corporation, and an independent establishment.

One must look to Section 104 of Title 5 of the United States Code to determine what is an independent establishment. It states:

> For the purpose of this title, "independent establishment" means —
>
> (1) an establishment in the executive branch (other than the United States Postal Service or the Postal Rate Commission) which is not an Executive department, military department, Government corporation, or part thereof, or part of an independent establishment; and
>
> (2) the General Accounting Office.

The NRC does not fall within the category of any of the described bodies to which the term "agency" applies and the Order is directed. It was established as an independent regulatory agency, not within the Executive Branch, under the provisions of the Energy Reorganization Act of 1974 (88 Stat. 1242; 42 U.S.C. § 5801) and Exec. Order 11,834 of January 15, 1975, effective January 19, 1975. As such it is not subject to Exec. Order 11,988.

The NRC is thus not required to establish regulations implementing the Order and Guidelines. Had it been required to do so, that matter still would not have presented the basis for a litigable contention. The issue of whether the Commission should promulgate regulations of general applicability would be the proper subject of a petition to issue regulations under the provisions of 10 C.F.R. §§ 2.759 and 2.802.

The agency has taken upon itself to be guided by the Order and Guidelines. It will review the subject of floodplain management in the Commission's environmental statement as called for by the Executive Order. In acting in accordance with the Order, the NRC cannot be faulted for not proceeding in a timely manner. The matter at issue is whether the facility should be licensed to operate. Notification to the public that the Beaver Valley Unit 2 facility will have structures and construction activities located in the floodplain, contained in the Notice of Opportunity for Hearing of June 1, 1983, was furnished in accordance with procedures and time frames followed in processing operating license applications. There will be time to contest the evaluation in the environmental statement, prior to deciding the application.

Not at issue here is the Commission's granting of a construction permit for the site. It occurred on May 3, 1974, more than 3 years prior to promulgation of the Executive Order (May 24, 1977) and issuance of the Guidelines (February 10, 1978). As with the National Environmental Policy Act of 1969, to which it relates, there is nothing in the Order that would give it *ex post facto* applicability, if it were possible.

By its proposed contention OCRE would bar the NRC from following the procedures and making the evaluation called for by the Executive Order and Guidelines to determine whether the requested licensing would be compatible with floodplain protection. Without adequate grounds Petitioner has already concluded that the Executive Order and Guidelines prohibit from use the type of facility Applicant proposes to operate on the site. OCRE overlooks that the Executive Order and implementing Guidelines allow for practicable alternatives and actions where possible. It also ignores that only some of the facility's structures, shared with Unit 1, are within a floodplain and the effect this could have on whether the unit can rightfully operate there.

Petitioner, after disregarding the foregoing, concludes, "[t]he Commission appears willing to support the licensing of a hazardous facility in a floodplain, contrary to the Executive Order, by explaining away deleterious effects in a forthcoming environmental impact statement." OCRE presents nothing to litigate in an adjudicatory proceeding with its prejudgment of future Staff action. The hearing process is to determine facts that are in controversy and to apply the law. That is not what OCRE is seeking to have done. The Licensing Board cannot decide the validity of actions that are yet to happen.

Petitioner's Proposed Contention Two presents nothing to litigate at this time and must be denied. The discussion of floodplain management in the Commission's environmental statement may give rise to matters that should be decided by the adjudicatory process. Until the statement is issued and its contents known, any treatment of it is speculative, premature and does not provide a basis for an admissible contention. For a discussion on how a proposed contention should be submitted when the unavailability of relevant documents makes it impossible for a petitioner to assert adequately specific contentions at an earlier date, *see Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983).

Because OCRE has not submitted a litigable contention it cannot be permitted to participate as a party in the proceeding. Under the provisions of 10 C.F.R. § 2.714(b) its petition to intervene is denied.

## DETERMINATION OF THE ECNP PETITION

ECNP is a nonprofit citizens' organization composed of groups and individuals who are interested in nuclear power and the nuclear fuel cycle. It is concerned about the providing of safe, clean, reliable and affordable electric and other energy supplies. Petitioner has participated in a number of proceedings before the NRC.

In its petition of June 30, 1983, to intervene in the subject proceeding and to hold a hearing, it named members of the organization including Dr. Robert Freedman, who were stated to reside within 50 miles of Beaver Valley Unit 2. No addresses were furnished. It was further stated that organization members live, work and raise their families within the given area and are at risk of injury, latent disease, having genetic defects in their children and loss of value, possession or access to their property from an accident at Beaver Valley Unit 2. Dr. Judith H. Johnsrud, its co-director, was reported to have been appointed by the "Board of Directors of ECNP, including members living within the designated distance from Beaver Valley Unit 2" as legal representative for ECNP and its members to protect their interests in any and all proceedings before the Commission. The pleading was signed by Dr. Johnsrud, who is not an attorney-at-law.

In response, Applicant cited *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station), ALAB-535, 9 NRC 377, 393 (1979) in support of its allegation that ECNP failed to demonstrate its standing to intervene as a matter of right.

Staff found the petition deficient because ECNP, seeking representational standing, did not submit an affidavit from a member showing the affiant's interests that may be affected by the facility and that the affiant had authorized the organization to represent it in the proceeding. It relied on *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-536, 9 NRC 402, 404 (1979) and *Allens Creek, supra,* in support of its allegation.

By memorandum and order of August 4, 1983 (unpublished), we ordered the holding of a special prehearing conference on October 12, 1983, pursuant to 10 C.F.R. § 2.714a to pass upon unresolved issues on standing and on proposed contentions to be submitted. Supplemental or amended petitions were to be filed on September 9, 1983. Pursuant to an unopposed request of ECNP to set up a document room in State College, Pennsylvania, we requested that Staff set one up, with the proviso that Petitioner should use existing document rooms pending the availability of documents at State College so that the ordered schedule would not be disrupted.

In a supplemental memorandum served September 9, 1983, ECNP advised that no library had been set up at State College and Petitioner had been severely hampered in its ability to comply with the schedule for the filing of proposed contentions. It stated that ECNP members residing within the 50-mile radius of Beaver Valley Unit 2 had been notified to submit directly to the Licensing Board Chairman affidavits regarding their interest and authorization of the organization to act on their behalf. There was contained in the memorandum a listing of eleven proposed contentions of the organization.

An affidavit dated September 9, 1983 was received from Dr. Robert W. Freedman, 5028 Debra Drive, Pittsburgh, Pennsylvania 15236. It stated he lives less than 50 miles from Beaver Valley Unit 2; the operation of the plant would create an unacceptable health and safety risk to himself and his family; and as a member of ECNP he designates Dr. Judith Johnsrud to be his legal representative for opposition to the granting of an operating license to the Applicant.

The affidavit was defective in that the notary recited that it was the Chairman of the Licensing Board who appeared before her as the affiant.

Applicant in its response found the Freedman affidavit inadequate to establish representational standing by ECNP because it failed to establish the standing of Dr. Freedman in that it was not shown how the operation of Beaver Valley Unit 2 would cause him any injury. His assertion of an "unacceptable health and safety risk" was deemed insufficient to qualify. It was further asserted Dr. Freedman failed to authorize Petitioner to represent him by designating Dr. Johnsrud as his representative. It was also noted that the notarization was invalid.

Applicant contended all of ECNP's contentions were defective in one or more respects, lacking the basis and specificity required by 10 C.F.R. § 2.714(b). It requested that the petition to intervene should be denied for failing to submit at least one adequate contention.

Staff's position was it did not object to the validity of Dr. Freedman's statement of interest in the proceeding. It did question his designation of Dr. Johnsrud as his legal representative. It was claimed if Dr. Johnsrud was to represent the organization, it was up to that body to select her, which showing had not been made. The deficiency in the notarization was also noted.

Staff denied it had failed to comply with the requirement to establish a local public document room. It is its custom to place only transcripts at such facilities and as yet none had been prepared in the proceeding. Staff further alleged ECNP's contentions were vague and completely lacking in specificity and basis. It requested that the petition for leave to intervene be denied because Petitioner failed to provide an admissible contention.

At the special prehearing conference, Dr. Johnsrud explained that she has been legal representative of ECNP for the past 6 years during which the organization has appeared before the NRC. Her status had been reaffirmed at a meeting of the Board of Directors within a week of the conference. She indicated written confirmation would be provided.

She further advised that Dr. Freedman was a member of the Board of Directors residing in Pittsburgh, within 50 miles of the subject site. She considered the erroneous notarization an awkwardness of wording. Staff took the position that the inadequacy of the affidavit was the failure of Dr. Freedman to authorize ECNP to represent his interest. Dr. Johnsrud agreed to have the matter of the notarization corrected and to have the language in the affidavit changed to meet Staff's objection. She agreed to have Dr. Freedman submit another affidavit within 15 days of the close of the special prehearing conference which occurred on October 12, 1983. No such affidavit was received by the Licensing Board.

On the issue of a local public document room and its contents, Dr. Johnsrud's experience had been that in addition to transcripts there were contained the major documents of the application. She had not been able to make use of the public document room established at Alliquipa, Pennsylvania because of other obligations and finances. Staff maintained that its responsibilities only extended to providing transcripts at local public document rooms and that any other documents had come from Applicants. Duquesne said it had not made such application documents available at State College because it had not been requested to do so. It agreed to make the final safety analysis report and environmental report available to Petitioner and did so at the special prehearing conference.

ECNP requested additional time to make its proposed contentions more specific with the availability of the basic documents. After a review of all of Petitioner's proposed contentions, ECNP requested additional time to perfect 6, 7, 8, 9, 10 and 11. The Licensing Board determined that of all of the proposed contentions, only 6, 7, 8, 10 and 11 could relate to the information contained in Applicant's final safety analysis report and environmental report. To assure that Petitioner would not in any way be prejudiced by not having the documents available when it prepared its proposed contentions, ECNP was given the 15 days it requested to add specificity to the last-enumerated five proposed contentions. Petitioner never responded to the October 26, 1983 due date.

As the result of Petitioner's failure to have a revised affidavit submitted by Dr. Freedman and to file additional information as to proposed contentions 6, 7, 8, 10 and 11 within the time set, we issued an order for ECNP to show cause as to why the Licensing Board should not rule on the issues of interest and standing and admissibility of the proposed contentions on the basis of the information Petitioner submitted up to and through the special prehearing conference. The order was served on November 30, 1983 and ECNP was given 14 days to respond. No response was received from ECNP. Based upon ECNP's failure to make the above filings, good cause exists for deciding the issues of Petitioner's interest and standing and the admissibility of its Proposed Contentions 6, 7, 8, 10 and 11 on the basis of the record made through the special prehearing conference on October 12, 1983. The record is therefore closed on the issues as of that date.

Under the existing law on intervention, close geographical proximity of a petitioner's residence to the facility, standing alone, is sufficient to satisfy the interest requirements of 10 C.F.R. § 2.714. Residence within 50 miles of the facility generally has been found to be acceptable. *Tennessee Valley Authority* (Watts Bar Nuclear Plant, Units 1 and 2), ALAB-413, 5 NRC 1418, 1421 n.4 (1977); *Texas Utilities Generating Co.* (Comanche Peak Steam Electric Station, Units 1 and 2), LBP-79-18, 9 NRC 728, 730 (1979); *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), LBP-79-10, 9 NRC 439, 443-44 (1979); *Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), LBP-79-1, 9 NRC 73, 78 (1979).

An organization can gain standing as the representative of a member or members of the organization who have interests which may be affected by the outcome of the proceeding. *Public Service Co. of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-322, 3 NRC 328, 330 (1976). Where an organization's entitlement to intervene is wholly dependent on the personal standing of its members, at least one of those members must be identified with sufficient specificity so that the matters stated can be independently verified. There must be a demonstration that the member has authorized the organization to represent the individual's interest in the proceeding. *Allens Creek, supra.*

Dr. Freedman, by his statement, has satisfied the requirements for establishing his interest in the proceeding. He has specifically identified himself. His residence in Pittsburgh, Pennsylvania, within 50 miles of the facility is sufficient to satisfy the interest requirements of 10 C.F.R. § 2.714, as provided by the cited cases. Considering Beaver Valley Unit

2 is within approximately 35[1] miles of any point in Pittsburgh, his residence is probably much closer than the indicated distance.

In order for ECNP to obtain representational standing on the basis of the interest of its member, Dr. Freedman, it must be established he has authorized the organization to represent his interest in the proceeding. This has not been done.

The garbled and confusing affidavit of Dr. Freedman does not designate ECNP to represent his interest in the proceeding. It may be argued that when the Freedman statement is read in conjunction with ECNP's petition to intervene of June 30, 1983, it can be inferred he intended to designate the organization as his representative and that it satisfies the requirement. We are unwilling to draw that inference from a document that has been drawn and submitted with such lack of care. The affidavit is unclear as to its import, stating that as a member of the organization Dr. Freedman wants Dr. Johnsrud to be his legal representative. It is devoid of any statement that he wants the organization to represent him. It is unwarranted to infer the foregoing in light of the deficiencies in the document and where the opportunity was provided to revise the document and was ignored.

ECNP has not established its standing to participate as a party intervenor in the proceeding under 10 C.F.R. § 2.714. Its petition to intervene is therefore denied.

ECNP has not sought discretionary intervention in the proceeding, so there is no requirement to treat with the matter. Its failure to make requested filings on critical issues rules against our giving it such consideration. ECNP's performance in this proceeding has been such that it cannot be expected to assist in developing a sound record. This grave fundamental failure of Petitioner outweighs all other factors to be considered in deciding the question of discretionary intervention, if the matter were at issue. In that Petitioner has already submitted its proposed contentions, we will review them to determine if ECNP has presented anything of merit to consider. In accordance with our prior determination, we will analyze the proposed contentions on the record made through October 12, 1983.

[1] The Notice of Opportunity for Hearing places the facility approximately 22 miles northwest of Pittsburgh. Official notice of the size of Pittsburgh, obtained through the 1982 edition of the *Rand McNally Road Atlas*, shows the city to be no more than 13 miles at its most distant point from its northwestern boundary.

### Proposed Contention One

> It is contended that interrelated financial associations and ownerships among the Applicant, the reactor vendor and other suppliers, specifically the Westinghouse Corporation, and financial institutions, specifically the Mellon Bank headquartered in Pittsburgh, are sufficiently great to constitute conflicts of interest with respect to the safe construction and quality of equipment utilized in the construction of Beaver Valley 2 and with respect to adequate assurance of safe operation of the plant.

Petitioner does not believe there is a legal prohibition against such financial interconnections but is concerned that the relationship could impede the utility from adequately performing its function. (Tr. 67-68). ECNP would rely on discovery to add specificity to the proposed contention. (Tr. 65).

Applicant would deny the contention for lack of specificity and basis. It contends that the proposed contention is inconsistent with the NRC's statutory charter and regulations; that any interlocking parties would have a vital interest in assuring the safe and reliable operation of Beaver Valley Unit 2; and that it is nothing more than a generalization of what ECNP considers applicable policies ought to be.

Staff finds the proposed contention to be without basis in that there is no explanation as to why an interrelated financial association should affect safe construction or operation of the plant.

We find Proposed Contention One to be without a legal or factual basis that would permit it to be admitted as a litigable contention.

Assuming *arguendo* the interrelationships as stated, there are no statutes or regulations pertaining to safety or the environment that prohibit such affiliation.

Petitioner has provided nothing from which it could reasonably be concluded the interrelationships would result in possible safety problems in construction, quality of equipment or operating the unit.

Under Commission practice the proposed contention must be denied. Commission regulations do not allow the filing of vague, unparticularized contentions, to be followed by an attempt to flesh them out through discovery of Applicant or Staff. *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), ALAB-687, 16 NRC 460, 468 (1982).

### Proposed Contention Two

> It is contended that the economics of safe disposal of radioactive wastes that will be generated by the operation of Beaver Valley 2 remain uncertain and that, in the continuing absence of either approved sites or demonstrated effectiveness of such waste disposal, the Beaver Valley Unit 2 nuclear reactor must not be permitted to receive an operating license. This cost uncertainty has not been fully and properly eval-

uated in the cost-benefit comparison with alternatives to the reactor as is required by the National Environmental Policy Act of 1969, and as is indicated in the California decision by the Supreme Court.

Proposed Contention Two presents nothing to litigate before this Licensing Board. Contrary to Petitioner's assertion, the economics of the safe disposal of radioactive wastes that could be generated by operation of Beaver Valley Unit 2 do not remain uncertain. The Nuclear Waste Policy Act of 1982, 42 U.S.C. § 10,101 *et seq.* established a funding mechanism to cover the cost for the disposal of spent fuel and/or high-level radioactive waste and established the fee to be paid by those owning and generating this material at 1 mill per kilowatt hour. The Department of Energy has promulgated implementing regulations. 48 Fed. Reg. 16,590, *et seq.* (1983).

The premise of the proposed contention that the costs of disposal are uncertain is incorrect. There is no ground to support its conclusion that because of the alleged uncertainties in costs there has resulted an inadequate evaluation in the cost-benefit balance determination required by the National Environmental Policy Act of 1969. Even if there were an uncertainty that might alter the cost-benefit balance as it relates to alternative energy sources, it is not a matter for consideration by a licensing board at an operating license hearing. The issue is expressly prohibited by 10 C.F.R. § 51.53(c).

As to whether effective permanent high-level radioactive waste storage facilities will be available and spent fuel can be safety managed until such sites are available, the Commission has answered this question in the affirmative. *Rulemaking on Storage and Disposal of Nuclear Waste* (Waste Confidence Rulemaking), Nos. PR-50, PR-51, unpublished Commission Decision at 5-6 (May 16, 1983). *See also* 48 Fed. Reg. 22,730 (1983).

To the extent the proposed contention may imply that the environmental impacts of waste disposal are uncertain, that issue has been excluded from litigation by Table S-3 of 10 C.F.R. Part 51. *See Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), LBP-83-6, 17 NRC 153 (1983).

"The California decision by the Supreme Court" referred to by ECNP is understood to mean the recent case of *Pacific Gas and Electric Co. v. State Energy Resources, Conservation and Development Commission,* 103 S. Ct. 1713 (1983). It adds no support to the ECNP contention in that it deals with an unrelated matter. It upheld the right of California to condition the construction of new nuclear reactors upon the making of a finding by a State regulatory body that a demonstrated means for permanent waste storage had been found. The case has no applicability to plants under construction or to those outside of California.

The matters ECNP proposes to litigate under Proposed Contention Two have already been decided. The determinations are binding on this Licensing Board leaving it nothing to adjudicate. The proposed contention must be denied.

### Proposed Contention Three

> In view of the uncertainties remaining about the costs of the disposal of radioactive wastes generated by the Beaver Valley 2 reactor, the Applicant's projection of sales of electricity to be generated by this plant are not adequate or accurate enough to sustain the issuance of an operating license for Beaver Valley 2.

Proponent believes Proposed Contention Three follows from the prior one and nothing by way of further elaboration was offered other than that ECNP is a party to the Court action challenging the regulation prohibiting the consideration of the financial qualification of the utility at the operating license stage. (Tr. 75).

Proposed Contention Three asserts the cost of the disposal of radioactive waste is uncertain, and as is understood by the Licensing Board, ECNP claims it would impact on the cost of operation of the facility so as to raise questions as to its profitability and the financial capability of the owners to operate Beaver Valley Unit 2.

As found in Proposed Contention Two, the costs of disposal of radioactive wastes are not uncertain, making the premise of the contention incorrect. There is no basis provided to support its conclusion that the costs of disposal are such as to raise questions to its profitability and the financial qualifications of Applicant to operate the plant.

Even if it were the case, the effect the disposal costs would have on Applicant's financial capability to operate Beaver Valley Unit 2 is not a proper matter for consideration in an operating license hearing. Just as with need for power and alternative energy sources, licensing boards are prohibited from hearing whether the Applicant is financially qualified to engage in the activities to be authorized by the operating license. *See* 47 Fed. Reg. 13,750 *et seq.* (1982) and 10 C.F.R. § 2.104(c)(4).

Proposed Contention Three presents nothing to litigate and is denied.

### Proposed Contention Four

> The failure of the Commonwealth of Pennsylvania to assure the provision of safe isolation of low level radioactive wastes which will be generated by the Beaver

Valley 2 reactor, the failure of the Applicant to provide assured isolation of such wastes, and the mandate of the Congressional Low Level Radioactive Waste Policy Act of 1980 that each state must be responsible for the management of those wastes after January 1, 1986, taken in combination, give insufficient assurance that the low level wastes which will be generated by the operation of Beaver Valley 2 will be properly and safely isolated from the environment; hence the license must be denied.

The gravamen of the proposed contention is that there is insufficient assurance that the low-level wastes produced by the operation of Beaver Valley Unit 2 will be properly and safely isolated from the environment.

ECNP is concerned Pennsylvania has not committed itself as to how it will deal with the matter of the disposal of low-level waste.

The Low Level Waste Policy Act of 1980, 42 U.S.C. § 2021(b) *et seq.* placed upon each State the responsibility for providing available facilities within or outside the State, for the disposal of low-level radioactive waste generated within its borders. It provides that States may enter into compacts as may be necessary to provide for the establishment and operation of regional disposal facilities. Such compact does not take effect until Congress consents to it. After January 1, 1986, any such compact may restrict the use of the regional disposal facilities to nonsignators.

ECNP asserts that some States within the Northeast have approved such a compact but that Pennsylvania at present is only considering legislation (a) to join the compact and (b) to proceed with a demonstration radioactive waste site within the Commonwealth. (Tr. 76, 81).

Petitioner in addition to being concerned about there being facilities available for the waste, either arranged for or provided by Pennsylvania, is troubled by the possibility that no site would be obtainable by Applicant, on its own, outside of Pennsylvania. (Tr. 77).

Applicant contends the purpose of the Low Level Waste Policy Act of 1980 is to increase the availability of facilities by encouraging the designation of new sites. It states Duquesne presently has the option of shipping the nuclear wastes or it can store the matter on site. (Tr. 79). The onsite facility is currently being constructed. (Tr. 80). Duquesne considers that the proposed contention is highly speculative and does not lend itself to adjudication in a licensing proceeding.

Staff opposes the proposed contention for being vague and without legal or factual basis. It also considers the proposed contention to be speculative.

Proposed Contention Four presents no litigable issue for determination. Applicant is proceeding with construction that will enable the low-level radioactive waste to be stored on site. That alone renders the proposed contention moot because Applicant will not have to rely upon any of the alternative methods ECNP considers as possibly inadequate.

Furthermore, no sound basis was presented from which it can be concluded other means of waste disposal will not continue to be available.

The January 1, 1986 date alone is not a bar to the use of any additional sites that would become available through a regional compact even if Pennsylvania were not a party. The compact would have to prohibit use by Pennsylvania operators and it would need the approval of Congress.

Pennsylvania is considering becoming a compact State which could make a regional site available. It is also considering a location in Pennsylvania, which could provide another additional site.

There is nothing to indicate any currently available site would bar Duquesne from using it.

In summary, proposed Contention Four is not litigable. Petitioner is concerned there will not be a low-level waste disposal facility available to Applicant, whereas Duquesne will have one on site. Nothing was submitted to show Applicant would be barred from using currently available facilities. Pennsylvania is also considering two additional methods for making waste disposal sites available.

Considering all of the foregoing, Petitioner's concern that the low-level wastes to be generated by the operation of Beaver Valley Unit 2 will not be properly disposed of is without basis. A proposed contention without basis cannot be litigated and must be denied. Proposed Contention Four is therefore denied.

**Proposed Contention Five**

> It is contended that the health effects caused by the emission of radon gas into the environment as a result of the mining and milling and mill tailings piles created in support of the annual operations of Beaver Valley 2 remain uncertain in the absence of resolution of this issue (10 C.F.R. 51.20(e) Table S-3), and that the long-term impact of this radon gas will be unacceptably detrimental to the health of future human beings. Absent resolution of this issue by the Courts, a license to operate Beaver Valley 2 should not be granted.

Applicant asserts the proposed contention presents a generic issue that is pending before the Commission and is not litigable. It relies on the decision of the Commission in *Philadelphia Electric Co.* (Peach Bottom Atomic Power Station, Units 2 and 3), CLI-83-14, 17 NRC 745 (1983). In that decision the Commission considered the Appeal Board's opinion (ALAB-701, 16 NRC 1517 (1982)), which is a consolidated proceeding on the health effects of radon, in which the Appeal Board at

1528 found the "*incremental* health risk to the population stemming from the fuel cycle emissions (if indeed there is any) is vanishingly small."

In its consideration of the Appeal Board opinion, the Commission determined to hold "in abeyance its decision whether or not to review ALAB-701 pending a determination whether to initiate a further rulemaking to amend the mill tailings regulations and, if such a rulemaking is initiated, pending its conclusion." In so doing, the Commission instructed at 751:

> This action would stay the decision in ALAB-701 and, accordingly, licensing boards should continue to defer consideration of radon issues and appropriately condition licenses pending a final decision of the status of ALAB-701 after a determination regarding rulemaking as described above.

Applicant asserts the licensing board is bound by this directive. It further contends it would be an inappropriate issue to litigate on the grounds of collateral estoppel in that ECNP has been and is a full party to the consolidated radon proceeding.

A third reason it gives as to why the proposed contention should not be litigated is the holding of the Appeal Board in *Potomac Electric Power Co.* (Douglas Point Nuclear Generating Station, Units 1 and 2), ALAB-218, 8 AEC 79 (1974) that licensing boards should not accept in individual licensing cases any contentions which are or are about to become the subject of general rulemaking. (Tr. 88).

Staff would have the licensing board adhere to the Commission determination in *Peach Bottom.* It would bar ECNP from relitigation of what Petitioner has already litigated in the consolidated radon proceeding.

In rebuttal, ECNP is of the position that the health effects of radon continue to be a suitable subject for litigation in this case. Petitioner believes the Commission has only called for a deferral of the subject but has not forbidden it. (Tr. 85).

We find Proposed Contention Five is not an admissible contention. The Commission direction in its *Peach Bottom* decision, to defer consideration of the radon issue, effectively proscribes its consideration by the Licensing Board at this time. Conceivably the wording of the Commission's instruction to defer the matter leaves open the possibility that the Commission may authorize its handling in individual cases at some future time. More realistically the Commission has said not to consider the matter on an individual basis at this time because it will become the subject for a generic disposition. It is a reiteration of the holding in the *Douglas Point* case, that licensing boards should not accept in individual licensing cases contentions which are about to become the subject of general rulemaking. Albeit, the directive of the Commission is controlling of the Licensing Board, which finds Proposed Contention Five to be unlitigable at this time and not admissible.

In view of the foregoing determination, whether the doctrine of collateral estoppel should be applied is academic. It would be premature to seek to invoke the doctrine in that the consolidated radon proceeding has not been finally decided. Its review is being held in abeyance by the Commission. Until the prior proceeding is finalized, it cannot be said the matter was previously litigated and decided.

**Proposed Contention Six**

> It is contended that the issue of systems interaction has not been resolved by the Nuclear Regulatory Commission and that the uncertainties of safe operation of Beaver Valley 2 are therefore too great to permit issuance of an operating license for this reactor until the Commission has actually solved this issue.

Applicant asserts that this proposed contention presents another generic issue that is pending before the Commission and is not litigable. NUREG-0606, "Unresolved Safety Issues Summary," Vol. 5, No. 2, Task A-17 (May 1983). Duquesne claims ECNP's contention basically asserts that the NRC cannot issue an operating license until the generic issue of systems interaction has been resolved.

Staff opposes the proposed contention on the basis that it is a generic attack offering ECNP's view that until the systems interaction unresolved safety issue is resolved no licenses can be issued.

The criteria for accepting a contention based on a generic issue, as this is, is set out in *Gulf States Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760, 773 (1977) and *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245, 248 (1978). The party presenting such a contention must show:

> (1) That the undertaken or contemplated project has safety significance insofar as the reactor under review is concerned, and (2) that the fashion in which the application deals with the matter in question is unsatisfactory, that because of the failure to consider a particular item there has been insufficient assessment of a specific type of risk for the reactor, *or* that the short-term solution offered in application to a problem under Staff study is inadequate.

*River Bend, supra,* 6 NRC at 773.

The only supporting information offered by ECNP was a reference to a lack of progress in resolving the systems interaction issue by the NRC

Staff. (Tr. 91). This reference is to a differing professional opinion on the part of an NRC Staff member which had nothing to do with Beaver Valley Unit 2 or Unit 1. The matter of the kind of progress the Commission is making on the generic issue of systems interaction is not to be litigated by a licensing board. It is not within our jurisdiction and cannot be heard.

ECNP makes no effort to show how the proposed contention is either applicable to Beaver Valley Unit 2 or what systems at Beaver Valley Unit 2 are involved in the systems interaction issue. The proposed contention totally lacks specifics. There is no basis for our admitting the proposed contention under the criteria set forth in *River Bend.* The lack of specificity renders meaningless its claim "that the uncertainties of safe operation of Beaver Valley Unit 2 are therefore too great to permit issuance of an operating license."

ECNP's asserted concern with the NRC's lack of progress in the systems interaction area does not present a litigable issue. It has presented nothing more in regard to the subject facility. Proposed Contention Six is therefore denied.

**Proposed Contention Seven**

> It is contended that operability of auxiliary equipment necessary to the safe operation and shutdown of Beaver Valley 2 is dependent in part upon non-safety grade equipment whose performance cannot be relied upon to provide adequate protection of the public.

Applicant opposed the contention as unduly vague and totally lacking in any specificity or basis. It points out ECNP does not identify any of the equipment involved or indicate why it cannot be relied upon to provide adequate protection of the public. Duquesne asserts the proposed contention is so vague and lacking in requisite specificity and basis that it is not on notice as to the issue to be litigated.

Staff in its opposition concludes the proposed contention is totally lacking in specificity and is without basis needed for admission.

The only clue ECNP provided as to the basis of its concern was a statement made at the special prehearing conference, that over the past year a number of NRC Staff Board Notifications have indicated that problems exist relating to equipment that was not safety-grade interacting with auxiliary equipment. (Tr. 98).

ECNP never attempted to identify the specific problem it was concerned with or related it in any way with Beaver Valley Unit 2. The proposed contention is extremely vague and unrelated to the subject facility to the point one is unaware of the issue to be litigated. The proposed contention cannot be litigated and must be denied. Proposed Contention Seven is therefore denied.

### Proposed Contention Eight

> It is contended that the probability and consequences of the occurrence of pressurized thermal shock in Beaver Valley 2 have been inadequately and incompletely addressed by the Nuclear Regulatory Commission. In the absence of demonstrated proof testing in sufficient quantity to establish a margin of certainty, the risk of major accident to Beaver Valley 2 from this cause remains too great to permit the issuance of an operating license.

Applicant opposes this proposed contention because it lacks basis, is contrary to Commission regulations, and raises a generic issue. Staff also bases its objection to this contention on the grounds it raises a generic issue and lacks specificity to Beaver Valley Unit 2.

Pressurized thermal shock refers to the phenomenon of vessel embrittlement caused by radiation damage. The question of vessel embrittlement has arisen with respect to a small group of older PWRs with reactor vessels containing a relatively high copper content. In contrast, the reactor pressure vessel for Beaver Valley Unit 2 has low copper content. *See* FSAR at 5.3-1. (Applicant's Response, 9/26/83, at 20). ECNP does not provide any connection between Unit 2 and the reactors where embrittlement has been a concern.

NRC regulations deal extensively with reactor pressure vessel integrity. Part 50, Appendix G specifies fracture toughness requirements for pressure-containing-components which are fabricated from steel. Part 50, Appendix H specifies the surveillance program which all power reactor licensees must maintain to monitor irradiation-induced fracture toughness changes. The FSAR shows in detail how Beaver Valley Unit 2 complies with applicable fracture toughness and surveillance requirements.

ECNP appears to call for "proof testing in order to establish a margin of certainty" before Unit 2 may be licensed. Such testing is not required as a condition of licensing and may be in conflict with Appendices G and H of Part 50.

The NRC has designated pressurized thermal shock as Unresolved Safety Issue A-49. NUREG-0606, "Unresolved Safety Issues Summary," Vol. 5, No. 2 (May 1983) at 40. Again, the ECNP contention fails to meet the tests specified in ALAB-444 and ALAB-491, *supra,* for accepting a contention based on a generic issue. Specifically, ECNP has failed to show (1) that the issue has safety significance insofar as Unit 2 is concerned and (2) that the application fails to deal with the matter satisfactorily, pending completion of the NRC Staff study.

Because of its vagueness, lack of specificity and its attempt to raise a generic issue, without meeting Commission requirements, this proposed contention cannot be litigated and must be denied. Proposed Contention Eight is therefore denied.

### Proposed Contention Nine

> *It is contended that the operational record of Beaver Valley 1 constitutes a basis for uncertainty as to the management capability of the Applicant to operate safely two reactors at this site. In the absence of improved performance of management, an operating license for Beaver Valley 2 should be withheld.*

The gravamen of the contention is that the low operating capacity factor of Beaver Valley Unit 1 must be indicative of a failing on the part of management, which might extend over into the unsafe operation of two units at the site. (Tr. 106).

Applicant opposes the proposed contention on the grounds ECNP fails to identify any instance of management incapability and thus fails to support a challenge to the facility's management.

Duquesne finds no connection between low operating capacity factors and safe operation. It argues that low capacity may indicate the exercise of the highest degree of management capability from a safety standpoint, reflecting conservatism in avoiding safety problems. The utility placed the 1982 capacity factor of Unit 1 at 76% and at 90% from January to October 1983. Duquesne places earlier low capacity factors on taking the plant off-line because of problems of seismic analysis for pipe support. (Tr. 109).

Staff called for the rejection of the proposed contention for being without basis. At the conference, it contended that with Beaver Valley Unit 1 having an operating license for over 7 years, if ECNP wanted to establish that its record was unsafe, such information should have been developed. ECNP in response suggested an NRC 1981 Nuclear Safety Report had cited Beaver Valley management as particularly poor and below the average for reactors on a national rating scale. (Tr. 111).

The foregoing does nothing to alter the focus of the proposed contention. It is bottomed on the theory that the low operating capacity of Beaver Valley Unit 1 adversely reflects on the management capability of Duquesne as operator of the facility and translates into uncertainty as to its ability to operate Beaver Valley Unit 2. ECNP provides no nexus to connect its premise with its conclusion. Low operating capacity does not automatically mean management incapability to operate a facility safely, as Petitioner suggests. No basis was provided to support its assertion that because of a low capacity factor at Unit 1 there is uncertainty as to the management capability of Duquesne to operate the subject unit safely. Proposed Contention Nine does not present a litigable contention and is therefore denied.

**Proposed Contention Ten**

> Evacuation planning and emergency response capability in the event of an accident exceeding design basis are insufficient to assure the health and safety of the public. A license should be withheld pending demonstration of full evacuation in which the entire population within the emergency planning zone has participated under adverse conditions constituting a worst case evacuation.

The offsite emergency plans cover both Beaver Valley Units 1 and 2 and no change is planned when Unit 2 goes into operation. (Tr. 124). These plans have been available for 2 years. (Tr. 123). The proposed contention was amplified at the special prehearing conference by the statement that ECNP recalls that the publicly available reports of emergency response drills at Beaver Valley Unit 1 in the past have indicated inadequacies of performance and did not involve members of the public subject to evacuation under adverse, worst-case, conditions. (Tr. 116). Because Beaver Valley Unit 2 is assertedly located in a constricted river valley, ECNP believes this requires consideration of the capability of those plans to be executed and to fulfill the need satisfactorily.

The second sentence of the contention was interpreted by the parties to require full evacuation by the public as part of emergency exercises. (Applicant's Response, 9/26/83, at 22). (Staff's Response, 9/29/83, at 11-12). ECNP acknowledges that NRC regulations (Appendix E, 10 C.F.R. Part 50) do not mandate public participation, but ECNP states that this does not preclude the opportunity for public participation. ECNP believes that, while it is not required, the entire population should be included but that ECNP has "not used wording to indicate that such should be mandatory." (Tr. 115-17).

Applicant opposed Contention 10 (Applicant's Response, 9/26/83, at 22-23) as being a totally unspecific, generalized allegation and inconsistent with Commission regulations. Applicant referred to ECNP's statements with respect to inadequacies in Beaver Valley Unit 1 emergency plans as vague and a last-minute attempt to broaden the contention. (Tr. 119). They ask, What reports? When were they issued? Which exercises? What were the deficiencies? How is it related to contention? (op. cit.)

Applicant points out that NRC regulations have never required full evacuation by the public and that, indeed, the regulations specifically exclude mandatory public participation (Appendix E to 10 C.F.R. Part 50, § F.1). Applicant further points out that this Commission has twice denied petitions in which ECNP participated to amend NRC rules in this respect (40 Fed. Reg. 43,778 (1975) and 44 Fed. Reg. 32,486 (1979)). Applicant states that newly issued FEMA rules exclude public participation. (Tr. 117-18).

Staff concurs with Applicant in opposing Contention 10. (Staff's Response, 9/29/83, at 11-12). Staff claims there are no specifics to litigate based on the first sentence of proposed Contention 10 and, if there was something wrong with the Beaver Valley Unit 1 emergency plan, ECNP has to let us know what it is. (Tr. 119-20). Staff also states that NRC does not require a demonstration of full evacuation, under adverse conditions, of the entire population (Staff's Response, 9/29/83, at 12). Noting that ECNP seems not to contend this as mandatory, Staff points out that ECNP has not advanced any reason for making full evacuation a special requirement for Beaver Valley. (Tr. 120).

The Board concurs with the Staff and Applicant that Contention 10 as originally proposed and as amplified by ECNP at the special prehearing conference is without foundation in any specifics, but is only a vague, broad, unsubstantiated allegation. ECNP makes a reference to a constricted river valley but does not identify how emergency plans fail to consider this factor or any others that ECNP considers to be a basis for its allegation of insufficiency. ECNP has failed to take advantage of the additional time granted by the Board to perfect this proposed contention.

Although ECNP seems to deny this intent, this Board fails to see how proposed Contention 10, as written, can be interpreted, other than as seeking a mandatory requirement when it clearly states "[a] license should be withheld . . . ." As Applicant and Staff have pointed out, mandatory full evacuation of the entire population has been purposely excluded from NRC and FEMA regulations. If, as Staff suggests, ECNP may be proposing that a special requirement should be adopted for Beaver Valley Unit 2, then the Board finds insufficient justification advanced by ECNP for this attack on regulations. The Board denies admission of Contention 10 for the above reasons.

**Proposed Contention 11**

> The potential for cumulative radiation exposures of residents of the Beaver Valley area in excess of permitted levels as a result of the operation of the two Beaver Valley units plus activities associated with the proposed and pending decommissioning of Shippingport has not been properly assessed. Until the Applicant has demonstrated that such potential multiple exposures will not result in adverse health effects for the residents of the surrounding area, an operating license should not be issued.

ECNP believes that prior studies have not been done in concert in such a manner as to provide an adequate amount of information concerning the permissible, normal routine releases from the operating reactor, plus any reliable estimation of the releases that may be associated with the decommissioning of a reactor such as Shippingport. (Tr. 126). ECNP is also concerned with the best estimate possible for the summation of nonroutine accidental releases based on operational histories of these plants and other reactors. (Tr. 126-27). ECNP is particularly concerned that there has been no experience with the decommissioning of a plant of the type and size of Shippingport. (Tr. 127). ECNP thinks the Applicant should be required to review the operational history for normal and abnormal release events across the range of commercially licensed operating reactors, plus those that have been under the control of the Defense Department. ECNP claims information on Shippingport and military reactors is not readily available to intervenors. (Tr. 127-127(a)).

Applicant opposes this contention as totally vague and unspecific in that it fails to specify what the "potential for cumulative radiation exposures" is, to define or identify "permitted levels" of radiation exposure that will potentially be exceeded, or to specify the exposures ECNP believes would result from operation of the Beaver Valley Units and decommissioning of the Shippingport Atomic Power Station (Applicant's Response, 9/26/83, at 24). Applicant refutes ECNP's claim of a lack of availability of information on Shippingport (Applicant's Response, 9/26/83, at 25; Tr. 128-29, 132-34).

Applicant points out that the Department of Energy completed the Final Environmental Impact Statement for decommissioning the Shippingport Atomic Power Station in May 1982 (DOE/EIS 0080F). (Applicant's Response, 9/26/83, at 25). That EIS shows the radiation dose to the residents of the Beaver Valley area from the decommissioning would be trivial — less than $1 \times 10^{-4}$ man-rems per year (DOE/EIS at 2-13). (Applicant's Response, 9/26/83, at 25). Cumulative doses from operation of Beaver Valley Units 1 and 2 are in the final environmental statements for Unit 1 operation and Unit 2 construction, both of which were issued in 1973. (Tr. 128-29). Applicant refers also to litigation of combined releases from operation of the two Beaver Valley units and Shippingport at the Beaver Valley Unit 2 construction permit phase where the Licensing Board found that "accumulation, if any, is so small it may be disregarded." (Applicant's Response, 9/26/83, at 26.). ECNP was a party to the Unit 2 construction permit proceeding, LBP-74-25, 8 AEC at 712 (1974). Applicant states that the effect from Shippingport decommissioning is a much smaller number than Shippingport operation (Tr. 129).

Staff also opposes this contention as being unclear and not specific. (Staff's Response, 9/29/83, at 12). Allowable releases of radiation from operating reactors are set forth in Part 20 and Part 50, Appendix I. Staff points out that a challenge to these limits is inappropriate in this licensing proceeding and, furthermore, ECNP has not provided any reason to believe that the cumulative effects of those limits may be harmful to residents of the Shippingport area. If ECNP is questioning the plant's capability to meet those limits, Staff says there is no specific information which would lead one to believe that the plant might have trouble meeting either of the limits. (Tr. 131). The Staff concludes this contention utterly fails to provide any reason to believe the operation of Beaver Valley Unit 2 will have cumulative detrimental effects on Beaver Valley area residents and should be rejected (Staff's Response, 9/29/83, at 12).

The Board finds that this contention is fatally deficient in specifics. Levels of permitted radiation exposure that will potentially be exceeded are not identified and specific reasons for believing there will be excesses are not explicit. The Board rejects ECNP's claim that information was not available. The FES for decommissioning Shippingport was available in May 1982 and environmental reports for Beaver Valley Units 1 and 2, in 1973. ECNP has not indicated any specific deficiency in any of these reports. They fail to indicate there will be a cumulative detrimental effect on Beaver Valley area residents. The Board sees no need to review all military and commercial reactor data since specific reports are available dealing with Shippingport decommissioning and operation of Beaver Valley Units 1 and 2, the specific reactors whose cumulative effects are questioned.

For reasons stated above, the Licensing Board rejects Contention 11.

## DETERMINATION OF THE PENNSYLVANIA PETITION

In response to the Notice of Opportunity for Hearing, Pennsylvania solely petitioned for leave to participate in the captioned proceeding as an interested State under 10 C.F.R. § 2.715(c). The section provides, *inter alia,* that a presiding officer will afford representatives of an interested State a reasonable opportunity to participate and to introduce evidence, interrogate witnesses, and advise the Commission without requiring the representative to take a position with respect to the issue. It further allows the presiding officer to require such representative to indicate with reasonable specificity, in advance of the hearing, the subject matters on which the representative desires to participate.

Applicant did not object to Pennsylvania's participation as an interested State in the event a hearing is held. Neither did Staff object. At the special prehearing conference we accepted Pennsylvania as a participant under 10 C.F.R. § 2.715(c). (Tr. 146).

Pennsylvania, on November 30, 1983, voluntarily filed with the Licensing Board a statement of its concerns pursuant to its responsibility to protect public health and safety. The State reported it had a particular interest in ECNP's proposed Contentions 6, 7, 8 and 10. It did nothing to elaborate on the proposed contentions, to modify or to adopt them. Its stated interest was accompanied by a caveat, "the Commonwealth does not specifically adopt or endorse the language of the contentions. Further, in highlighting certain contentions, the Commonwealth does not intend to either restrict the scope of its participation in the proceeding, or imply a position regarding the merits of any of the proposed contentions."

Pennsylvania in stating it had a particular interest in ECNP's proposed contentions 6, 7, 8 and 10 did nothing to cure their deficiencies as proposed contentions and they remain legally insufficient for the reasons previously discussed. The Pennsylvania filing does nothing to alter their status as nonlitigable and for which they were rejected from this proceeding.

The filing and acceptance of the Pennsylvania petition pursuant to 10 C.F.R. § 2.715(c) only permits it to participate in the adjudicatory hearing if one is held. The Atomic Energy Act of 1954, as amended, does not prescribe a mandatory hearing for deciding an operating license application. Section 189a. A need for a hearing has not been established in this proceeding. No petitioner has submitted a litigable contention as required by 10 C.F.R. § 2.714, to necessitate the holding of a hearing. The filing and acceptance of the Pennsylvania petition to participate under the provision of Section 2.715(c) does not trigger a hearing. *See Northern States Power Co.* (Tyrone Energy Park, Unit 1), CLI-80-36, 12 NRC 523, 527 (1980); *Niagara Mohawk Power Corp.* (Nine Mile Point Nuclear Station, Unit 2), LBP-83-45, 18 NRC 213, 216 (1983). The State has not sought a hearing in this matter. It opted to have the Licensing Board explore proposed contentions of a petitioner, which, after review, were found not to warrant consideration because they failed to meet Commission standards. Pennsylvania could have sought full party status under 10 C.F.R. § 2.714, for filing its own contentions, which it chose not to do. *See Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-354, 4 NRC 383, 392 (1976).

The acceptance of Pennsylvania as a participant to the proceeding as an interested State under 10 C.F.R. § 2.715(c) and its filing of a statement of concerns has not presented the Licensing Board with anything to adjudicate and for which a hearing should be held.

## DISPOSITION OF THE WALKER PETITION

In response to the Notice of Opportunity for Hearing of June 1, 1983, Ralph P. Walker, an individual of New Brighton, Pennsylvania submitted a petition to intervene. It complained of: unspecified extra costs from Beaver Valley Unit 1 being passed to consumers; Beaver Valley Unit 1 posing a constant threat to health and safety in parts of three States; and the high cost of electricity discouraging business and industry from locating in the area, and presenting a serious problem to low- and fixed-income customers. He urged that Beaver Valley Unit 2 never be licensed as a nuclear facility and instead its design be altered to make it into a coal-fired facility. The petition had attached four pages of what are basically copies of news articles.

Applicant responded to the petition asserting it failed to set forth Mr. Walker's interest in the proceeding or how the interest might be affected as required by 10 C.F.R. § 2.714. Duquesne considered the petition to be outside the scope of the proceeding because it concerned Unit 1, and his economic interest as a ratepayer, a matter not relevant to the NRC.

Staff's position in response was that Mr. Walker resides close enough to the facility to establish standing if he had alleged some specific interest within the protection of the Commission that may be harmed if Beaver Valley Unit 2 is granted a license. It concluded Petitioner had not alleged any interest that can be affected by the outcome of this proceeding, thus failing to establish standing. Staff also asserted Walker failed to identify, as required, specific aspects of the subject matter of this proceeding, as to which he intends to participate.

Mr. Walker did not respond to our order of August 4, 1983 requiring petitioners to file amendments and/or supplements to their petitions, including proposed contentions by September 9, 1983 and to appear at the special prehearing conference of October 12, 1983. No explanation was received for the failure to comply.

At the special prehearing conference, Applicant moved that Mr. Walker be dismissed from further participation and his petition for leave to intervene be denied for failure to comply with the Licensing Board's order to submit contentions. We deferred ruling on the motion in order to take it up in this report.

Mr. Walker sent a letter to the Commission, dated September 28, 1983, but date-stamped as being received by the NRC on October 31, 1983, expressing concern about the effect Beaver Valley Unit 2 would have on taxpayers and rate structures. He also stated that nuclear plants release radiation that could endanger and harm small or large parts of the population and requested that the Commission identify the physical parts of Beaver Valley Unit 2 which would cause releases of harmful radiation due to human error. Mr. Walker also asked the Commission to describe what could happen in case of a meltdown. Attached to the letter were copies of three pages of articles taken from material mailed by Duquesne advising about its nuclear operations.

Applicant responded to the letter it received on October 31, 1983, arguing it provides no grounds to support Mr. Walker's further participation in the proceeding. Duquesne stated he did not offer a contention which meets the basis and specificity requirements of the Commission's regulations or which has a nexus to Beaver Valley Unit 2. It requested that the pending motion to dismiss Mr. Walker from the proceeding be granted.

Section 2.707 of Title 10 of the Code of Federal Regulations governs actions that may be taken in the event of a default. It provides that on failure to file a pleading within the time described or to appear at a prehearing conference, the presiding officer may make such orders in regard to the failure as are just. We have determined to decide the petition on the basis of what was filed rather than to treat with the matter solely as a default.

In order to be admitted as a party intervenor in an application proceeding, a petitioner, under 10 C.F.R. § 2.714(a)(2) must show the interest of the petitioner in the proceeding, how that interest may be affected by the results of the proceeding, including why petitioner should be permitted to intervene and the specific aspect of the subject matter of the proceeding as to which petitioner wishes to intervene.

These considerations require a showing that the action being challenged could cause injury in fact to the petitioner, and that such injury is arguably within the zone of interest, protected by the Atomic Energy Act or the National Environmental Policy Act. *Worth v. Seldin*, 422 U.S. 490 (1975); *Sierra Club v. Morton*, 405 U.S. 727 (1972).

Close geographical proximity of a petitioner's residence to the facility is sufficient to satisfy the interest requirements of 10 C.F.R. § 2.714. *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-522, 9 NRC 54 (1979). There is no dispute that New Brighton, Pennsylvania is in close proximity to the subject facility, and official notice is taken of that fact. They are sufficiently close that Petitioner's interest may be inferred. Mr. Walker could have relied upon proximity of the facility to establish interest but chose to particularize the matter, none of which concerns is within the protection of the NRC and for which no relief can be granted.

His concern about costs and safety of Beaver Valley Unit 1 are not relevant to this proceeding involving an operating license for Beaver Valley Unit 2. Assuming his complaint about the high cost of electricity relates to the subject facility, it does not establish interest and standing. The statutes under which the Commission functions do not afford protection to a ratepayer. *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-582, 11 NRC 239, 243 n.8 (1980). His request that Beaver Valley Unit 2 not be licensed as a nuclear facility and instead its design be altered to make it into a coal-fired facility cannot be considered for two reasons. Mr. Walker presented no basis for not licensing the plant, and the request to alter the design to make it into a coal-fired facility is beyond the authority of this agency.

Another ground for denying the petition for intervention is that Mr. Walker failed to satisfy the requirements of 10 C.F.R. § 2.714(b) which denies participation as a party where one admissible contention had not been provided. Petitioner has failed to submit an admissible contention despite having been given the opportunity to do so.

The Walker letter received after the special prehearing conference does nothing to alter the above conclusions. Even if it were considered as a supplemental petition, for which there is no basis, the letter provides nothing new for consideration. The areas of interest expressed in the letter are again the effect on the ratepayer and an inquiry about human error, a matter on which Mr. Walker wants information. It contains nothing to satisfy the requirements of 10 C.F.R. § 2.714.

The petition of Ralph P. Walker is denied because it failed to establish interest and standing and did not contain a litigable contention, all contrary to the provisions of 10 C.F.R. § 2.714. This finding renders moot Applicant's motion to deny the petition because Mr. Walker failed to file a proposed contention, as required by order of the Licensing Board, and it is hereby denied for that reason.

## CONCLUSION

For the reasons set forth in the review of each of the petitions for intervention, for participation, and the holding of a hearing in the subject proceeding, no basis in fact or law has been provided for granting the relief sought. The concerns raised by the Petitioners should adequately and effectively be reviewed and treated through the standard review procedure of the Agency.

## FINDINGS AND ORDER

Upon consideration of all of the foregoing, with all judges concurring, it is hereby found:

1. OCRE has failed to submit a litigable contention and its petition to intervene as a party is denied under 10 C.F.R. § 2.714(b).
2. ECNP has failed to establish its standing and interest to intervene in the proceeding as required by 10 C.F.R. § 2.714(a) and did not submit a litigable contention as called for by 10 C.F.R. § 2.714(b), for which its petition to intervene as a party is denied.
3. The acceptance of Pennsylvania as a participant in the proceeding as an interested State under 10 C.F.R. § 2.715(c), and its filing of a statement of concerns has not presented the Licensing Board with anything to adjudicate and for which a hearing should be held.
4. Walker has failed to establish his standing and interest to intervene in the proceeding as required by 10 C.F.R. § 2.714(a) and did not submit a litigable contention as called for by 10 C.F.R. § 2.714(b), for which petition to intervene as a party is denied.
5. No basis in fact or law has been provided for holding an adjudicatory hearing in this matter. No hearing shall be held in this operating license application case and the matter shall be dismissed.

It is so *Ordered.*

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Morton B. Margulies, Chairman
ADMINISTRATIVE LAW JUDGE

Dated at Bethesda, Maryland,
this 27th day of January 1984.

This order is appealable under the provisions of 10 C.F.R. § 2.714 to the Atomic Safety and Licensing Appeal Board within ten (10) days after service of the Order. *See* 10 C.F.R. § 2.710.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**James L. Kelley,** Chairman
**Dr. James H. Carpenter**
**Glenn O. Bright**

**In the Matter of** **Docket Nos. 50-400**
**50-401**
**(ASLBP No. 82-468-01-OL)**

**CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY**
**(Shearon Harris Nuclear Plant, Units 1 and 2)** **January 27, 1984**

The Licensing Board rules on several motions for summary disposition concerning health effects associated with normal operation of a nuclear power plant, granting them in part and denying them in part. The Board found that under the circumstances they would be warranted in calling their own expert witness to the evidentiary hearing in order to ensure substantive consideration of the issues.

## RULES OF PRACTICE: SUMMARY DISPOSITION

Because the proponent of a motion for summary disposition has the burden of demonstrating the absence of a genuine issue of material fact, it does not necessarily follow that a motion supported by affidavits will automatically prevail over an opposition not supported by affidavits. The Board must scrutinize the motion to determine whether the movant's burden has been met.

## RULES OF PRACTICE: SUMMARY DISPOSITION

An opponent of a summary disposition motion must set forth specific facts showing that there is a genuine issue of fact. It would frequently not be sufficient for an opponent to rely on quotations from or citations to published work of researchers who have apparently reached conclusions at variance with the movant's affiants. Such public work is typically produced with other objectives in mind and may not focus directly on the precise issue in contention. While a licensing board may, in its discretion, consider publications referenced in opposition to (or in support of) a motion for summary disposition to determine whether a movant has met its burden, it is under no obligation to do so.

## RULES OF PRACTICE: SUMMARY DISPOSITION; HEALTH EFFECTS

The Commission's decision in *Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), CLI-80-31, 12 NRC 264 (1980) has the effect of differentiating health effects contentions from other contentions in the summary disposition context. An opponent of summary disposition in the health effects area must have some new (post-1975) and substantial evidence that casts doubt on the BEIR Report estimates. Furthermore, he must be prepared to present that evidence through qualified witnesses at the hearing.

## LICENSING BOARDS: AUTHORITY TO CALL WITNESSES

Adjudicatory boards should give the Staff every opportunity to explain, correct, or supplement its testimony before resorting to outside experts of their own, and must articulate good reason to suspect the validity and completeness of the Staff's work. A board must be satisfied that it has no realistic alternative to call in a board witness, that it simply cannot otherwise reach an informed decision on the issue involved.

## TECHNICAL ISSUES DISCUSSED

Cancer Risk Estimates.

# MEMORANDUM AND ORDER

**(Ruling on Motions for Summary Disposition of Health Effects Contentions: Joint Contention II and Eddleman Contentions 37B, 8F(1) and 8F(2))**

## I. INTRODUCTION

### A. The Pleadings

Joint Contention II and Eddleman Contentions 37B, 8F(1) and 8F(2) concern various health effects associated with the normal operation of the Shearon Harris nuclear plant. Joint Contention II and Eddleman Contention 37B challenge the NRC Staff's assessment in its environmental impact statement of the health effects of routine radiation releases during normal operation of the plant. Eddleman Contentions 8F(1) and 8F(2) address the Staff's assessment of the health effects associated with the uranium fuel cycle. Contention 8F(1) concerns the health effects of coal particulates emitted by coal-burning power plants involved in the fuel cycle; Contention 8F(2) questions the Staff's assessment of the health effects of the radiation released during the fuel cycle.

The Applicants have filed motions for summary disposition of all of these health effects contentions, supported by affidavits from technical experts and a memorandum of law. The NRC Staff filed responses in support of the motions on three of the contentions, and a separate motion for summary disposition on Contention 8F(1). The Staff's responses and motions were also supported by affidavits from technical experts. The Joint Intervenors and Mr. Eddleman filed responses in opposition to each of the motions for summary disposition, including a memorandum of law. These responses in opposition were not supported by affidavits; instead, they relied primarily on references to publications by persons apparently holding views contrary to those expressed by the affiants for the Applicants and Staff. Copies of some of these publications were provided.

Our initial review of the Intervenors' responses to the motions for summary disposition of Joint Contention II and Eddleman Contention 37B raised questions whether there was any realistic prospect that a Shearon Harris hearing on these complex generic contentions might be worthwhile. Therefore we issued an order directing the parties' attention to the Commission's *Black Fox* decision, *Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), CLI-80-31, 12 NRC 264, 277 (1980), and stating that: "[a]s we read *Black Fox*, an Intervenor seeking to withstand a well-supported motion for summary disposition must present some substantial evidence or at least 'present thinking' that raises serious questions about the moving party's position." November 23, 1983 unpublished Order at 2. We noted that the Applicants had provided substantial expert opinion in their affidavits supporting their motions and that the Intervenors had not presented any affidavits or anything else indicating their ability to present admissible evidence in support of their oppositions. We also informed the Intervenors that their approach to the health effects contentions was too wide-ranging and suggested that they narrow their focus. We gave the Intervenors an opportunity to file an additional response to the summary disposition motions indicating (1) the names of expert witnesses they will present on the health effects contentions and the specific issues those experts will address; and (2) which of Intervenors' "most critical disputes" listed in their responses to the motions will be the subject of expert testimony.

The Intervenors subsequently filed a response in which they identified their expert witnesses for Joint Contention II and Eddleman Contention 37B, as well as for Eddleman Contention 8F(2), and the issues on which those experts are expected to testify. This response did not significantly narrow the focus of matters the Intervenors wish to put in issue. The Applicants filed a response to the Intervenors' response arguing its insufficiency.

## B. General Principles

The proponent of a motion for summary disposition has the burden of demonstrating the absence of a genuine issue of material fact. It does not necessarily follow, therefore, that a motion supported by affidavits will automatically prevail over an opposition not supported by affidavits. In that situation, the Board must nevertheless scrutinize the motion to determine whether the movant's burden has been met. *See Adickes v. S.H. Kress & Co.*, 398 U.S. 144, 156-61 (1970); *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 752-54 (1977).

It is also true, however, that in the face of persuasive affidavits in support of a motion for summary disposition, an opponent of the motion runs a high risk of defeat if he fails to produce persuasive rebuttal affidavits. Under the controlling Rule of Practice, "a party opposing the motion may not rest upon the mere allegations or denials of his answer." Rather, the opponent's answer "must set forth specific facts showing that there is a genuine issue of fact." 10 C.F.R. § 2.749(b). In that connection, it would frequently not be sufficient for an opponent to rely on quotations from or citations to published work of researchers who have apparently reached conclusions at variance with the movant's affiants. *See generally* Wright, Miller & Kane, Federal Practice and Procedure, § 2722 (1983). For example, in the case of a health effects contention questioning the BEIR Committee conclusions, the opponent of a motion for summary disposition who wishes to rely on the works of such researchers as Gofman, Morgan or Bertell should obtain affidavits from those researchers. To be sure, that may involve difficulties and expense. But such difficulties and expense would be no greater than those involved in meeting the obligation to produce competent witnesses at a later hearing. The Intervenors should be aware that copies of the published work of researchers supporting their positions would not be admissible as substantive evidence at a hearing unless the researchers themselves were produced as supporting witnesses. Moreover, such published work is typically produced with other objectives in mind and may not focus directly on the precise issue in contention. For these reasons, while a licensing board may, in its discretion, consider publications referenced in an opposition to (or in support of) a motion for summary disposition to determine whether a movant has met its burden, it is under no obligation to do so.

The foregoing considerations are generally applicable to summary disposition motions on contentions in NRC proceedings. As we understand the Commission's *Black Fox* decision, cited above, an additional requirement of — for lack of a better term — "substantiality" must be met by the opponent of health effects contentions of the stripe now before us. In *Black Fox*, the Commission adopted a policy that "unnecessary adjudication [of health effects] should be avoided," noting that it would serve "no useful purpose to litigate [health effects] when there is no serious contest as to the result." 12 NRC at 277. The Commission went on to state that:

> [W]e believe that a Licensing Board could take official notice that releases within Appendix I levels result in radiation exposures that are small fractions of doses from natural background radiation and that the 1972 BEIR Report contains a "generally accepted evaluation of the effects of ionizing radiation." This does not mean of course that health effects of Appendix I releases cannot be contested. It only means that litigation regarding these issues need not begin on a clean slate, and that, for example, the BEIR estimates can be relied on in the absence of a contest and may be used, along with any other evidence, in ruling on summary disposition motions and rendering initial decisions.

*Id.* (Footnote omitted.) The Commission also noted that the Appendix I rulemaking was then (in 1980) 5 years old and that the hearing process might be a useful way to bring "present thinking" to bear on these health effects issues. As the Commission saw it, this would permit "the interested parties to present *the best available evidence on health effects*" in individual licensing cases. *Id.* (Emphasis added.)

The Commission's *Black Fox* decision, as we read it, has the effect of differentiating health effects contentions from other contentions in the summary disposition context. Under the rule (10 C.F.R. § 2.749) and licensing board practice, the mere existence of a material issue of fact, whether raised by the opponent or by a gap in the movant's showing, defeats the motion at least in part and entitles the opponent to a hearing. This is true whether or not the opponent has any substantive evidence to offer; indeed, he may, and frequently does, "make his case" entirely on the basis of cross-examination. By contrast, *Black Fox* says to us that an opponent of summary disposition in the health effects area must have some new (post-1975) and substantial evidence that casts doubt on the BEIR estimates. Furthermore, he must be prepared to present that evidence through qualified witnesses at the hearing. As we stated earlier: "It will not suffice merely to present an opposing case based entirely on cross-examination by a non-expert. Given the very complex nature of this generic issue, there is no reason to believe that such cross-examination alone will add anything to the sum of human knowledge on health effects." Order of November 23, 1983, at 2.

### C. The Motion Papers — General Considerations

We discuss in the next section each element of the health effects contentions, and make rulings in light of the principles we have just outlined. Before turning to that particularized analysis, however, we discuss certain general considerations that arose out of our reading of the motion papers.

As we indicated earlier, the Applicants' motion papers discuss the health effects contentions at length and in detail; they are supported by affidavits of seemingly well-qualified experts. The Staff's supporting response is likewise buttressed by expert affidavits.

By contrast, the Intervenors' opposition papers were not supported by affidavits or by any clear indication of how they proposed to rebut the Applicants' case at a hearing. The numerous references to published articles suggested that the Intervenors expected to introduce such articles without necessarily producing the author as a witness. Beyond that, the Intervenors' opposition papers were extremely broad and unfocused, seeking to contest virtually every aspect of the case, apparently without regard to the comparative importance of issues or the Intervenors' likely ability to make a contribution. *See,* in particular, Joint Intervenors' Response to Motion on Contention II at 7-10.

At that juncture, we had serious doubts whether the Intervenors would be able to mount a substantial challenge to the positions of the Staff and Applicants and to the BEIR Committee analyses on which they were based. We accordingly directed the Intervenors to supplement their opposition by telling us who they expected to produce as expert witnesses in the event of a hearing, and what issues each witness would address.

*The Intervenors' Proposed Witnesses*

In response to the Board's request, the Intervenors have stated their intention to call Dr. Ernest Sternglass as their lead witness and as an expert on a long list of matters, including most of the "critical disputes" they seek to litigate. Dr. Sternglass has been attempting to challenge assessments of the health effects of low-level radiation in NRC proceedings for over a decade. Time and again, his methodology has been found deficient and his conclusions of no value. The Appeal Board had this to say about Dr. Sternglass in *Trustees of Columbia University in the City of New York,* ALAB-50, 4 AEC 849, 859 (1972), *aff'd sub nom. Morningside Renewal Council, Inc. v. AEC,* 842 F.2d 234 (2d Cir. 1973), *cert. denied,* 417 U.S. 951 (1974):

> The Appeal Board is of the opinion that Dr. Sternglass' assertions have no valid scientific foundation. We find that the methodology employed is deficient, that many of the assertions are inconsistent and even self-contradictory, and his statistical methodology and selective sampling techniques are not scientifically credible.

Dr. Sternglass' positions were similarly rejected in *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-156, 6 AEC 831, 850 (1973) (*citing Columbia University*); *Toledo Edison Co.* (Davis-Besse Nuclear Power Station), 4 AEC 571, 585 (1971); and more recently in *Punnett v. Carter,* 621 F.2d 578, 583-86 (3d Cir. 1980).

Intervenors' December 5, 1983 response does not indicate that Dr. Sternglass has any new information to offer in the health effects area, or that he has foresworn the pseudo-scientific methods he has espoused in the past. Given Dr. Sternglass' track record in other NRC proceedings, there is no reason to think that his testimony could make any constructive contribution to this case. If Dr. Sternglass were the only person available as an opposing expert witness, we would grant the motions for summary disposition, notwithstanding the existence to some disputes over material facts. Since we are denying the motions in certain respects, the Intervenors may proffer Dr. Sternglass as a witness, subject to the possibility that he may not withstand a *voir dire* challenge.

The Intervenors' next proposed witness is Dr. Carl Johnson, an Associate Clinical Professor at the University of Colorado School of Medicine. Although not proffered over quite as wide a range as Dr. Sternglass, the Intervenors expect Dr. Johnson to testify on a number of diverse subjects. Some of these subjects appear to be irrelevant to the admitted contention — *e.g.*, radioactive releases from the Oyster Creek facility in New Jersey, the efficiency of the exhaust filters at the Harris facility. Another proposed topic appears to represent an attack on the rules — "that the current NRC standards for radiation exposure to the public are not adequately protective." Other proposed topics for Dr. Johnson may be within the admitted contentions, but not within those few parts that are surviving the summary disposition motions.

It is not clear from the Intervenors' sketchy submission about Dr. Johnson whether he is qualified to testify on these surviving parts of the admitted contentions. Assuming that he may be, however, Dr. Johnson's presentations in recent NRC proceedings do not give us a basis for confidence that he would make a substantial contribution to the case. In the *Waterford* case, *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076 (1983), Dr. Johnson's testimony was "generally critical of the health risk estimates that have been made in connection with projected routine radiation releases from Waterford" (*id.* at 1089)— also the crux of the controversy in this case. The Appeal Board found, however, that

> A principal source of Dr. Johnson's criticism of the health risk estimates is a study by the Heidelberg (West Germany) Institute for Energy and Environmental Research. Dr. Johnson's cross-examination, however, revealed his lack of familiarity with the methodology of that study and the extent of its acceptance *vel non* by the scientific community. Tr. 1948-54. Dr. Johnson was similarly unconversant with the Commission's regulations on the control of radiation emissions and the methodology for determining dose estimates. He was also not aware of the staff's and applicant's consideration of all the various ingestion pathways in their population dose estimates for Waterford (*see* p. 1084, *supra*), nor has he attempted to determine such estimates on his own. *See* Tr. 1853-55, 1875-76, 1886-87, 1901-12, 1947, 1964-65, 1994-95, 2002-03, 2006-07. In short, we find Dr. Johnson's testimony to be of essentially no value with respect to the staff and applicant dose estimates for Waterford 3.

17 NRC at 1090. *See also United States Department of Energy* (Clinch River Breeder Reactor Plant), LBP-83-8, 17 NRC 158, 222 (1983) where Dr. Johnson's attempted "apples and oranges" comparison between the Rocky Flats facility and the CRBR was rejected.

On the basis of the papers before us and particularly Dr. Johnson's recent appearances in NRC proceedings, we cannot conclude that his testimony would satisfy the *Black Fox* test of substantiality. Indeed, the available indications are to the contrary. As in the case of Dr. Sternglass, however, the Intervenors may proffer Dr. Johnson as a witness at the hearing, subject to *voir dire* challenge.

### *Proposed Testimony on Pain and Suffering*

The Intervenors expect to call Dr. Morris Lipton, a psychiatrist, and Dr. Barbara Wynn, a general practitioner, to testify on "the pain and suffering aspect of 37B." The apparent thrust of this testimony will be towards "the pain and suffering undergone by victims of cancer and other diseases." Intervenors' Response at 7.

Whether the subject of pain and suffering may be litigated under the admitted health effects contentions has not been squarely ruled on until now. The question did surface in a discovery dispute between the Applicants and Mr. Eddleman. In an October 6, 1983 Order (unpublished) we denied Mr. Eddleman's motion to compel discovery with respect to a series of interrogatories concerning pain and suffering. Without any extended discussion, we ruled that interrogatories on that subject were "either irrelevant or rhetorical, or both." Order at 13. We now address the pain and suffering question directly, with the Intervenors arguing the affirmative and the Applicants the negative of the proposition. We agree with the Applicants for the following reasons.

To begin with, none of the admitted health effects contentions (Mr. Eddleman cites 37B in particular) refer to pain and suffering or encompass it by fair implication. Rather, those contentions address the kinds and extent of diseases allegedly caused by nuclear plant radiation. By contrast, at the initial stage of the case we rejected proposed Contention 37A — which referred both to psychological stress and to pain and suffering associated with cancer — on the basis of the Commission's Policy Statement on Psychological Stress. LBP-82-119A, 16 NRC 2069, 2096 (1983). Upon a request for reconsideration, we deferred any further ruling, pending issuance of the Staff's DES. Order of January 11, 1983 (unpublished). Although we suggested at that time that the question might be discussed at an upcoming prehearing conference, no such discussion occurred. The Staff's DES issued thereafter and, under the Board's standing instructions applicable to deferred rulings on contentions, Mr. Eddleman was then obliged to advise the Board whether Contention 37A was being submitted for ruling, revised or withdrawn. 16 NRC at 2072-73. Mr. Eddleman did file a response to the Staff's DES, dated June 20, 1983, in which he discussed the effect of the DES on many of his contentions. Although he stated generally that "no contentions are withdrawn herein," he did not discuss or even refer to

Contention 37A. Thus he did not comply with the Board's standing instructions with respect to that deferred contention. As a result of this default, reconsideration of our initial ruling on 37A is denied and that contention remains rejected.

In his June 20 response to the DES, Mr. Eddleman did discuss his admitted Contention 37B, indicating his view that that contention encompasses "pain and suffering." A similar reference is contained in Mr. Eddleman's filing on the "five factors" dated July 29, 1983. However, a party may not inject a new element into an admitted contention by his own bootstrap assertion. This Board has the final say on the scope of contentions. To repeat, "pain and suffering" are not referred to in Contention 37B, the thrust of which is toward diseases other than cancer allegedly being caused by radiation. As a result, "pain and suffering" is not presently a litigable issue in this case and the proposed testimony of Drs. Lipton and Mills would not be germane.

In holding that pain and suffering evidence is not admissible under the admitted contentions, we do not mean to imply a legal conclusion, as suggested by the Applicants, that pain and suffering are simply not litigable under NEPA. The Supreme Court's recent decision holding psychological stress beyond the reach of NEPA rested largely on the fact that the stress involved there grew out of apprehensions over risks of accidents that might never occur. *See Metropolitan Edison Co. v. People Against Nuclear Energy,* 75 L. Ed. 2d 534 (1983). The Court concluded that Congress had not intended to reach such attenuated effects. But if it were to be shown, for example, that routine releases from a nuclear plant do cause large numbers of cancer deaths, it would be but a short next step to consider the pain and suffering associated with those deaths.

In any event, we think it would make little sense to attempt to litigate such a subjective matter as pain and suffering until one has first established a solid factual predicate for that inquiry. Pain and suffering of the orders the Intervenors seek to prove in this case assume that the facility will cause cancers and other diseases in large numbers of people, far larger than the numbers predicted by the Staff on the basis of the BEIR estimates. Given the weight *Black Fox* authorizes us to attach to the BEIR estimates, there would be no practical justification for embarking on pain and suffering litigation at this point. Should the Intervenors refute the Staff's estimates and establish at a hearing that far larger numbers of lethal cancers and other diseases will ensue from the plant's operation, we can reconsider then whether the associated pain and suffering should be weighed in the NEPA balance.

### D. The Board's Proposed Witness

As discussed below, the Board is denying the motions for summary disposition in several respects — including in particular the Staff's estimates of radiation-induced cancer, which are in turn derived from the BEIR I estimates. We bind this issue over for hearing because Dr. John Gofman's recent estimates of radiation-induced cancers conflict sharply with those of the Staff, and that conflict produces a material issue of fact. *See* Gofman, Radiation and Human Health, 314 (1983). Its exploration at a hearing promises to be a constructive exercise, provided Dr. Gofman can appear as a witness to discuss and explain his work. To that end, the Board proposes to call Dr. Gofman as a Board witness.

The Board is mindful of the limitations on its authority to call its own witnesses. As the Appeal Board explained in the *Summer* proceeding, the Commission's established framework for licensing proceedings, "gives the staff, as a representative of the public interest, a dominant role in assessing the radiological health and safety aspects of the involved facilities." *South Carolina Electric and Gas Co.* (Virgil C. Summer Nuclear Station, Unit 1), ALAB-663, 14 NRC 1140, 1156 (1981). The improvident exercise of our power to call expert witnesses could undermine the Staff's role and perhaps duplicate its work. *Id. Summer* teaches that "adjudicatory boards should give the staff every opportunity to explain, correct, or supplement its testimony *before* resorting to outside experts of their own." *Id.* (Emphasis in original). Furthermore, "boards must articulate good reason to suspect the validity and completeness of the staff's work." *Id.* Finally, the board must be satisfied that it has no realistic alternative to calling a Board witness, that it "simply cannot otherwise reach an informed decision on the issue involved." 14 NRC at 1163. As we shall explain, we believe that our proposed action in calling Dr. Gofman as a witness conforms with the *Summer* criteria insofar as they are applicable here, and that those criteria to some extent do not fit our circumstances.

To begin with, the situation in *Summer* was quite different from the one that faces us. There, a hearing had been held on the merits of the seismic issue that gave rise to the witness dispute, and the Board had been dissatisfied with the Staff's presentation. Although the Staff then offered supplemental testimony, the Board adhered to its decision to call Board witnesses. In our case, we are at the summary disposition phase. The Staff has filed no testimony and we do not know what the Staff would say at a hearing, or who their witnesses would be. All we know of the Staff's position is what they have said in their response and affidavits supporting the motion for summary disposition. This might suggest that the Board defer calling any Board witness, but for one dispositive consideration. Unless Dr. Gofman is to appear on the cancer risk estimate question, there will be no hearing on health effects on this case. We will grant the summary disposition motions under the Commission's guidance in *Black Fox* because the oppositions to those motions are insubstantial and there is no prospect that a hearing would serve any useful purpose.

In these circumstances, there is, of course, no further Staff testimony to wait for. In reviewing the Staff's summary disposition papers, we find the Staff response did not address Dr. Gofman's risk estimates, except in a passing reference to a "somatic risk estimator" attributed to Gofman and others and said to be only "about two times higher than the upper end of the range of values used in the DES." Branagan Affidavit at 4. Dr. Gofman's cancer risk estimates are about 5 to 10 times higher than the BEIR I estimates on which the DES estimates are based. Thus in calling Dr. Gofman as a witness, far from duplicating the Staff's work, we would be focusing on matters the Staff has apparently ignored. For their part, the Applicants' expert, Dr. Fabrikant, does undertake to discredit Dr. Gofman's work, but his broad-brush criticisms do not demonstrate particular flaws in Dr. Gofman's data or methodology. *See* Fabrikant Affidavit at 33-35. Nevertheless, in response to the Appeal Board's *Summer* directive to "give the staff every opportunity to explain, correct, or supplement its testimony," we are giving the Staff a further opportunity to explain its apparent position that Dr. Gofman's cancer risk estimates are not valid and why, if that is the Staff's view, the Board should not call Dr. Gofman as a witness. Any such Staff filing should be made by February 10, 1984.

We add one further point on the Staff's position and role in the process. Lest we be viewed as encroaching on the Staff's technical review territory, it is unimportant to us whether Dr. Gofman comes to the hearing as our witness or the Staff's, provided that is agreeable with Dr. Gofman. Although the Staff may not agree with Dr. Gofman on some substantive issues, the Staff as "a representative of the public interest" may and should from time to time call a witness holding a differing viewpoint, not as the witness' sponsor but simply in the interest of having that viewpoint heard.

The Board believes that Dr. Gofman's appearance as a witness will be critical to an informed decision on the central issue here — the risk estimates for numbers of cancers to be caused by the Shearon Harris facility. Only three cancer risk estimates have been developed to date — those of the BEIR Committee, the United Nations Scientific Committee on the Effects of Atomic Radiation, and Dr. Gofman's. Dr. Gofman's estimates are the most recent and by far the highest. He sets out and compares those estimates in his book, as follows.

| Source of Estimate | Radiation-Induced Cancer Deaths per Million Person-Rads, Delivered to a Population of Mixed Ages |
|---|---|
| BEIR, relative risk method (p. 342)* | 177-353 |
| BEIR, absolute risk method (p. 342)* | 70-124 |
| UNSCEAR (p. 414) | 100 |
| This author (see above) | 3,771 |

*BEIR (1979)

The UNSCEAR value is 37.7 times lower than this author's. The highest BEIR value is 10.7 times lower than this author's, and the lowest BEIR value is 53.9 times lower than this author's.

The Board expresses no view on the merits of any of these estimates, noting only that we are authorized by the Commission's *Black Fox* decision to accept the BEIR estimates in the absence of a contest. We offer these observations, however. First, Dr. Gofman's experience and qualifications, including MD and Ph.D. degrees, are impressive. Second, Dr. Gofman's 1981 book, Radiation and Human Health, runs to 853 pages, addressing a broad range of subjects. At least in the absence of some persuasive reasons *not* to credit Dr. Gofman's estimates, we think this Board should give them close scrutiny.

It might be suggested that we should address the Gofman estimates through the other parties' witnesses. While that is sometimes a feasible approach, we question its sufficiency here. Some expert should present the Gofman work in a reasonably objective manner. Inasmuch as the Applicants and Staff reject the Gofman estimates we doubt whether their experts could address them with the desired degree of objectivity. Moreover, unlike the more usual case where some assigned value is at issue — *e.g.*, is the Hosgri Fault capable of generating a $M_S7$ earthquake — it is a particular researcher's estimates that are at issue here. That researcher, Dr. Gofman, is best able to answer questions about his own work.

## II. ANALYSES AND RULINGS ON THE SUMMARY DISPOSITION MOTIONS AND RESPONSES

### A. Joint Contention II

Joint Contention II is lengthy; it provides as follows:

> The long term somatic and genetic health effects of radiation releases from the facility during normal operations, even where such releases are within existing guidelines, have been seriously underestimated for the following reasons:
>
> (a) The work of Mancuso, Steward, Kneale, Gofman and Morgan establishes that the BEIR-III Report (1980 report of the National Academy of Sciences' Committee on the Biological Effects of Ionizing Radiation, entitled "The Effects on Population of Exposure to Low-levels of Ionizing Radiation") (1) incorrectly understood the latency periods for cancer; (2) considered only expressed dominant genetic defects; and (3) failed to use a supralinear response rather than a threshold or linear-or-less model to determine low-level radiation effects.
>
> (b) Insufficient consideration has been given to the greater radiation effects resulting from internal emitters due to incorrect modeling of internal absorption of radionuclides, and underestimation of the health and genetic effects of alpha, beta and neutron radiation on DNA, cell membranes and enzyme activity. (Reference: sources cited in Eddleman 37(F).)
>
> (c) The work of Gofman and Caldicott shows that the NRC has erroneously estimated the health effects of low-level radiation by examining effects over an arbitrarily short period of time compared to the length of time the radionuclides actually will be causing health and genetic damage.
>
> (d) Substantial increases in cancer mortality rates have been observed in the vicinity of nuclear facilities. Sternglass, "Cancer Mortality Changes Around Nuclear Facilities in Connecticut," February, 1978.
>
> (e) The radionuclide concentration models used by Applicants and the NRC are inadequate because they underestimate or exclude the following means of concentrating radionuclides in the environment; rainout of radionuclides or hot spots; radionuclides absorbed in or attached to fly ash from coal plants which are in the air around the SHNPP site; and incomplete mixing and dispersion of radionuclides.
>
> (f) In computing radionuclides concentration in the environment, less reactive rather than more reactive forms of radionuclides are used in the computation, and certain radionuclides are ignored. (Reference: source cited in Eddleman 37(10)).

The Board's approach to Joint Intervenors' Contention II was first to examine the source term to see if the estimated normal operation radioactive releases from SHNPP are reasonable; second, to examine the dose models used by the Staff and Applicants; third, to examine the estimates of resulting health effects. This appeared to be a more logical approach than the order of the subparts in the contention, because the health impacts are based on radiation dose and the dose, in turn, is dependent on the source term. If the radiation source term is small, then the radiation doses and health effects should also be small.

### *1. Source Term*

Joint Intervenors' Contention II(f) states, in part, that "certain radionuclides are ignored" in the source term. Neither the Staff nor the Applicants deny that some radionuclides have been left out of the source term. However, the omitted radionuclides would contribute less than 1% to the source term and consequently would not contribute significantly to the dose (Mauro at 11). Staff affiant Branagan also states that the source term as developed by the Staff includes all significant dose-contributing radionuclides (aff. Branagan at 23). The source term was developed in accordance with NUREG-0017, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors (PWR-GALE Code)" (1976).

Joint Intervenor's Contention II(b) states, in part, that "the health and genetic effects of alpha, beta, and neutron radiation on DNA, cell membranes and enzyme activity" have been underestimated. However, there are no alpha or neutron emitters in the normal-operation liquid or gaseous source terms themselves (aff. Branagan at 23). Many of the discharged radionuclides are beta-emitters; these will be considered in the dose and health effects evaluations that follow.

The only alpha-emitting radionuclide that can be expected from the effluents from the Harris plant (Table D-4 in the DES) is from the *decay* of Np-239 (half-life 2.35 days). The product of the decay is Pu-239 (half-life 24,400 years), which is an alpha-emitter. However, the conversion of the expected Np-239 release of $2 \times 10^{-5}$ curies per year into Pu-239 would result in the formation of $5 \times 10^{-12}$ curies per year of Pu-239, five trillionths of a curie, which would contribute insignificantly to the dose estimate.

In summary, the submissions of the Applicants and NRC Staff demonstrate that all significant radionuclides have been included in the source term for normal operation of the Harris facility and that the only alpha radiation from the source term would arise from Pu-239 at insignificant levels. The Intervenors' opposition papers do not controvert those showings and, therefore, summary disposition as to those portions of the contention pertaining to the source term (subparagraphs (b) and (f)) is granted.

## 2. *Mathematical Dose Modeling*

### (a) *Joint Intervenors' Contention II(f)*

Joint Intervenors' Contention II(f) states, in part, that "less reactive rather than more reactive forms of radionuclides are used in the computation of the radionuclides" in the environment. As pointed out by NRC Staff affiant Branagan, the only radionuclides that are specifically identified by the Intervenors are isotopes of plutonium as found in the reference to Eddleman 37(10) on page 113 of "Supplement to Petition to Intervene by Wells Eddleman, *pro se*," dated May 14, 1982. The Staff's estimate of the materials that might be released from the Harris plant during normal operation are presented in Table D-4 of the FES. Isotopes of plutonium are not listed in Table D-4.

The Intervenors in their response do not question the Staff estimate that insignificant amounts of plutonium will be released during normal operation of the Harris plant. The Board finds no issue of material fact and, therefore, grants summary disposition on this subpart of the contention.

Joint Intervenors' Contention II(e) states, in part, that the Staff's and Applicants' radionuclide concentration models are inadequate because they exclude radionuclides attached to fly ash from coal plants. Both Staff and Applicants acknowledge that fly ash has not been considered in dose modeling and both argue that the inhalation dose would be decreased in the case of radionuclides that might become attached to fly ash, thus forming a larger particle. They state that an effective increase in particle size would be expected to lower the deep-lung deposition and thus the dose (aff. Branagan at 21; aff. Whipple at 10). In addition, affiant Whipple states that fly ash particles tend to be highly insoluble and that the attachment of radioactive gases and soluble radioactive materials to fly ash would make them less available for transport along food pathways (Whipple at 11).

The Board feels that the affiants' statements "miss the mark" with regard to the point raised in this part of the contention. Both state conclusions without supporting analysis. Whipple's comments about transport along food pathways does not have any obvious relationship to the question of dose to the lungs from deposited particles. Clearly, if radionuclides were only associated with large particles, Staff's point could be significant. However, Staff makes no presentation of evidence that radionuclides will not become associated with particles with sizes of 0.5 microns or smaller. If that were the case, aggregation of such particles with fly ash that is also submicron would not produce significant retardation of the tendency to be deposited in the deep lung.

Joint Intervenors state at length the basis for this part of the contention (*see* Joint Intervenors' November 22, 1983 Response to NRC Staff at 2). They postulate that radionuclides as gaseous atoms may become associated with fly ash particles. Comparison of their statements with those of Staff and Applicants leads the Board to the conclusion that there is a material issue of fact to be litigated. Summary disposition for this part of the contention is, therefore, denied.

*(b) Joint Intervenors' Contention II(b)*

This portion of the contention states that:

> Insufficient consideration has been given to the greater radiation effects resulting from internal emitters due to incorrect modeling of internal absorption of radionuclides, and underestimation of the health and genetic effects of alpha, beta and neutron radiation on DNA, cell membranes and enzyme activity. (Reference: Sources cited in Eddleman 37F.)

Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (1977) describes the modeling of the doses from internal emitters from both inhalation and ingestion pathways (liquid and other foods). Staff affiant Branagan describes the primary features of the mathematical modeling and dose conversion factors (aff. Branagan at 7-11). He further describes the contract reviews that have examined 200 references in the scientific literature to evaluate the reliability of the input parameters and the variability in the dose estimates (aff. Branagan at 12 and 13). Branagan concludes that "exposures to offsite individuals in the vicinity of the Shearon Harris Nuclear Power Plant are estimated to be below the annual dose design objectives in Appendix I of 10 C.F.R. Part 50, and about two orders of magnitude or more below the public health and safety limits which can be derived from 10 C.F.R. Part 20" (aff. Branagan at 15).

Joint Intervenors have not filed any affidavits on this contention and we have before us only the responses written by Mr. Eddleman for the Joint Intervenors. Mr. Eddleman's response to the NRC Staff response to the summary disposition motion does not challenge or even mention the Staff position on dose modeling. With regard to Applicants' motion and affidavits, Mr. Eddleman also prepared the Joint Intervenors' response. His only statement with regard to dose modeling is "NRC models do use less reactive forms of radionuclides in figuring transfer factors." Such a broad statement which does not specify the radionuclides of concern to him or even refer to supporting scientific research is not sufficient to raise a genuine issue of fact. The Board grants summary disposition on this part of the contention.

As previously discussed, the Board accepts the source terms developed by the NRC Staff as appropriate. Insignificant quantities of radionuclides that emit alpha or neutron radiation are expected to be released from the Harris plant during normal operation. The remaining issue in this subpart of the contention is the health and genetic effects of beta radiation.

Applicants' affiant, Dr. Fabrikant, discusses this subpart of the contention at 58-61. The Board notes that Dr. Fabrikant was a member of the BEIR Committees and he is well qualified to describe the Committees' work. Dr. Fabrikant states that "the BEIR I and III reports thoroughly and correctly explain and apply current knowledge concerning radiation effects including effects on DNA, cell membrane and enzyme activity where available." Fabrikant states further that "effects on DNA are well studied in the scientific literature. It is the basis of all understanding of cell lethality and cell death, cell transformation and carcinogenesis, and cell (genetic) mutagenesis." With respect to this particular issue, Intervenors have not pointed to any particular omission in the BEIR Committee considerations of the effects of radiation on DNA. The Board is unable to identify an issue of fact that could be usefully litigated.

With respect to beta radiation effects on enzymes, the Intervenors have not identified any particular enzyme or enzyme system and Dr. Fabrikant does not discuss any particular enzyme system. Rather Dr. Fabrikant presents the view that damage to enzymes is undoubtedly involved in mechanisms of carcinogenesis or genetic effects. It appears to the Board that there is no argument that beta radiation affects enzymes and that these effects are part of the mechanisms for processes that are expressed as genetic effects and carcinogenesis. It is the Board's view that these effects are inherent in radiation effects on cells and whole organisms. We cannot find, in the pleadings of the Intervenors, allegations of any specific effect that has been neglected or that could be usefully litigated with specificity.

With regard to beta radiation effects on cell membranes, Dr. Fabrikant states that "after a review of the scientific literature, the BEIR III Committee determined not to accept the presentation of Sternglass that health effects were being underestimated because of postulated effects on cell membranes. (BEIR III report at 464-469)." The Board takes official notice of the referenced portion of the BEIR III Report. The Report states that:

> Ernest J. Sternglass appeared before the Committee to present a number of comments about the effects of low-level radiation on man. Part of Dr. Sternglass's pres-

> entation alleged that fallout from Chinese bomb-testing in 1976 led to an increased amount of radioactivity in milk in some areas of the United States. He concluded that there was an increase in infant mortality in the eastern-seaboard states from Delaware to New England shortly after these events — an increase that he ascribed to the radioactivity. Although Dr. Sternglass stated that his analysis was incomplete, the Committee received no further data on this subject. We have concluded that the alleged association did not fit the time course for radioisotope movement into the cow-milk food chain; nor was there clear evidence of a universally applicable change in infant-mortality rates. Thus, the Committee did not believe that the allegation was substantiated.
>
> Most of Dr. Sternglass's material was directed at evidence, chiefly from Dr. A. Petkau of Canada, indicating effects of various kinds of radiation at low doses and low dose rates on membranes similar to cell membranes. The Committee contacted Dr. Petkau, who kindly provided reprints of his work, as well as personal comments concerning it. The following material has been developed as a result of consideration of evidence provided by Dr. Sternglass, Dr. Petkau, and others.
>
> The experimentally demonstrated effects of ionizing radiation on cell membranes provide an alternative or conjunctive damage mechanism in addition to effects on DNA, which are generally accepted as the primary modes of damage in biologic systems. Radiation damage to cellular and intracellular membranes is manifested by alterations in permeability, which lead to altered distribution of various intracellular molecules and ions and disruption of membrane-associated biochemical processes. Although it is well recognized that membrane integrity is essential for normal cell function, there is inadequate basic understanding of membrane structure and function on which to base a detailed theory or radiation-induced damage mechanisms. (Footnote omitted.)

The BEIR III Committee then continues for several pages to review the literature on the observed effects of x-radiation or gamma photons primarily on *model* membrane systems. They note that this literature draws attention "to the potential significance of membrane-mediated damage in biologic systems." They conclude that the research they reviewed strongly suggests that membrane damage may be part of the mechanisms in carcinogenesis and that "thus there is a need for additional studies in this field."

The Board reads this section of the BEIR III Report as expressing the view that membrane damage, as a function of dose rate, needs continuing consideration but that "the available data relative to the effects of low-dose or low-dose-rate exposures on carcinogenesis in humans and experimental animals do not, in general, support the hypothesis of an increased probability of induction at low dose rates." The intervenors do not bring to our attention any study or group of studies that would directly challenge the views of the BEIR III Committee.

The Applicants' Fabrikant affidavit and the Board's reading of the BEIR III Report lead us to conclude that there is not an issue of material fact that could be litigated that might substantially alter the Staff's estimation of the health and genetic effects of beta radiation, and therefore this subpart of the contention is dismissed.

### *(c) Joint Intervenors' Contention II(e)*

This portion of the contention states that:

> The radionuclide concentration models used by Applicants and the NRC are inadequate because they underestimate or exclude the following means of concentrating radionuclides in the environment: rainout of radionuclides or hot spots; radionuclides absorbed in or attached to fly ash from coal plants which are in the air around the SHNPP site; and incomplete mixing and dispersion on radionuclides.

In response, both Staff and Applicants argue that incomplete mixing is accounted for in Regulatory Guide 1.11 modeling. ("Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors" (1977).) Applicants' witness Whipple agrees that at certain times and places, incomplete mixing, incomplete dispersion, or precipitation (wet deposition or rainout) will produce concentrations higher than the annual average concentration. He continues that Regulatory Guide 1.11 models account for those factors and arrive at conservative resultant plume concentrations. Further, wet deposition would be of concern only if the rainy season coincides with the local grazing season — not the case in North Carolina (aff. Whipple at 7 and 8).

Staff witness Spickler states that the averaging concepts embodied in the Staff models are appropriate for calculation of the dispersion parameter ($X/Q$) and that poor diffusion conditions with limited mixing are considered in the calculation of annual average $X/Q$ values. He continues that "hot spots" may occur for very short periods, but these are sufficiently random in time and space over an annual cycle and are reflected in the calculation of annual average $X/Q$ values through consideration of stable atmospheric conditions accompanied by low wind speeds (aff. Spickler at 3).

The Joint Intervenors have not filed any affidavit challenging the appropriateness of the spatial and temporal averaging on an annual basis used in the dose modeling derived from Regulatory Guides 1.109 and 1.11. Mr. Eddleman responded for Joint Intervenors and was critical of the Spickler affidavit. While the Board finds that Mr. Spickler's points would have been clearer with a little more detail, the Board agrees with him that periods of above-average doses are included in the mathematical modeling of annual doses by the use of site-specific information describing the frequency of stable atmospheric conditions accompanied by low wind speeds. We find no issue of fact and, therefore, grant summary disposition of this part (II(e)) of the contention.

*(d) The Heidelberg Report*

Intervenors cite NRC Translation 520, a translation of the so-called "Heidelberg Report," to support allegations in various parts of the contention that calculation parameters and dose conversion factors — and thus the radiological dose — used by the Staff and Applicants are incorrect. The Applicants present a Summary of Critiques which casts grave doubts on the "Heidelberg Report" (aff. Mauro, Exhibit B). This summary presents a brief review of NUREG-0668 — the NRC Staff analysis of NRC 520 and critiques of NRC 520 by five other sources.

In NUREG-0668, "Staff Review of 'Radioecological Assessment of the Wyhl Nuclear Power Plant' " (1980), the NRC Staff found that the liquid and gaseous source terms used in the Heidelberg Report were many times greater than the average source terms from operating plants in the U.S. and therefore do not reflect U.S. operating experience (aff. Mauro, Exhibit B).

The NRC Staff found that the methodology by which the atmospheric dispersion factors in the report were derived was severely flawed and combinations of wind speed, direction and stability class are used which have no meaning in reality. The Staff believes that the peak *X/Q* used in the Heidelberg Report may be high by a factor of 10 or more. *Id.*

The NRC Staff found that the soil-to-plant concentration factors (Biv) were not supported by the literature cited. Specifically, the Cs and Sr Biv values were selected at the high end or well beyond the high end of the experimental data. *Id.* Also, the dose conversion factors for Cs-137 and Sr-90 are much higher than those used by the NRC and are not supported by experimental data. The Staff's conversion factors are based on International Council on Radiation Protection ("ICRP") guidelines. *Id.*

The NRC Staff determined that if the Heidelberg models were valid, then high, easily detectable levels of I-131 and Cs-137 would be found in the vicinity of operating reactors, when in fact they are not. *Id.*

The University of Heidelberg neither prepared nor sponsored the Report and in fact sought to restrain the authors from using the University's name. A representative of the German government nuclear power plant licensing agency was extremely critical of the Report, referring to it as "less a serious scientific report but rather a public relations paper of opponents against nuclear energy. All European institutions, which dealt with the Report, came to similar statements." *Id.* In addition, the German Society for Reactor Safety was very critical of meteorological modeling used in the Heidelberg Report. They believe the Report's long-term diffusion factor and the deposition velocity of airborne particles are high by factors of 3 and 4, respectively.

Intervenors do not marshal any respectable scientific support for the so-called "Heidelberg Report." These faults in the "Heidelberg Report" lead the Board to give it little or no weight and we do not find any basis for support of the various parts of the contention.

### 3. *Health Effects*

#### (a) *Joint Intervenors' Contention II(a)*

Joint Intervenors' Contention II(a) states that:

> The work of Mancuso, Stewart, Kneale, Gofman and Morgan establish that the BEIR-III Report (1980 report of the National Academy of Sciences' Committee on the Biological Effects of Ionizing Radiation, entitled "The Effects on Population of Exposure to Low-levels of Ionizing Radiation") (1) incorrectly understood the latency periods for cancer; (2) considered only expressed dominant genetic defects; and (3) failed to use a supralinear response rather than a threshold or linear-or-less model to determine low-level radiation effects.

The Board has considered the source terms and dose modeling developed by the Staff and Applicants as a basis for assessing the extent of possible health effects. We believe that the Staff's source term is reliable and that the models are sufficient to describe the radiation exposures and doses to humans that would occur in the vicinity of Shearon Harris. From these source terms, dose models and the FES, the Board notes that the resultant doses to humans from Shearon Harris liquid and gaseous effluents will be within 10 C.F.R. Part 50, Appendix I, dose design requirements.

Subpart II(a) of the contention is focused on the report of the BEIR III Committee. We recognize the substantial qualifications and experience of Dr. Fabrikant, the Applicants' principal affiant. We take those factors into account in evaluating Dr. Fabrikant's presentation of estimated health effects and his discussion of the work of BEIR I and III Committees.

#### (i) Latent Periods for Cancers

To state as the Intervenors do that the BEIR III Committee did not understand cancer latent periods seems highly unlikely to this Board. This is especially true considering the expertise represented on the

BEIR Committee in the fields of radiological health effects. Dr. Fabrikant explains the BEIR III approach to cancer latent periods (Fabrikant at 41-44). He discusses the Committee's understanding of the term and the basis for the Committee's estimation of the latent periods for various cancers. He states that "except where the data clearly showed a particular disease had a maximum latent period . . . the committee assumed that the cancer risks for a particular exposure continue throughout an individual's lifetime."

The Joint Intervenors' response was prepared by Mr. Eddleman. So far as we are aware, he has no expert qualifications in radiation health effects. Mr. Eddleman cites a paper by Rosalie Bertell as showing "that the BEIR III limited the latency period considered, looking at deaths 11-30 years after exposure, only." The article by Bertell references Table V-14, at 198 of the BEIR III Report, and that table is clearly identified as "estimated excess cancer (excluding leukemia and bone cancer) per million persons per year per rad, 11-30 yrs. after exposure, by site, sex and age at exposure." However, other tables in the BEIR III Report do consider lifetime exposure; for example, Table V-25, at 212.

The Board finds that the selective citation (by reference) to *one* table in the BEIR III Report does not support the allegation that the BEIR III Committee did not consider lifetime risk of cancer. This subpart of the contention is dismissed.

(ii) Genetic Defects

The Fabrikant affidavit discusses the estimates of genetic effects in the BEIR III Report and the estimates contained in the recent book by Dr. Gofman (Fabrikant at 47-50). Dr. Fabrikant cites an excerpt from a *draft* Nuclear Regulatory Commission-Harvard University Report on Revision of the Radiological Health Effects Model (1982-1983) that is in extensive disagreement with the analyses of genetic effects by Dr. Gofman. We do not consider excerpts from draft reports, at least standing alone, to be a substantive basis for a motion for summary disposition. Apart from the excerpt, Dr. Fabrikant's criticisms of Dr. Gofman's work are too general to carry any weight.

The Board's view is that there are genuine issues of fact in this genetic defects part of the contention as to Dr. Gofman's recent work. *See* Radiation and Human Health, Chapt. 22. Summary disposition of this part of the contention is denied. As discussed above, we intend to call Dr. Gofman to present and defend his views on genetic effects.

(iii) Supralinear Response

This subpart of the contention criticizes the Staff and Applicants for using a threshold or linear-or-less dose-response model to estimate radiation effects rather than a supralinear dose-response model. If the supralinear dose response is used, the calculated health effects from exposure to ionizing radiation are greater than those that result from using the dose-response models of the Staff and Applicants. Intervenors refer to reports of Mancuso, Stewart and Kneale, Gofman, and Morgan to support their claim that the supralinear dose-response model should be used.

The Staff and Applicants (Branagan and Fabrikant, respectively) rely on reports of the BEIR I and III Committees as well as ICRP (1977), NCRP (1975), and UNSCEAR (1982) for their dose-response models — namely the linear (L) and linear-quadratic (LQ-L) models. Staff's computations of somatic and genetic risks were based on the BEIR I linear nonthreshold dose-response model which yields higher estimates than the BEIR III Report model for exposure to low-LET radiation (DES at 5-28). The Staff is aware of the higher Mancuso-Stewart-Kneale risk estimator (about 2 times that of the upper range of the Staff estimator) but notes that the resulting estimates do not change the Staff's conclusion that "these risks are very small in comparison to natural cancer incidence from causes other than the operation of Shearon Harris" (DES at 5-33). Dr. Fabrikant describes a number of dose-incidence curves for cancer induction in irradiated populations. He states that the supralinear dose-response curve is not used by recognized organizations for risk estimation for low-dose, low-LET radiation exposure, since there is no experimental evidence or epidemiological evidence that this dose-response relationship is appropriate for risk estimation (aff. Fabrikant at 23). As discussed by Dr. Fabrikant, the 1972 BEIR Committee considered it scientifically appropriate to adopt a no-threshold linear hypothesis of dose response to estimate the cancer risk at very low-level (low-LET, whole-body) radiation exposure. *Id.* at 24. The 1980 BEIR III Committee considered the linear-quadratic, no-threshold dose response as the preferred model. They considered that the linear dose-response model was unduly conservative and would lead to overestimation of risks. This change from the 1972 BEIR Report was based upon the available epidemiological surveys, experimental and cell culture evidence and current microdosimetric theory. Not a single member of the twenty-three experts on the BEIR III Committee advocated supralinearity. *Id.* at 25. Fabrikant continues by describing BEIR III, NCRP and ICRP critiques of the work of the previously mentioned authors. These critiques concluded that the experimental and epidemiological data simply do not support the use of a supralinear dose-response model for radiation carcinogenesis following low-LET exposure. Further, no evaluations in the peer-reviewed literature of any recent reports on epidemiological studies suggest in any manner that the linear hypothesis is not conservative. *Id.*

Accordingly, except as supralinearity may be related to Dr. Gofman's estimates, we grant summary disposition as to Joint Contention II(a)(3).

(iv) Dr. Gofman's Cancer Risk Estimates

Contention II(a) alleges in substance that the work of Dr. Gofman (and others) establishes that the BEIR III (and, by implication, BEIR I) estimates of cancer risk are "seriously underestimated" for the reasons discussed in the preceding three sections. Contention II(c) (discussed below) also focuses specifically on Gofman's work, contending that the Staff examines effects over an arbitrarily short period of time. It is not clear to us that the exact points cited in these parts of Contention II are the exclusive reasons that Gofman's estimates diverge from the BEIR estimates.

This much is clear, however. There are large differences between the Gofman and BEIR estimates, as shown by the table on p. 444, above. It is considerably less clear just how these wide differences arose, or even how wide they are. In his affidavit, Fabrikant states that Gofman's "worst case" is a 40% increase in cancers per rad over the estimates in BEIR I. Fabrikant references page 218 in Gofman's book for this statement. Affidavit at 77. We find no reference or comparison with BEIR I on page 218. Rather, the material appears to relate to a calculation of the peak percent increase per rad for some data on the Hanford workers. The Intervenors cite the table from Gofman's book (*see* p. 444, above) which is a comparison of his estimates with those in a draft (1979) of the BEIR III Report. This comparison shows a difference between Gofman's estimates and those of BEIR III of 1000 to 2500%, rather than the 40% value indicated by Fabrikant for BEIR I. For his part, Dr. Gofman has this to say:

> There is no mystery at all about how this author arrived at his estimates; all the evidence, every assumption, and each step of every calculation are presented in this book. Unfortunately, the reader of the BEIR III Report will have extreme difficulty ascertaining how BEIR members did their analysis, because the presentation of that analysis is simply inadequate.

We should add that we have not assimilated all of Dr. Gofman's long book and do not have a full understanding of how he arrived at his estimates. That understanding can be developed at the hearing.

In summary, we conclude that Dr. Gofman's cancer risk estimates are fairly encompassed within Contention II(a) and (c) and summary disposition with respect to those parts of Contention II is denied as they relate to the correctness of the Gofman estimates. We are granting summary disposition with respect to parts (1) and (3) of Contention II(a) relating to latency periods and supralinearity, subject to the conditions (1) that Dr. Gofman may address those points if they are necessary or helpful in explaining how he derived his estimates and (2) that any opposing party may seek to rebut his testimony.

(v) Time Period for Estimating Health Effects

Contention II(c) reads as follows:

> (c) The work of Gofman and Caldicott shows that the NRC has erroneously estimated the health effects of low-level radiation by examining effects over an arbitrarily short period of time compared to the length of time the radionuclides actually will be causing health and genetic damage.

The annual dose commitment calculated by Staff is the total dose that would be received by an individual over a 50-year period following the intake of radioactivity for 1 year under the conditions existing 20 years after the station begins operation. This formulation produces an "annual dose" calculation based on operations in an "average" year — *e.g.*, 0.008 cancer death per year. The contention argues that the Staff's estimates should extend over the time the radionuclides actually will be causing health and genetic damage. In their papers, the Intervenors contend that it should extend to the entire life of all nuclides, or at least to some 11 million years.

Staff's methodology, as the Board understands it, consists of calculating the dose commitment and from this to arrive at a determination of potential health effects through the use of appropriate risk estimators. The resulting health effects are then compared with the potential health effects expected as a result of natural background radiation. This method provides a perspective from which reasonable judgments of incremental risk can be made, and, up to that point, we agree with the Staff's approach.

We question, however, whether the Staff should confine itself, as it has done in this case, to computations of annual doses and effects. In the first place, although this is not the principal thrust of the contention, it seems to us that the Staff's impact statement, whose purpose is to make a clear and full disclosure of risks associated with the facility, should disclose the total risk represented by the life of the plant. If this were done, the annual risk figures now in the impact statement would have to be multiplied by 30 to 40 times. More fundamentally, the Staff's annual risk approach does not appear to take into account the incremental impact on people who live near the facility for many years. For example, the risk to such a person over 20 years would presumably be many times larger than the risk to a 1-year resident.

On the other hand, we do not believe that the Intervenors' 11-million-years proposal has any merit. After all, the facility will be decommissioned after 40 years or less and its emissions will virtually cease. Furthermore, the very long-lived radionuclides are, generally speaking, less hazardous. Beyond that, projections of health effects into the millions of years are purely speculative; they have been rejected largely on that basis. *See Philadelphia Electric Co.* (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-701, 16 NRC 1517, 1526 (1982).

In light of the foregoing, we deny summary disposition on Contention II(c). We have cited two instances where the Staff may be required to justify its present approach. There may be others. We have also indicated that we will bar wholly speculative efforts to predict the effects of routine releases millions of years into the future.

## B. Eddleman Contention 37B

This contention reads as follows:

> The work of I.D.J. Bross (Ph.D.), Rosalie Bertell (Ph.D.) and others shows that radiation exposure increases the risk not only of cancer but a host of other diseases, allergies, and causes of death including heart disease, heart attack, and others. The estimates of the numbers of such victims made by the preceding workers *et al.* are more accurate than the estimates (if any) used by Applicants or NRC Staff or BEIR committee reports.

Applicants, in proposed Material Facts 15 and 28, state that diseases other than cancer and genetic defects are not produced by routine releases from Shearon Harris. Fact 15 states that routine releases will not increase the risk of any other diseases and Fact 28 states that all diseases other than cancer and genetic effects have threshold doses beyond those imposed by routine releases. Dr. Fabrikant discusses two classes of disease associated with radiation — those having no threshold of dose response and those having dose-response thresholds (aff. Fabrikant at 69). In the first of these, disease conditions potentially arise when effects take place in one or a few cells and appear in the population as tumors or hereditary effects. Further, their incidence is related to dose — the incidence increases with increasing dose. In the other class are those diseases which potentially arise when the effects take place in many cells simultaneously and appear as tissue or organ damage in individuals in an irradiated population. According to Dr. Fabrikant, these diseases will not appear unless the dose is above a particular threshold. He uses cataract induction (500 rads), heart disease (4,000 rads), and muscle atrophy (10,000 rads), as examples. In a similar vein, BEIR III states that "[f]or doses of less than approximately 300 rads of low-LET radiation, the principal mechanism of life-shortening is the induction of neoplastic diseases." Doses greater than 300 rads are orders of magnitude greater than routine release doses from Shearon Harris.

The NRC Staff affiant Branagan also supports the position that diseases other than cancer and genetic defects cannot be caused by the levels of routine releases involved here. He quotes UNSCEAR 1982, Annex J, "Non-Stochastic Effects of Irradiation" (aff. Branagan at 25) as recognizing that some symptoms (such as degeneration of heart muscle and skin reddening) have been associated with exposures to ionizing radiation, but that such doses are more than a thousand times greater than doses projected for normal operations at Shearon Harris.

The contention as stated is extremely broad, referring to an unspecified "host of other diseases." Beyond that, there is substantial and uncontradicted evidence before us that no diseases other than cancer and genetic defects can be caused by routine releases from Shearon Harris. We have nothing in opposition to that evidence except references to publications of Bross and Bertell. Summary disposition of Contention 37B is granted.

### C. Eddleman Contention 8F(1)

This contention states that:

> Appendix C of the FES underestimates the environmental impact of the effluents in Table S-3 for the following reasons: (1) health effects of the coal particulates 1,154 MT per year, are not analyzed nor given sufficient weight.

The FES for the Harris facility does not contain a specific analysis of the health effects of coal particulates emitted during the uranium fuel cycle. However, Staff maintains that the level and environmental impacts of coal particulate emissions have been considered on a generic basis. The Staff determined that such environmental impacts need not receive a separate analysis due to the insignificant level of such emissions. The

FES considers only those environmental impacts which reasonably appear to be significant. (Affidavit of Charles W. Billups in Support of NRC Staff's Motion for Summary Disposition at ¶¶ 4, 5 and 28; hereinafter Billups Affidavit). (*See also*, Staff Motion at 7 n.1).

The particulate emissions in question are set forth in Table S-3 as being 1,154 mt/year. This is approximately 0.02% of the national annual release of coal particulates, or a contributory effect of about one part in 5000.[1] (FES, App. C at C-4). It appears to the Board that it is not unreasonable for the Staff, in its judgment, to consider that the contribution of the S-3 amount of particulates to the total national burden is insignificant.[2]

That the level of particulates is insignificant in the national context does not, however, rule out the possibility of localized health effects. The NRC Staff affidavit takes the position that the health effects of 1,154 tons of particulates were evaluated on a generic basis in WASH-1248 (Environmental Survey of the Uranium Fuel Cycle) and in the FES for the individual fuel enrichment plants. As the Board reads WASH-1248, there is no explicit evaluation of the health effects of the postulated particulate emission. The NRC Staff submitted as Exhibit A to its affidavit a portion of the FES for the Portsmouth plant. The Board does not find an explicit evaluation of the health effects of the postulated particulate emission in this document either. In sum, on the basis of the Staff's papers, we are unable to verify that the health effects have been evaluated on a generic basis.

Applicants' motion is supported by an affidavit from Dr. Hamilton. He performs an apparently conservative health effects analysis of the postulated 1,154 tons of particulates and concludes that they would cause about 0.1 death per year if discharged from one of the coal plants providing power for the gaseous diffusion process. While this analysis is helpful, it cannot support summary disposition of the contention because the 0.1 death estimate appears to us to be possibly significant in the NEPA evaluation. We do not mean to imply that such an estimate

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[1] The Board notes that, given the need for the power to be generated, the utilization of the Harris plant instead of coal-fired units could reduce the production of particulates by as much as a factor of 20, although apparently no credit for this reduction is taken in the FES. If it were to be shown at hearing that the Harris plant will displace coal-fired units and that this will result in a substantial net reduction in particulate emissions, that presumably would dispose of this contention.

[2] Both the Staff and Applicants argue that the actual quantity of particulates would be much less than 1,154 mt/year because EPA particulate emission standards have become much more stringent since that figure was set in the rule. We agree with Mr. Eddleman that these arguments are an impermissible attack on the rule. 10 C.F.R. § 51.23(e) plainly states that the impacts of fuel cycle particulates "shall be evaluated on the basis of impact values set forth in Table S-3." If the Staff and Applicants think the particulate value is too high, they should petition for waiver or amendment of the rule. Otherwise, this Board must assume that the 1,154 mt/year value is correct.

might be enough to tip the balance by itself. But on the other hand we cannot say that it might not make some significant contribution to the analysis. Short of a fuller exploration of this matter at a hearing, we cannot determine that the 0.1 death estimate is so insignificant that it does not even have to be mentioned in the FES.

The motions for summary disposition are denied.

## D. Eddleman Contention 8F(2)

Eddleman Contention 8F(2) states:

> The FES assessment of the health effects of the radiological effluents specified in Table S-3 is inadequate in that (i) effects are considered for too short a time period; (ii) food chain concentration analyses are wrong; (iii) radionuclide concentration values are not conservative in view of NRC Translation 520; and (iv) radiation doses from internal and external emitters are underestimated.

### *1. Introduction*

Table S-3, as referenced in the contention, lists the radiological effluents released in the uranium fuel cycle. The FES assessment of potential health effects from the effluents released from the uranium fuel cycle is found in Section 5.10 and in Appendix C, and that assessment may be summarized as follows:

> The NRC Staff has determined that the environmental impact of this facility on the U.S. population from radioactive gaseous and liquid releases (including radon and technetium) due to the uranium fuel cycle is very small when compared with the impact of natural background radiation.

FES, Section 5.10. Contention 8F(2) challenges this conclusion in two basic respects. In 8F(2)(i), Mr. Eddleman contends health effects were assessed for too short a period. Contention 8F(2)(ii-iv) alleges that the NRC Staff's dose calculation supporting its health effects estimate underestimates the dose due to the use of improper concentration and dose values for the radionuclides listed in Table S-3. The contention in this latter respect constitutes a challenge to particular aspects of the NRC Staff's dose-modeling techniques. Applicants contend, briefly, that the NRC Staff assessment for uranium fuel cycle health effects demonstrably covers an adequate period of time, and further, that the underlying model does not underestimate dose.

*2. Duration of Health Effects Estimates*

In the FES, effects are considered for the 100-year period and the 1000-year period associated with each year of plant operation. These periods are those in which the fuel-cycle to natural-background ratios are the highest that might occur, as the dose attributable to the uranium fuel cycle will decrease over time while the background radiation will not. Affidavit of John J. Mauro and David Michlewicz in Support of Applicants' Motion for Summary Disposition of Intervenor Wells Eddleman's Contention 8F(2) (hereinafter Joint Affidavit), ¶ 12.

In Mr. Eddleman's response to Applicants' Motion, he avers that some of the effluents have half-lives of up to 4.5 billion years, and that the time period should therefore be extended to cover such substances. The Board does not find this argument persuasive. As we noted above in rejecting summary disposition of Joint Contention II(c) (*see* p. 458, *supra*), estimation of health effects for the time periods urged by Mr. Eddleman would be a speculative exercise. Furthermore, the concern we expressed there about the possibility of aggregate doses to people living near the facility does not apply to fuel cycle effluents, which are dispersed over many different geographical areas. In light of these considerations, and the Staff's determinations concerning the maximum ratio of health effects from the fuel cycle to those from background radiation, additional calculations are unnecessary. We therefore find that Contention 8F(2)(i) raises no issue of material fact.

*3. Calculation of Doses*

Contention 8F(2)(ii-iv) asserts that the dose calculation in the FES for the uranium fuel cycle is underestimated because the Staff used inappropriately low radionuclide concentration values in the food-chain pathway and for internal and external emitters.

Applicants state that Dr. Mauro and Mr. Michlewicz have reviewed the methodology used by the Staff in calculating health effects from the uranium fuel cycle and the most recent literature relevant to transfer factors (food-chain pathway) and dose conversion factors (internal and external emitters). The modeling procedure used in the FES is set forth in the NRC's "Final Generic Environmental Statement on the Use of Recycled Plutonium in Mixed-Oxide Fuel in Light-Water Cooled Reactors" (NUREG-0002) (August 1976). Pathways considered in the procedure include (1) external exposure to airborne radioactivity; (2) inhalation of radioactivity; (3) external exposure to radioactivity deposited on the ground; and (4) ingestion of foodstuffs containing radionuclides from terrestrial and aquatic food pathways. The models and parameters in NUREG-0002 were found to be consistent with standard methodologies widely used in the nuclear industry, are reasonable and do not lead to dose underestimate. Joint Affidavit, ¶ 17.

Dr. Mauro and Mr. Michlewicz also compared the parameters used by the NRC Staff in performing the dose calculations found in Appendix C to the DES with the most recent literature describing ongoing research and data. They found that many of the parameters used by the NRC Staff remain unchanged while some have increased and others have decreased. *Id.*, ¶¶ 18, 19. Overall, the parameters used and the dose and health effects calculated by the NRC Staff in the DES are reasonable and within the range of values observed or calculated in the scientific literature. *Id.*, ¶ 19.

To confirm the calculations performed by the NRC Staff, Dr. Mauro and Mr. Michlewicz independently calculated the dose and health effects using the NRC Staff estimates of radionuclide releases from the uranium fuel cycle. A description of the calculational methodology is set forth in Attachment 5 to the Joint Affidavit. The NRC Staff estimated a population dose of approximately 600 man-rem (not including radon) over a 100-year period due to the radionuclide releases required to support 1 year of operation of the referenced light water reactor. Joint Affidavit, ¶ 20. Dr. Mauro and Mr. Michlewicz calculated a total 100-year dose commitment of 620 man-rem. Given the statistical uncertainties in the calculational parameters, these numbers are essentially identical. *Id.* Dr. Mauro and Mr. Michlewicz also independently calculated the 1000-year population dose and health effects. Their independent calculation resulted in less than one cancer death per 1000 years and is roughly comparable to the NRC Staff estimate of 0.13 cancer death. *Id.*, ¶ 21.

Mr. Eddleman principally bases his allegations on the so-called "Heidelberg Report," otherwise referred to as NRC Translation 520. The Report has been thoroughly discredited by the scientific community and NRC Staff analysis. Motion at 12-16 and discussion at pp. 452-53, above. The Board chooses not to belabor this point, but finds that the report and Mr. Eddleman's challenge to the FES modeling practices based thereon can be given no weight and cannot raise any issue of material fact in this proceeding.

The Board finds that, based upon the facts presented to it, that Contention 8F(2) presents no issue of material fact to be litigated in the proceeding. Applicants' Motion for Summary Disposition is therefore granted.

## III. SCHEDULE

Dr. Gofman advises us that he might be willing to serve as a Board witness, but that he would not be available for a hearing until late Spring. The Applicants advise us that Dr. Fabrikant is unavailable in May. Therefore the Board is setting a tentative evidentiary hearing beginning date on the environmental issues for June 5, 1984. We are also scheduling a prehearing conference for May 1, 1984. Any party should advise the Board by February 10, 1984 if these dates are not acceptable, and alternative dates should be proposed.

## IV. RESUMPTION OF DISCOVERY

On January 3, 1984, the Applicants filed a motion to resume discovery on Joint Contentions I and VII, subject to certain conditions. The Staff filed a response in support of the motion. The Joint Intervenors filed an untimely opposition to the motion, which had been served by express mail on counsel for one of the Joint Intervenors on January 3, 1984. Any response was due on January 16, 1984. 10 C.F.R. § 2.710. The Joint Intervenors' response was filed on January 24, 1984, eight days late. Although the Joint Intervenors ask that their response "be deemed timely filed" there is no showing whatever of good cause for the delay.

All parties to this case, including the Joint Intervenors, are expected to make timely filings or to seek extensions in a timely manner. Failing that, a party must make a showing of good cause why a late filing should be considered. The Joint Intervenors' response in opposition to the Applicants' motion of January 3, 1984 is rejected as untimely. The Applicants' motion is granted.

THE ATOMIC SAFETY AND LICENSING BOARD

James L. Kelley, Chairman
ADMINISTRATIVE JUDGE

Dr. James H. Carpenter
ADMINISTRATIVE JUDGE

Glenn O. Bright
ADMINISTRATIVE JUDGE

Bethesda, Maryland
January 27, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**Before Administrative Judges:**

**Peter B. Bloch,** Chairman
**Dr. Kenneth A. McCollom**
**Dr. Walter H. Jordan**

**In the Matter of** **Docket Nos. 50-445**
**50-446**
**(Application for Operating License)**

**TEXAS UTILITIES GENERATING COMPANY,** ***et al.***
**(Comanche Peak Steam Electric Station, Units 1 and 2)**

**January 30, 1984**

# MEMORANDUM
## (Records Retrieval)

This memorandum discloses the Board's thinking about the adequacy of the record concerning the computerization of certain deficiency records for construction and the adequacy of the system for retrieving and utilizing these deficiency records. The purpose of this disclosure is to assist the parties in focusing on matters the Board considers important when they file Proposed Findings or submit additional relevant proof. Since the findings in this memorandum are preliminary, tentative and nonbinding they may not be referenced as authority for filings and they are not subject to motions for reconsideration.

## I. COMPUTERIZATION

During the June 16, 1983 hearing session, Mr. Stuart Treby, Staff Counsel, conducted a cross-examination of Mr. Ronald Tolson, designed to ascertain whether Applicants' use of punchlists (Deficiency Listings), attached to Inspection Reports (IRs) constituted compliance with Part 50, Appendix B, Criterion XVI. The cross-examination begins at Tr. 8537. At Tr. 8537, line 13, Judge Bloch asked how the deficiency listings would be used to track a separation problem that affected two adjoining electrical conduits. Mr. Tolson answered that the computer "would automatically show it against both conduits."

Following the hearing, the Board took a site visit to the Comanche Peak plant. Our first stop was the computer center, where we asked the operator to pull onto the screen an unresolved nonconformance that had been detected in a component by an inspection report. When the operator was unable to do that, we were directed to a second location, where the operator of the second computer also could not do it.

Upon arriving in Washington after the hearing, the Chairman requested an explanation from Applicants. The first response, a letter of September 14, 1983, was that the report the Board had sought at the site could not be obtained from the particular system but could have been obtained from other systems. Because the Board was not fully satisfied by this answer, an affidavit was requested. In the responsive affidavit, filed on October 11, 1983, Mr. Tolson stated that "most site groups" use the computer system for tracking open IRs that require action by these groups. He also stated, however, that "prior to mid-September, 1983" . . . "*some* open items were entered into and tracked with the computer system." Affidavit at 2 [emphasis added].

We note that in his initial testimony Mr. Tolson relied on the computer system as part of his explanation of how IRs were used by Applicants. However, the October 11, 1983 affidavit produced the clarification that only some deficiency listings were available in the computer at the time of Mr. Tolson's initial testimony. Hence, we conclude that Mr. Tolson's initial explanation was incomplete.

## II. RECORDS

Our concerns go beyond the completeness of Mr. Tolson's testimony, however. Applicants were installing a computer system for IRs for some purpose, although the purpose does not seem to appear in our record. Presumably, a computer tracking system was considered to be helpful in handling the complex mass of documents being generated by construction. Considering the lack of success in using the computer system for that purpose,[1] the Board is concerned about whether the manual system is adequate.

Our concern is heightened by Staff documents and testimony. Mr. Compton testified that there is no trending done on punchlists.[2] Mr. Ford stated that IRs "are not dispositioned."[3] Mr. Ford stated that hold tags might be needed for IRs but that he did not believe they were used.[4] Mr. Beach stated that he was concerned about whether IRs are properly dispositioned.[5] Mr. Beach also stated that engineering approvals of "use as is" for an IR item would not be stated in writing.[6] Although the Staff subsequently seemed satisfied with Applicants' explanation about these matters, we do not think our record satisfactorily reveals how the Staff arrived at that position or whether it did an empirical check on the adequacy of the Applicants' answers.

The need for a further empirical explanation is heightened by the Staff's report of the final walkdown inspection of the Fuel Building, filed October 12, 1983 (dated July 27, 1983) at 17-18, 19, finding that there was no procedural control or historical record for punchlists. Mr. Taylor, the Resident Inspector, heightened our concern by testifying that he does not know of any study of the reliability with which manual records are being used.[7] Furthermore, it was the subjective and undocumented view of the Resident Inspector that the error rate in Brown & Root work, at Comanche Peak, is double the error rate on a typical nuclear plant.[8]

We note that our concern about the adequacy with which manual QA records are being used extends also to CMCs[9] and other design-deficiency documents that have not been computerized.

Assuming that nonconformances are carefully recorded, difficulty in retrieving the documents could lead to an unacceptable level of uncorrected deficiencies. Indeed, given the massive size of a nuclear plant, the lack of availability of an adequate report on open deficiencies also

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[1] Inspection Report 50-445/83-24; 50-446/83-15, ff. Tr. 8917, at 12 concludes that computer-based data on deficiencies were inadequate to conduct as-built inspections as of April 4, 1983, and that no further inspections using computer-based data were planned.

[2] Tr. 8160.

[3] *Id.*

[4] Tr. 8162.

[5] Tr. 8164.

[6] Tr. 8180.

[7] Tr. 8976-78.

[8] Tr. 8968. *See also* Tr. 9005.

[9] *See* Tr. 8955-59.

could make adequate final walkdown inspections hard to make or to trust.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Walter H. Jordan (by PBB)
ADMINISTRATIVE JUDGE

Kenneth A. McCollom (by PBB)
ADMINISTRATIVE JUDGE

Bethesda, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung**, Director

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| **In the Matter of** | **Docket No. 50-440** |
| | **(10 C.F.R. § 2.206)** |
| **CLEVELAND ELECTRIC ILLUMINATING COMPANY, *et al.*** | |
| **(Perry Nuclear Power Plant, Unit 1)** | **January 9, 1984** |

The Director of the Office of Inspection and Enforcement denies a petition requesting an independent analysis of a crane accident during construction of Perry Unit 1, access by the general public to the plant, and initiation of show-cause proceedings to revoke the construction permit. The Director found that adequate analyses of the accident had been performed and that appropriate corrective actions had been taken.

## RULES OF PRACTICE: PETITIONS UNDER 10 C.F.R. § 2.206

The staff will not initiate immediate action to grant the relief requested in a § 2.206 petition in the absence of a demonstration that an imminent hazard to public health and safety exists which warrants immediate relief.

## RULES OF PRACTICE: INITIATION OF SHOW-CAUSE PROCEEDINGS

Show-cause proceedings may be initiated if a substantial health and safety issue is raised, but the Commission will not institute such proceedings to explore the purely economic impacts of licensed activities.

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

By petition dated September 27, 1983, Terry Jonathan Lodge, on behalf of Steven Sass and the Sunflower Alliance, Inc. (hereinafter referred to as the petitioners), requested pursuant to 10 C.F.R. § 2.206 that the Director of the Office of Inspection and Enforcement take the following specific actions with regard to Unit 1 of the Perry Nuclear Power Plant:

- Appoint immediately an independent consulting engineering firm, at the expense of the Cleveland Electric Illuminating Company (CEI or licensee), to conduct a thorough investigation of a construction accident that occurred during the attempted lift of the steam separator (moisture separator) from the reactor vessel on September 15, 1983, and to evaluate exhaustively the safety, structural and economic impacts said events will have upon the Perry construction timetable, said results to be made immediately available to the public.
- Open immediately the containment structure and all related facilities at the Perry plant to grant access to all members of the public wishing to inspect the steam separator and reactor vessel.
- Convene a public hearing into the events of September 15, 1983, at such time as the independent study is prepared, the purpose of said hearing to be to determine whether CEI's construction permit for Unit 1 should be permanently revoked.

On October 14, 1983, the Director acknowledged receipt of the petition and informed the petitioners that their request for immediate action was denied, because no imminent hazard to public health and safety required the immediate relief which the petitioners requested. Moreover, such action was not necessary to ensure the staff's ability to evaluate the matter or its ability to ultimately grant the relief requested. A notice that the petition was under consideration was published in the *Federal Register*. 48 Fed. Reg. 49,713 (1983). The staff has now completed its evaluation of the petition and, for the reasons stated in this decision, the petitioners' request is denied.

## BACKGROUND

The Cleveland Electric Illuminating Company holds Construction Permits No. CPPR-148 (Unit 1) and No. CPPR-149 (Unit 2), issued by the Nuclear Regulatory Commission in 1977, which authorize construction of the Perry plant. The Perry plant is located on Lake Erie in Perry County, Ohio approximately 35 miles northeast of Cleveland, and consists of two boiling water reactors of General Electric design and related facilities for use in the commercial generation of electric power.

On September 15, 1983, General Electric Company was removing certain internal reactor components from the Unit 1 reactor vessel, where they had been placed temporarily while some work was being performed in the upper refueling area, their normal storage location. The first component to be removed, the steam dryer, was lifted from the vessel without incident. However, the next component to be removed, the moisture separator, had not been unbolted from the core shroud prior to the lifting effort. Twenty-eight of the thirty-two holddown bolts for the moisture separator were tightened and the other four bolts were loosely engaged at the time the lift was attempted. The lifting rig broke due to excessive force. The lifting rig, which is an X-shaped I-beam structure, and which is not part of the containment crane, was rated to 53 tons and had previously been tested to 59 tons. The containment crane is rated at 125 tons.

On September 16, 1983, the NRC Senior Resident Inspector and an NRC structural specialist performed a visual inspection of the moisture separator, the areas where the moisture separator was bolted to the core shroud in the reactor vessel, and the crane. No damage was found during that inspection. Subsequent detailed inspections of the crane, the moisture separator, and the reactor vessel by the licensee and by General Electric have found no indications of damage to those components.

In addition to the inspections discussed above, results of metallurgical tests on the break area in the lifting rig have been factored into a conservative analysis by General Electric to estimate the maximum loads which could have been imposed on the lifting rig and on the other components involved. The results of that analysis show that reactor assembly components experienced stresses which are less than maximum allowable values. Additional analysis by Gilbert Associates, Inc. and Nuclear Plant Services resulted in less conservatively estimated loads. NRC Region III representatives reviewed and discussed these analyses during a meeting with CEI at the NRC regional offices in October 1983.

On October 5, 1983, following receipt and testing of a new lifting rig, the moisture separator (with holddown bolts disengaged) was removed from the Unit 1 reactor vessel without incident.

## DISCUSSION OF PETITIONERS' SPECIFIC CONCERNS

As a basis for their request for action, the petitioners cite their understanding of the circumstances and consequences of the lifting incident. The petition (¶ 5) states that, during the attempted lift, there occurred the "breakage of the strongback portion of the crane" and "the body of the containment crane was lifted some eighteen inches (18″) from its track." The lifting rig (or "strongback") which broke is not part of the crane. In addition, there is no indication that the body of the crane lifted 18 inches from its tracks. There were no loads or reactions involved which would have caused such a movement, considering the weight on and configuration of the rolling assemblies. As noted above, inspection of the containment crane found no damage of the type which would be expected if such a movement had occurred.

The petition (¶ 7) also states that "damage to the reactor vessel . . . [occurred] in the form of partially or wholly tearing the vessel from its base, and destroying certain of the vessel's engineered seam welds. . . ." The analysis by General Electric noted above indicated that the stresses imposed on all analyzed parts of the reactor assembly were below the allowable values. In addition, the inspections performed by NRC, the licensee, and General Electric have found no evidence of damage. The reactor vessel (including the other reactor components in place at the time of the incident) weighed approximately 940 tons and therefore could not have been lifted from its base by the maximum upward forces determined by the analysis.

The petition (¶ 9) states that "unquantified stresses have occurred to the steam separator and to the portions of the reactor vessel to which the separator is attached." The stresses are "unquantified" in the sense that they cannot be determined exactly. However, General Electric's analysis determined a maximum loading which could have been applied during the lifting incident. That loading was then used to arrive at conservative (or upper limit) stress values on reactor assembly components, including the moisture separator and the parts of the reactor vessel to which the moisture separator was attached. As noted above, those stresses were, in all cases, below the allowable values.

The petition notes that the records maintained by CEI and its agents did not show that the moisture separator was still bolted down on September 15, 1983 when the initial lift of the moisture separator was attempted. NRC and licensee reviews of General Electric lifting procedures have found inadequacies which resulted in the failure of the records to note the bolted condition. CEI, as the licensee responsible for the proper conduct of licensed activities, has been cited in a Notice of Violation, transmitted to the licensee on December 12, 1983, with NRC Region III Inspection Report 50-440/83-34 and 50-441/83-33 for lack of adequate control measures. The applicable procedures were revised as required and were used on October 5, 1983 to lift the moisture separator from the reactor vessel.

## CONSIDERATION OF PETITIONER'S REQUESTED RELIEF

Neither the matters set forth in the petition nor the circumstances surrounding the lifting incident warrant the relief requested by the petitioners. The petition does not raise a substantial health and safety issue which would cause the staff to initiate show-cause proceedings. *See Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear-1), CLI-78-7, 7 NRC 429, 433-34 (1978), *aff'd sub nom. Porter County Chapter of the Izaak Walton League, Inc. v. NRC,* 606 F.2d 1363 (D.C. Cir. 1979).

Petitioners request that the Commission appoint an independent consulting engineering firm to conduct an investigation of the moisture separator lifting incident, and to evaluate the safety, structural and economic impacts of the event. The staff does not believe that such an additional investigation is warranted by the facts, because General Electric, Gilbert Associates, Inc. and Nuclear Plant Services have performed "worst case" analyses of the structural and safety effects of the incident and have found that, in all cases, stresses imposed were below the allowable values. In addition, inspections by the NRC and the licensee have found no evidence of damage, other than to the lifting rig. Appropriate corrective action has been taken to correct the procedural deficiencies that contributed to the incident. With regard to economic impacts, the Commission will not institute proceedings to explore the purely economic impacts of construction activities or deficiencies at a site. *Cf. Commonwealth Edison Co.* (Byron Station, Units 1 and 2), DD-81-5, 13 NRC 728 (1981), *aff'd sub nom. Rockford League of Women Voters v. NRC,* 679 F.2d 1218 (7th Cir. 1982).

General Electric performed the analysis of the incident and provided a new lifting rig. The lift of the moisture separator from the reactor vessel, which was not a schedule critical path item, was made successfully on October 5, 1983. Required revisions to procedures were completed prior to the lift. The successful lift of the moisture separator, which was observed by the NRC Senior Resident Inspector, and a subsequent lift of the reactor vessel head (which weighs approximately 100 tons) onto the reactor vessel on November 1, 1983, have indicated that the revised procedures are being implemented properly. The NRC Senior Resident Inspector will observe other reactor assembly lifts as appropriate to verify that the lifting procedures are being followed. Because the licensee has taken adequate measures to review the consequences of the lifting incident and to implement corrective action, and because the staff has sufficient information available to resolve its concerns over the safety significance of the incident, an independent investigation by a "consulting engineering firm" is not warranted.

No compelling reasons would require the licensee to open the containment structure and related facilities to the public for inspection of the moisture separator and the reactor vessel. Conversely, there are considerations involving protection of the reactor assembly equipment which would militate against such general access. Such access is unnecessary to discharge properly the Commission's responsibility to ensure adequate protection of public health and safety, nor is such access necessary to ensure that the licensee meets its responsibilities under the Construction Permit and the Commission's regulations. The NRC's inspectors have immediate unfettered access to all parts of the Perry plant, including the moisture separator and the interior of the reactor vessel, and have used that access to inspect those components. *See* 10 C.F.R. § 50.70. The licensee has been cooperative in making information concerning the lifting incident available to the staff, has reviewed the incident, and has taken appropriate corrective actions. The staff has no reason to suspect subterfuge or other deliberate wrongdoing in the licensee's handling of the incident.[1] In sum, there is no adequate basis to order the licensee to provide general access to the plant.

The petitioners also request that the Commission convene a public hearing into the events of September 15, 1983, to determine whether CEI's construction permit for Unit 1 should be revoked. The petitioners' request is essentially for the initiation of show-cause proceedings in accordance with 10 C.F.R. § 2.202. Initiation of show-cause proceedings is not warranted in these circumstances. As discussed in this decision, the lifting incident does not raise a substantial safety issue that would warrant initiation of such proceedings or that would call for the extreme remedy of construction permit revocation under the Commission's enforcement policy. *See* 10 C.F.R. Part 2, Appendix C, § IV.C(3), *published in* 47 Fed. Reg. 9987, 9992 (1982). Appropriate enforcement action has been taken for the procedural deficiencies associated with the incident in the form of a Notice of Violation under 10 C.F.R. § 2.201, and the licensee has taken action to correct the deficiencies. Thus, based on the analyses and inspections discussed above, which have shown that no damage to the reactor assembly resulted from the lifting incident, and the fact that the procedural inadequacies have been corrected, no

[1] While noting that the licensee informed the NRC of the lifting incident, the petitioners allege that the licensee did not "voluntarily disclose" the incident to the public. Although the licensee may have been required to report the incident to the Commission, NRC requirements do not impose an obligation to report such incidents directly to the press or other members of the public.

sufficient basis exists to initiate proceedings to revoke the Perry Unit 1 construction permit.

## CONCLUSION

For the reasons stated in this decision, the petitioners' request has been denied. A copy of this decision will be filed with the Office of the Secretary of the Commission for the Commission's review in accordance with 10 C.F.R. § 2.206(c) of the Commission's regulations. This decision will become the final action of the Commission 25 days after the date of issuance unless the Commission, on its own motion, institutes a review of the decision within that time.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 9th day of January 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung,** Director

**In the Matter of** **Docket Nos. 50-329**
**50-330**
**(10 C.F.R. § 2.206)**

**CONSUMERS POWER COMPANY**
**(Midland Plant, Units 1 and 2)** **January 12, 1984**

The Director of the Office of Inspection and Enforcement grants a portion of a petition granted in part and denied in part on October 6, 1983 (DD-83-16, 18 NRC 1123).

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

On October 6, 1983, I issued a Director's Decision Under 10 C.F.R. 2.206, DD-83-16, 18 NRC 1123, which granted in part and denied in part a petition dated June 13, 1983, submitted by Billie Pirner Garde of the Government Accountability Project on behalf of the Lone Tree Council and others. The petitioners had requested that, among other relief, the Commission require a management audit of Consumers Power Company's performance on the Midland project. In my decision, I determined that a management audit was not necessary as a condition for going forward with the licensee's program to complete construction of the Midland project. However, I noted that the "staff [would] continue to review information concerning the licensee's performance in other areas to determine whether an audit is required." 18 NRC at 1131.

I have completed my review of information related to a violation of a condition of the Midland construction permits which was imposed by the Director of Licensing, Office of Nuclear Reactor Regulation, in accordance with an order of the Atomic Safety and Licensing Board dated April 30, 1982. *See Consumers Power Co.* (Midland Plant, Units 1 and 2), LBP-82-35, 15 NRC 1060, 1072-73 (1982). This violation is an addition to the history of quality assurance problems at the Midland site which demonstrates that the licensee's management has not been effective in providing the attention to detail and high quality standards necessary to assure the proper construction of this facility. In view of this history, and the recently identified violation of the Midland construction permits, I have now determined that an appraisal of Consumers Power Company's management of the Midland project is required. The reasons for this action are explained more fully in the Confirmatory Order that I have issued today. The order requires Consumers Power Company, within 30 days of its effective date, to submit to the Region III Administrator for review and approval, a plan for an independent appraisal of site and corporate management organizations and functions. The management appraisal is to develop recommendations where necessary for improvements in management communications, control and oversight. Upon its approval, the plan will be implemented in accordance with a schedule of milestone completion dates.

In view of the issuance of the Confirmatory Order, the petitioners' request pertaining to a management audit is granted.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 12th day of January 1984.

[The Confirmatory Order has been omitted from this publication, but has been published in the *Federal Register*, 49 Fed. Reg. 2562 (Jan. 20, 1984)]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung**, Director

**In the Matter of** **Docket No. 50-358**
**(10 C.F.R. § 2.206)**

**CINCINNATI GAS & ELECTRIC COMPANY, *et al.***
**(William H. Zimmer Nuclear Power Station, Unit 1)** **January 13, 1984**

The Director of the Office of Inspection and Enforcement denies a petition submitted by Thomas Devine of the Government Accountability Project on behalf of the Miami Valley Power Project requesting action with respect to the William H. Zimmer Nuclear Power Station.

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

By letter dated December 14, 1983, Thomas Devine of the Government Accountability Project, on behalf of the Miami Valley Power Project (MVPP), requested pursuant to 10 C.F.R. § 2.206 that the Commission defer any action on the Course of Action proposed by the Cincinnati Gas and Electric Company (CG&E) for the William H. Zimmer Nuclear Power Station until three events had occurred: (1) the Commission was briefed by whistleblowers nominated by MVPP from Bechtel Power Corporation's nuclear projects; (2) public release of a pending report of the Commission's Office of Investigations into alleged wrongdoing at Zimmer; and (3) comments were received from the public regarding the Commission's whistleblower briefing and the OI report. The Commission referred MVPP's request to the staff for appropriate action. In a letter dated December 16, 1983, which transmitted the denial of another 2.206 request filed by MVPP, MVPP was informed that its latest request had been denied and that the detailed rationale for the denial would be forthcoming.

In support of its request, MVPP relies on various comments it has filed with the Commission related to CG&E's proposed Course of Action. MVPP submitted another letter in support of its request, dated December 16, 1983, focusing on Bechtel Power Corporation's competence.

CG&E proposed the Course of Action in response to the requirements of the Commission's Order to Show Cause and Order Immediately Suspending Construction which was issued in November 1982. CLI-82-33, 16 NRC 1489 (1982). Submission of the Course of Action was required by the order as one of the initial steps toward any eventual resumption of safety-related construction at Zimmer. The Course of Action was approved by James G. Keppler, Regional Administrator of NRC Region III, by letter dated December 16, 1983. Mr. Keppler's letter and its appended documents set forth the rationale for the agency's approval in that matter.

The issues raised in MVPP's most recent request were considered by the staff prior to approval of the Course of Action. Throughout the staff's review of the proposed Course of Action, MVPP's submittals, as well as the comments of other persons, were reviewed. Consideration was given to the performance of the Bechtel Power Corporation in other nuclear projects. The staff also considered the results, to date, of the Office of Investigations' ongoing investigation. Of the five reasons cited by MVPP in support of its petition, none provided the staff with new information such that deferral of approval of the Course of Action was justified.

As to the specifics of the five reasons cited by MVPP, the first concerned the retention of the Henry J. Kaiser Corporation to verify the quality of work covered by the American Society of Mechanical Engineers' (ASME) Code. This role for Kaiser is the minimum necessary to discharge its responsibilities under its "N" stamp for ASME Code work performed to date at Zimmer. Kaiser's verification activities will be subject to review by the National Board of Boiler and Pressure Vessel Inspectors and all of the verification activities will be overseen by Bechtel. The staff has considered this reduced role of Kaiser and found

[1] Although MVPP's December 14, 1983 letter is being treated as a petition under 10 C.F.R. § 2.206 of the Commission's regulations, it does not fall squarely within the class of requests for relief provided for under that regulation. In particular, MVPP does not request initiation of a proceeding, as contemplated by 10 C.F.R. § 2.206(a).

it acceptable with the oversight role of Bechtel. Should the OI investigation develop information that justifies a change to the approved Course of Action, appropriate action will be taken.

MVPP also asserted as part of its first reason that Bechtel's assumption of the role of constructor created a greater conflict-of-interest than was raised by the question of whether Bechtel could serve as both independent management reviewer and Project Director. The staff does not agree that a conflict-of interest question is raised by Bechtel serving as both Project Director and constructor. This question was reviewed as part of the Course of Action, and the staff found Bechtel's expanded role acceptable.

As a second reason for its petition, MVPP refers to alleged abuses by Bechtel in the reverification program at Diablo Canyon and to Bechtel's cost estimate for completion of the Zimmer project. In approving the Course of Action, the staff reviewed the experience of the key Bechtel personnel assigned to the Zimmer project, conducted interviews with most of them, and concluded that Bechtel had assigned well-qualified personnel to the Zimmer project. The staff also considered Bechtel's extensive experience in nuclear power plant construction, particularly at plants where Bechtel assumed responsibility from other architect-engineers or constructors when construction was well under way. Nevertheless, the allegations of MVPP regarding Bechtel's performance at Diablo Canyon will be reviewed by the NRC and, if substantiated, the staff will take whatever action is appropriate with respect to Bechtel's activities at Zimmer. In addition, the staff will be mindful of the issues raised by those allegations during its inspection process at the Zimmer facility and in the review of the Plan to Verify the Quality of Construction (PVQC).

The past or present cost estimates of Bechtel will not control the outcome of the review of the PVQC or the Continuation of Construction Plan (CCP). Both of these documents have been submitted to the Regional Administrator and are currently under review. The decision on the acceptability of the PVQC and CCP will be based on the staff's judgment of what is required for the public health and safety and not any previously agreed upon cost estimates made by either Bechtel or CG&E.

The third reason cited by MVPP relates to statements by CG&E regarding its intent to shave Bechtel's cost estimates and whether there is any longer any valid issue with respect to the safety of Zimmer. MVPP infers from the licensee's "oft-announced intent to shave the Bechtel cost-estimate" that CG&E is faced with a financial conflict-of-interest regarding Zimmer. MVPP Letter at 2 (Dec. 14, 1983). The staff does not object to the statements of CG&E that it desires to "shave the Bechtel cost estimates" so long as quality is not compromised. Should any evidence be developed that quality is being adversely affected by economic considerations, appropriate regulatory action will be taken.

MVPP also references a November 10, 1983 letter written by Joe Williams, Jr., Senior Vice President for Nuclear Operations of CG&E, as illustrative of CG&E's poor judgment with respect to safety issues. The staff asked Mr. Williams to provide an explanation as to the meaning of his remark in the letter that "the issue of safety is no longer a valid one." By letter dated December 16, 1983, Mr. Williams stated that his remark was intended to mean that safety issues will be resolved by CG&E's proposed plans to verify construction and correct prior inadequate work. The staff has no basis to draw a negative inference from Mr. William's letter, as clarified.

The staff takes no position as to Mr. Williams' refusal to attend a public forum regarding Zimmer. While attendance at a community meeting may be beneficial to the company's public image, it is not the NRC's responsibility to monitor CG&E's actions and statements for this purpose. The NRC is interested in the licensee's ability and the actions it takes to manage the construction of a plant in accordance with the Commission's requirements.

The fourth reason cited by MVPP is the alleged lack of an NRC staff response to comments submitted by MVPP on December 5, 1983 concerning the proposed Course of Action. MVPP had requested timely responses to eighteen issues raised in its December 5th comments in order to determine "whether other legal initiatives" were deemed necessary. The staff's responses to these issues had not been transmitted as of the time of MVPP's December 14th petition. The staff's review was reflected in the "NRC Response to Comments on CG&E's Course of Action," issued as an attachment to Mr. Keppler's December 16th approval letter.[2]

MVPP also asserts that there has been an absence of public participation since Bechtel assumed the role of constructor from Kaiser. The staff did, however, consider public comments and the comments in MVPP's December 5th submittal in reaching its decision on the Course of Action. Additionally, opportunity for public comment on the Course of Action had been previously provided and, in accordance with the staff plan of action dated December 22, 1982, there will be opportunities for additional public comment during the remainder of the review process

[2] Matters raised in MVPP's letters which bear on the adequacy of CG&E's PVQC and CCP will be considered in connection with the staff's review of those proposals.

for the PVQC and CCP. Although this process for soliciting public comment is not mandated by statute, regulation or the Commission's order, the staff initiated this process because it viewed public comment as potentially helpful in assessing the proposals which CG&E is required to make under the Commission's order. In sum, there will be adequate opportunity for the public to comment on the PVQC and CCP.

As a final reason for deferring approval of the proposed Course of Action, MVPP raises, as it has before, the issue of Mr. Keppler's impartiality. Previously MVPP had recommended that Mr. Keppler be removed from the approval process under the Commission's order. Mr. Keppler is viewed not only by the Director, but also by the Commission, as a man of integrity and competence. He is dedicated to ensuring that construction of the Zimmer facility will only proceed in accordance with the Commission's requirements.

The specific points raised by MVPP do not support their attack on Mr. Keppler's impartiality. MVPP incorrectly states that Mr. Keppler recommended to the Commission in the fall of 1982 that Zimmer not be shut down. To the contrary, Mr. Keppler's October 1982 recommendation to the Commission was that safety-related construction at Zimmer should be suspended. MVPP also mischaracterizes meetings held on November 17, 1982 between NRC Region III and CG&E and then among Region III, CG&E and Bechtel as having included advice to CG&E and Bechtel on how to obtain the Commission's approval of Bechtel as the independent management reviewer under section IV.B.1 of the Commission's order. The record (a publicly available November 24, 1982 memorandum for Region III files) indicates that this meeting was held for the entirely appropriate purpose of assuring that CG&E and Bechtel fully understood the order and what was required under it.[3]

MVPP also refers to allegations made by James McCarten, a former investigator in Region III, regarding the handling of the Zimmer investigation. These allegations were referred to the Commission in the summer of 1983. *See* "Report to the Chairman on Allegations of Thomas Applegate Concerning Conduct of the Office of Inspector and Auditor" (the "Hoyt Report") at 18 n.29 (July 12, 1983). Although the Commission has not expressly addressed the McCarten allegations, subsequent to the issuance of the Hoyt Report the Commission issued a

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[3] The staff did advise CG&E of concerns it had with the performance of Bechtel's Ann Arbor Power Division at Midland and stated that CG&E would have to address these concerns. The staff made no commitments, however, as to whether it would find Bechtel qualified to conduct the independent management review at Zimmer. Additionally, there is no basis for MVPP's assertion that the November 17th meeting was improper because of *ex parte* considerations. Since Mr. Keppler was not acting in an advisory capacity to the Commission in the exercise of its adjudicatory responsibilities, the *ex parte* provisions of the Commission's rules of practice (10 C.F.R. § 2.780) had no applicability to this meeting.

memorandum to Mr. Keppler expressing the Commission's "continued support and confidence" in him, and stating that: "The Commission continues to have high regard for your contribution to the agency." Memorandum from Chairman Palladino to James G. Keppler (October 6, 1983).

Moreover, Mr. Keppler was not the only decisionmaker with regard to the Course of Action. In addition to the advice and counsel provided by his regional staff, Mr. Keppler worked closely with senior NRC officials and their staff in the Offices of Nuclear Reactor Regulation, Inspection and Enforcement, and the Executive Legal Director. His decision was the product of careful review and analysis and is fully supported by the staff.

Finally, the assertion that whistleblowers have not had any avenue available to them to bring to the Commission's attention concerns regarding Bechtel's qualifications to assume the role of constructor at Zimmer is without support. As noted above, the allegations brought to the Commission's attention in MVPP's December 5, 1983 submittal will be reviewed and the implications of any findings for Bechtel's role at Zimmer will be considered by the staff.

Any person is free to contact representatives of the Commission at any time with safety information. The staff intends to follow up concerns expressed by persons who have information regarding the construction of the Zimmer plant.

For the reasons set forth in this decision, MVPP's request to defer any judgment or decision regarding CG&E's Course of Action has been denied.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 13th day of January 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Gary J. Edles, Chairman**
**Dr. W. Reed Johnson**
**Howard A. Wilber**

**In the Matter of** **Docket No. 50-537-CP**

**UNITED STATES DEPARTMENT OF ENERGY**
**PROJECT MANAGEMENT CORPORATION**
**TENNESSEE VALLEY AUTHORITY**
**(Clinch River Breeder Reactor Plant)** **February 29, 1984**

Acting on appeals by two intervenors from Licensing Board actions (following termination of the Clinch River project and the Licensing Board's dismissal of the intervenors from the proceeding for a construction permit (CP) for the project) that, *inter alia,* limited the intervenors' participation in the Limited Work Authorization (LWA) proceeding (on remand to consider issues of site redress) to giving limited appearance statements, the Appeal Board vacates the Licensing Board action limiting LWA participation and denies the remainder of the appeals.

## LIMITED WORK AUTHORIZATION (LWA): AVAILABILITY

Under 10 C.F.R. § 50.10(e), an applicant for a construction permit may seek early approval of certain types of site preparation activity by requesting issuance of an LWA.

**CONSTRUCTION PERMIT PROCEEDINGS: INITIAL DECISION**

A licensing board is required to issue an initial decision in a case involving an application for a construction permit even if the proceeding is uncontested. 10 C.F.R. § 2.104(b)(2) and (3).

**LICENSING BOARDS: AUTHORITY TO REGULATE PROCEEDINGS**

Licensing boards have the authority to regulate the course of a proceeding and to limit an intervenor's participation to issues in which it is interested. 10 C.F.R. §§ 2.718, 2.714(e) and (f).

**RULES OF PRACTICE: RESPONSIBILITY OF PARTIES**

Parties may not dart in and out of proceedings on their own terms and at their convenience and expect to enjoy the benefits of full participation without responsibilities. *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-691, 16 NRC 897, 907 (1982).

## APPEARANCES

**Barbara A. Finamore** and **S. Jacob Scherr,** Washington, D.C., for the appellants Natural Resources Defense Council, Inc., and the Sierra Club.

**George L. Edgar** and **William D. Luck,** Washington, D.C., for the appellees Project Management Corporation and the U.S. Department of Energy.

**Sherwin E. Turk** for the Nuclear Regulatory Commission staff.

# MEMORANDUM AND ORDER

**Opinion for the Board by Messrs. Edles and Wilber:**

## I.

This proceeding involves a request by the joint applicants for a permit to construct the Clinch River breeder reactor. Under the Commission's regulations, an applicant for a construction permit may seek early approval of certain types of site preparation activity by requesting issuance of a limited work authorization (LWA).[1] The applicants did so in this case and, in a partial initial decision issued on February 28, 1983, the Licensing Board disposed of various site suitability issues and authorized issuance of the LWA.[2] Intervenors Natural Resources Defense Council (NRDC) and the Sierra Club filed an appeal from the Board's decision, accompanied by a request for a stay pending appellate review. We denied the request for stay,[3] and appellate proceedings began.

While the appeal was pending, the Licensing Board was moving forward with the remaining construction permit phase of the proceeding. On June 21, 1983, the intervenors filed a motion with the Licensing Board to withdraw their contentions on the outstanding permit issues because limited resources prohibited their continued full participation in the upcoming evidentiary hearings. At a conference with the parties held on June 29, the Licensing Board granted the intervenors' request to withdraw their contentions.[4] The Board went on to observe, however, that "it would appear to the Board . . . that the Intervenors no longer are parties to this proceeding . . . and that the Intervenors will be dismissed as parties to the construction permit proceeding."[5] The intervenors did not appeal the Board's decision. Evidentiary hearings in this now uncontested phase of the proceeding were held during August 1983, and the parties thereafter submitted the usual proposed findings of fact and conclusions of law as the final step before issuance of the Board's initial decision.[6]

In October 1983, Congress declined to appropriate further funds for the Clinch River project and it became clear that the project would soon be terminated. On November 23, NRDC filed a motion with the Licensing Board seeking to re-enter the proceeding in order to raise the issue of the effect of the termination of the project on that part of the case still pending before the Board. On the same day, the intervenors filed a

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[1] *See* 10 C.F.R. § 50.10(e).

[2] LBP-83-8, 17 NRC 158. In fact, site preparation activities began even before issuance of the LWA because the Commission granted the applicants an exemption from the requirement to obtain an LWA before beginning work. CLI-82-23, 16 NRC 412 (1982). The exemption was challenged in court by the intervenors and the Commission's decision was reversed and remanded. *Natural Resources Defense Council v. NRC*, 695 F.2d 623 (D.C. Cir. 1982). Site preparation work went forward, however, because the court declined to grant a stay of the Commission's decision. The Commission clarified its earlier decision and reaffirmed its grant of the exemption in an opinion issued on January 5, 1983. CLI-83-1, 17 NRC 1.

[3] ALAB-721, 17 NRC 539 (1983).

[4] Tr. 7329.

[5] *Id.* at 7330.

[6] A Licensing Board is required to issue an initial decision in a case involving an application for a construction permit even if the proceeding is uncontested. 10 C.F.R. § 2.104(b)(2) and (3).

motion with us to terminate appellate proceedings, vacate the partial initial decision in the LWA phase of the case, and authorize revocation of the LWA. On December 15, we granted the motion insofar as it requested termination of appellate proceedings and vacation of the LWA partial initial decision, but remanded the matter to the Licensing Board for consideration of whether any conditions to ameliorate environmental impacts of site preparation should be imposed.[7]

On January 20, 1984, the Licensing Board issued an order denying NRDC's request to be readmitted to the proceeding.[8] The Board concluded that "[t]he attempts of NRDC to re-intervene after deliberately withdrawing all remaining contentions and terminating its status as a party, are not conducive to orderly practice."[9] Simultaneously, the Board issued a 90-page Memorandum of Findings containing its analyses and conclusions regarding numerous issues in the construction permit proceeding.[10] Lastly, the Board issued a notice in response to our remand order setting March 14, 1984, as the date for a conference to discuss appropriate measures for site redress from the activities conducted under the LWA. That notice authorized "former intervenors" such as NRDC and the Sierra Club to participate in the conference "by making appropriate limited appearance statements (10 C.F.R. § 2.715)."[11]

NRDC appeals from the Board's order denying intervention and both NRDC and the Sierra Club appeal the Board's determination restricting them to "limited appearance" status in the proceedings on remand. Regarding its request to re-enter the construction permit phase of the case, NRDC claims that the need for orderly practice should not bar readmission because the Board's dismissal was without prejudice. NRDC also objects to the Board's failure to address the criteria for late intervention. As to the limitation of both groups to "limited appearance" status at the upcoming conference, the intervenors claim that they have participated fully with respect to LWA issues from the outset of the case and that termination of the LWA appeal proceedings should have had no effect on their ability to participate in those LWA proceedings still pending before the Licensing Board.

The applicants and the NRC staff filed answers opposing the intervenors' appeals. Essentially, the applicants claim that the intervenors' action in seeking and obtaining termination of appellate proceedings and vacation of the partial initial decision in the LWA portion of the case,

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[7] ALAB-755, 18 NRC 1337.

[8] Order Regarding NRDC Motion to Intervene (unpublished).

[9] *Id.* at 5.

[10] LBP-84-4, 19 NRC 288.

[11] Notice of Conference with Parties (Jan. 20, 1984) at 2 (unpublished).

coupled with their voluntary withdrawal from the remainder of the case, effectively extinguished whatever contingent rights may have existed. The staff argues that the Licensing Board's dismissal of NRDC and the Sierra Club as parties was consistent with the limitation of NRC proceedings to parties manifesting an interest in discrete issues. The staff also asserts that the requirement that NRDC and the Sierra Club make only limited appearances at the upcoming site redress conference is proper because neither is any longer a party to the proceeding.

For reasons explained below, we find that NRDC and the Sierra Club are entitled to participate fully as intervenors in the proceedings on remand, but that the Licensing Board did not act unreasonably in refusing to authorize NRDC to re-enter the remainder of the construction permit phase of the case.

## II.

A. At the time we issued our order terminating appellate proceedings in connection with the LWA decision, and remanding the case to the Licensing Board for consideration of site redress issues, NRDC and the Sierra Club were parties to the proceeding. Nothing in our remand order was designed to alter their status as intervenors in the LWA portion of the case. Thus, to the extent the Licensing Board, without explanation, now purports to restrict their participation in the upcoming conference to "limited appearance" status, its action is inconsistent with the remand ordered in ALAB-755. Moreover, although the Licensing Board's recent announcement clouds the issue of these parties' continued status in the overall case, nothing in that Board's June 29 decision suggests an intent to deprive either NRDC or the Sierra Club of its right to pursue LWA issues.[12]

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[12] The following excerpts from the June 29 conference before the Licensing Board are illuminating.

Chairman Miller: "You are a party as to whatever you may have raised or done on appeal. I think there is no controversy as to that. But above and beyond that which is pending now before the Appeal Board and which it has jurisdiction, this Board does not have any jurisdiction of matters that pass to the Appeal Board . . . There is one other matter that has been alluded to. That is whether our order dismissing Intervenors as parties to this or future construction permit proceedings should be without prejudice. Let me say simply that the Board does not intend to rule upon that matter . . . Therefore, this ruling at this time is neither with prejudice or without prejudice. *We will abide by whatever is done by the Appeal Board in whatever decisions it might make* or this Board might make in the future." Tr. 7318, 7332-33 (emphasis added)

Mr. Edgar (applicants' counsel): ". . . [W]hatever rights Intervenors may have vis-a-vis the LWA proceeding and the Board's decision exists and they are not affected by what the Board may do at this time . . ." Tr. 7314.

Mr. Turk (staff counsel): "[W]e certainly do not not want them to be prejudiced from prosecuting the appeal which they have already filed in the first PID. And in the event the first PID is reversed and further evidence must be taken . . . then we would not oppose their participation as to those matters." Tr. 7316-17.

The applicants argue that, insofar as the LWA proceeding was concerned, the intervenors never advanced any contentions related to the environmental impact of site preparation activities.[13] The staff similarly contends that NRDC had no interest in the redress issue as long as no construction permit was to be issued.[14] We reject these arguments.

To begin with, it cannot be said that the intervenors have manifested no interest in the question of site preparation. They argued before the Commission during the exemption proceedings that site preparation activities alone may result in significant adverse impacts. The Commission rejected their arguments in part because site redress was available in the event the project were to be terminated.[15]

Moreover, the focus of the proceeding before the Licensing Board thus far has been on the use of the land for construction of the project.[16] (In that context, as the applicants and the staff point out, the matter has been uncontested.) Now before the Board, for the first time, is the question of site *redress* in light of the abandonment of the project. These intervenors have been active participants in connection with the LWA aspects of the case and we do not believe that their decision to concentrate their attention on technical issues unrelated to use of the land at the earlier stages of the proceeding should prevent their participation now that site redress has become the only issue in the case. Redress is a matter with which the intervenors are concerned and we see no public interest purpose in circumscribing their participation at this stage.[17]

B. NRDC's request to re-enter that phase of the case dealing with non-LWA issues stands on a somewhat different footing. Although the LWA and construction permit aspects of the case are simply separate phases of the same proceeding, licensing boards have the authority to regulate the course of the proceeding and limit an intervenor's participation to issues in which it is interested.[18] In this case, as the staff points out, the Board's June 29 decision had the effect of declaring that neither NRDC nor the Sierra Club would be permitted to participate further in that portion of the case still pending before the Licensing Board on

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[13] Applicants' Answer to Intervenors' Appeal (February 21, 1984) at 2.

[14] NRC Staff's Brief in Opposition (February 21, 1984) at 21.

[15] CLI-82-23, *supra*, 16 NRC at 424 n.4.

[16] *See*, for example, LBP-83-8, *supra*, 17 NRC at 247-50.

[17] *Cf. Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-244, 8 AEC 857, 864-70 (1974), *reconsideration denied*, ALAB-252, 8 AEC 1175, *aff'd*, CLI-75-1, 1 NRC 1 (1975) (intervenor should not be "benched on the sidelines" even if it was not an original proponent of an issue).

[18] 10 C.F.R. §§ 2.718 and 2.714(e) and (f).

issues unrelated to the LWA.[19] The intervenors did not appeal the Board's determination. Rather, they assumed the risk that the Licensing Board might eventually take some action on non-LWA matters inconsistent with their interests. Although termination of the project is a new development, it is plainly not one that should have been wholly unanticipated when the intervenors withdrew. At that time, it was not unreasonable for the intervenors to assume either that the Licensing Board would simply complete the remaining phase of the case and, in due course, issue a further partial initial decision, or that Congress might decline to provide further funds for the Clinch River project, with the result that the project would be terminated. Thus, we cannot find that the Licensing Board acted unreasonably in denying the request to be readmitted. As we observed in our *Midland* decision:

> Parties may not dart in and out of proceedings on their own terms and at their convenience and still expect to enjoy the benefits of full participation without the responsibilities.[20]

NRDC argues that, in reaching its decision, the Licensing Board did not in terms consider the five factors ordinarily evaluated when deciding whether or not to permit late intervention.[21] We do not believe that an express evaluation of those factors would lead to a different result.

To begin with, and most important for decisional purposes, we think that NRDC's participation in the remanded LWA proceedings will fully protect its interests. Thus, we find against NRDC on factor 2. The project, after all, has been terminated, and the only issues that need to be resolved concern site redress. NRDC sought to re-intervene in the non-LWA proceedings to argue that (i) the conditions for grant of a construction permit cannot or have not been met, and (ii) the program objectives for the project will not be achieved.[22] The applicants concede that NRDC is correct in both respects,[23] and nothing in any of the Licensing Board's issuances is to the contrary. So we see little left for

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[19] NRC Staff's Brief in Opposition (February 21, 1984) at 13.

[20] *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-691, 16 NRC 897, 907 (1982).

[21] Those five factors, set forth in 10 C.F.R. § 2.714(a)(1), are as follows:
- (i) Good cause, if any, for failure to file on time.
- (ii) The availability of other means whereby the petitioner's interest will be protected.
- (iii) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.
- (iv) The extent to which the petitioner's interest will be represented by existing parties.
- (v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding.

[22] *See* Motion of Natural Resources Defense Council, Inc. to Intervene (November 23, 1983) at 5-6.

[23] *See* Applicants' Answer to Intervenors' Appeal (February 21, 1984) at 3.

further resolution. To be sure, the Licensing Board has issued, over NRDC's objection, what it styles a "Memorandum of Findings." The *nature* of that Memorandum is unclear. But its *status* is perfectly clear: it has no operative effect.[24] Thus, we perceive no genuine harm to NRDC from its issuance.

Second, this is not the most compelling case of good cause for late intervention. As explained above, intervenors were already participating in the construction permit phase of the case and elected to withdraw at a time when termination of the Clinch River project should clearly have been one foreseeable outcome (indeed, the intervenors may well have elected to conserve their limited resources in contemplation of that outcome).

It is unclear to what extent factors three, four or five are relevant to a proceeding that is effectively over and will soon be terminated formally for mootness. In any event, we believe that factor two is decisionally overriding in the context of this case.

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The Licensing Board's determination to limit the intervenors' participation in the proceedings on remand to a limited appearance statement is *vacated as inconsistent with ALAB-755.* In all other respects, the appeals are *denied.*

It is so ORDERED.

FOR THE APPEAL BOARD

Barbara A. Tompkins
Secretary to the
Appeal Board

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[24] The Memorandum is not a partial initial decision in the usual sense. Although various issues are "resolved" in the applicants' favor and none appears to remain for later resolution, the Memorandum does not authorize issuance of any form of license. Nonetheless, the Board specifically declined to characterize its Memorandum as an "advisory opinion." LBP-84-4, 19 NRC 288, 293. Rather, it observed: "[T]his Memorandum of Findings [is] somewhat unprecedented procedurally .... It is sufficient to issue only a memorandum tailored to the unusual posture of this proceeding, for whatever assistance it may provide to the NRC now or in the future." *Id.* at 291, 293. Perhaps, "as with the subject of a once popular song, being a combination thereof, it is neither swan nor goose, but truly 'swoose.'" *Saginaw Transfer Co. v. United States,* 275 F. Supp. 585, 588 (E.D. Mich. 1967), *quoting Chemicals in Aggregate Shipments — Midland, Mich. to the East,* 326 I.C.C. 657, 665 (1965).

**Opinion of Dr. Johnson, dissenting in part:**

I do not agree that the intervenors are still a party to the proceedings on remand. Our remand of the case to the Licensing Board was only for the purpose of considering site redress, clearly a matter unrelated to any issue then on appeal. Thus, the Licensing Board's notice according NRDC and the Sierra Club the right to participate on only a limited appearance basis is not inconsistent with ALAB-755.

My interpretation of the intent of the Licensing Board's determination of June 29, 1983 (*see* note 12 of the majority opinion, and accompanying text, *supra*), is that intervenors were dismissed except for matters they had raised expressly on appeal. The majority finds (and I agree) that a licensing board may, pursuant to 10 C.F.R. § 2.714(e) and (f), limit participation to those issues in which a party has demonstrated a genuine interest. In my view, the intervenors have not manifested any genuine interest in the redress issue sufficient to justify their participation as full parties. Significantly, when they sought immediate termination of the LWA appellate proceedings, they did not attempt to raise the redress issue. Rather, they urged us simply to order revocation of the LWA, presumably satisfied to leave to the applicants and the staff alone whatever redress may be needed. They have also not demonstrated any genuine expertise in the question of redress, and I see no public purpose to be served by their participation on the redress issue above and beyond that allowed by the Licensing Board.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Herbert Grossman**, Chairman
**Glenn O. Bright**
**Dr. Jerry Harbour**

**In the Matter of** **Docket No. 50-460-CPA**
**(ASLBP No. 83-485-02-CPA)**

**WASHINGTON PUBLIC POWER SUPPLY SYSTEM**
**(WPPSS Nuclear Project No. 1)** **February 1, 1984**

In a proceeding to determine whether Applicant has demonstrated "good cause" for the construction completion date in the construction permit to be extended, the Licensing Board grants Applicant's and NRC Staff's motions for summary disposition in Applicant's favor.

## CONSTRUCTION PERMIT: EXTENSION OF COMPLETION DATE (GOOD CAUSE)

Where the Applicant has demonstrated valid reasons for delaying construction, the Board will permit the construction completion date to be extended without reaching a judgment on the advisability of completing the plant.

## CONSTRUCTION PERMIT: EXTENSION OF COMPLETION DATE (REASONABLENESS OF PERIOD)

The reasonableness of the period of the requested construction completion date extension cannot be challenged on grounds of insufficiency.

CONSTRUCTION PERMIT: EXTENSION OF COMPLETION DATE (HEALTH, SAFETY AND ENVIRONMENTAL EFFECTS)

A consideration of the health, safety or environmental effects of delaying construction cannot be heard at the construction permit extension proceeding, but must await the operating license stage.

# MEMORANDUM AND ORDER

**(Granting Applicant's and NRC Staff's Motions for Summary Disposition)**

## Memorandum

This is a proceeding to determine whether Applicant should be granted an amendment to extend the completion date stated in its construction permit. Intervenor contends that "good cause" does not exist for the extension of the construction permit completion date, as required by Section 185 of the Atomic Energy Act and 10 C.F.R. § 50.55(b), and that the extension requested is not for a reasonable period.

Applicant and NRC Staff have moved for summary disposition on the basis of affidavits and other documents annexed to their respective motions. Intervenor opposes the summary disposition motions and requests that an evidentiary hearing be convened.

We grant Applicant's and NRC Staff's motions for summary disposition and dismiss Intervenor's admitted contention.

### I. BACKGROUND

On July 21, 1981, Applicant filed an application for an extension of its construction permit completion date from January 1, 1982 until June 1, 1986. On March 18, 1982, Intervenor, the Coalition for Safe Power (CSP), filed a request for hearing. On October 8, 1982, the Commission issued an Order, CLI-82-29, 16 NRC 1221, concerning CSP's request for hearing, which provided Commission guidance on the scope of construction permit extension proceedings and determined that only one contention raised by CSP would be litigable if properly particularized and supported. The Commission Order referred the petition filed by CSP to a licensing board to determine if the other hearing requirements of the Commission's regulations had been met and, if so, to conduct an appropriate proceeding.

On January 17, 1983, Applicant served on the Board and the parties copies of a request to the Staff that its pending amendment request for an extension to June 1, 1986 for completion of construction be modified to allow completion by June 1, 1991. Applicant stated therein its understanding that the request would be treated as a modification of the pending amendment rather than as a new amendment request.

The original requested extension, until June 1, 1986, was premised on the construction having proceeded slower than anticipated. Intervenor challenged that extension on the ground that poor management practices had resulted in delay and that, consequently, there was no good cause for the delay. Intervenor acknowledged that Applicant had not intentionally delayed construction.

The supplemental request for extension from June 1, 1986 until June 1, 1991, however, was necessitated by Applicant's intention to halt its construction for up to 5 years. Intervenor challenged that additional period of requested extension as not satisfying the "good cause" requirement of the Atomic Energy Act and Commission regulations, and the 5-year period as not being a reasonable period of time.

In our unpublished Orders of February 23, 1983 and March 23, 1983, we rejected any contentions that might relate to the original period of requested extension in the pending application, from January 1, 1982 until June 1, 1986. We determined that allegations of poor management practices resulting in construction delays are not sufficient to satisfy the Commission's guidance in CLI-82-29, *supra,* that equated a lack of good cause with being dilatory. Since Intervenor had made no showing that Applicant's requested extension until 1986 was the result of Applicant's being dilatory, we would not entertain any contentions regarding that time period.

However, with regard to the supplemental period of extension, from June 1, 1986 until June 1, 1991, we admitted the following contention:

> *Amended Contention No. 2*
>
> Petitioner contends that the Permittee's decision in April 1982 to "defer" construction for two to five years, and the subsequent cessation of construction at WNP-1, was dilatory. Such action was without "good cause" as required by 10 C.F.R. 50.55(b). Moreover, the modified request for extension of completion date to 1991 does not constitute a "reasonable period of time" provided for in 10 C.F.R. 50.55(b).

It is this contention, the only one admitted in this proceeding, that Applicant and Staff move to dismiss in their respective motions for summary disposition.

## II. STATEMENT

According to Applicant's discussion of the facts, it was the Bonneville Power Administration (BPA), an agency of the U.S. Government established by the Bonneville Project Act of August 20, 1937, that required the halt in construction of WNP-1. Applicant has under construction three nuclear projects, WNP-1, WNP-2, and WNP-3. The financing of WNP-1 has been solely through the sale of bonds. Under agreements to which Applicant and BPA are parties, Applicant has agreed to construct WNP-1 and has assigned 100% of the capability of the facility to BPA. BPA is accorded substantial oversight responsibility and contract approval authority. In addition, the issuance of all bonds is subject to approval by BPA. Because the construction of WNP-1 is financed entirely through the sale of bonds, Applicant asserts that BPA controls the pace of construction as a result of its authority to withhold approval for bond sales.

As Applicant further describes the situation, in April of 1982 BPA published a draft powerload forecast which indicated that WNP-1, WNP-2 and WNP-3 were needed in the region, but that short-term surpluses of electricity could occur prior to 1990. Therefore, BPA recommended that construction of WNP-2 and WNP-3 proceed at full pace while the completion schedule for WNP-1 be delayed for a period of up to 5 years. Applicant developed alternatives to the BPA recommendation, but BPA advised Applicant that none of these alternatives was acceptable; that the BPA recommendation was the *only* prudent course of future conduct; and that BPA would not approve any financing plan inconsistent with its recommendation. As a result, Applicant decided to defer the construction of WNP-1, recognizing that BPA would not permit the sale of bonds needed to continue construction of the facility.

In support of its motion for summary disposition in Applicant's favor, Staff also relies upon BPA's refusal to approve further bond issuances for continued construction of WNP-1 as "good cause" for deferring construction. Staff agrees that Applicant would lack the financial resources to complete construction without BPA's support. Staff also relies upon one of the reasons cited by BPA for recommending deferral of WNP-1, a slower growth rate of electrical power demand than originally projected, as constituting a valid purpose for deferring construction. NRC Staff Motion at 5.

Intervenor, on the other hand, concludes that Applicant, rather than BPA, was responsible for the deferral of WNP-1. Intervenor submits that Applicant requested the deferral from BPA and concurred in it. Rosolie Affidavit at 2; Intervenor's Answer to Summary Disposition

Motions at 6-7. Intervenor asserts that Applicant had options other than deferral: it could have placed the project in indefinite mothball as it did with Projects WNP-4 and WNP-5; it could have terminated the Projects; or it could have entered into negotiations with the private utilities owning 30% of WNP-3 in order to have them defer WNP-3, rather than having to defer WNP-1. *Id.* at 7-9.

Furthermore, Intervenor asserts that the Board should go behind the decision to halt construction (whether made by Applicant or BPA) to consider the reasons for not financing continued construction at this point. Intervenor asserts that the reasons given by Staff and Applicant as inducing the BPA decision (which Intervenor asserts was Applicant's decision) to defer construction, a temporary lack of financial resources and a slower growth rate of electrical demand, are not the full story. It contends that escalating rates caused by the WPPSS construction program was a significant factor; that the private utilities would not agree to deferring WNP-3 in lieu of WNP-1, although WNP-1 was more complete; that more recent analyses by BPA show electrical growth to be even less than projected; and that there may be no future financing available to resume construction. Intervenor would like to call an expert witness to support its position that there will be a lack of need for power from WNP-1 (in addition to a consequent lack of future financing) in order to support a finding that there is no good cause to *extend* the construction completion date, notwithstanding that there might have been good cause to *delay* construction. In other words, whatever causes exist to delay construction, such as currently low electrical demand and temporary lack of financing, are more extreme, namely, even lower electrical demand and a permanent lack of financing, so as to require cancellation of construction. *Id.* at 10-11.

Intervenor also contends that the requested extension of completion date is not for a reasonable period of time by dint of its being insufficient. According to Intervenor, BPA and Applicant may well be considering a 5–12-year deferral of WNP-1, not a 2–5-year deferral, according to other documents. Furthermore, because of the downward trend in forecasting electrical demand and the unavailability of financing within the time period requested, Intervenor contends that Applicant cannot meet its burden of proving that financing will exist to resume construction within the 5-year period requested. Intervenor's Answer at 12-16. Finally, Intervenor asserts that the safety and environmental significance of the requested delay must be considered for at least the reasons that there is some concern over equipment deterioration during the extensive delay in completion of construction and that the original cost-benefit analysis at the construction permit stage is completely outdated. *Id.* at 16-19.

## III. OPINION

### A. Good Cause

Section 185 of the Atomic Energy Act, as amended, 42 U.S.C. § 2235, states, in pertinent part:

> All applicants for licenses to construct or modify production or utilization facilities shall, if the application is otherwise acceptable to the Commission, be initially granted a construction permit. The construction permit shall state the earliest and latest dates for the completion of the construction or modification. Unless the construction or modification of the facility is completed by the completion date, the construction permit shall expire, and all rights thereunder be forfeited, unless upon good cause shown, the Commission extends the completion date.

In furtherance of this section, 10 C.F.R. § 50.55 reads in pertinent part, as follows:

> (a) The permit shall state the earliest and latest dates for completion of the construction or modification.
>
> (b) If the proposed construction or modification of the facility is not completed by the latest completion date, the permit shall expire and all rights thereunder shall be forfeited: *Provided, however,* That upon good cause shown the Commission will extend the completion date for a reasonable period of time. The Commission will recognize, among other things, developmental problems attributable to the experimental nature of the facility or fire, flood, explosion, strike, sabotage, domestic violence, enemy action, an act of the elements, and other acts beyond the control of the permit holder, as a basis for extending the completion date.

In its guidance to this Licensing Board in CLI-82-29, *supra,* the Commission interpreted the foregoing statute and regulation as affording only a narrow scope to this proceeding within which Intervenor was free to prove only that "WPPSS was both responsible for the delays and that the delays were dilatory and thus without 'good cause'." 16 NRC at 1231. In *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), ALAB-722, 17 NRC 546 (1983), involving only WNP-2, the Appeal Board elaborated on those directions from the Commission to the Licensing Board. It interpreted "dilatory conduct in the sense used by the Commission" as meaning "intentional delay of construction without a valid purpose." *Id.* at 552. Consequently, it held that, "unless the applicant was responsible for the delays and acted in a dilatory manner (*i.e.,* intentionally and without a valid purpose), a contested construction permit extension proceeding is not to be undertaken at all." *Id.* at 553. Since, with regard to WNP-2 there had not been any Intervenor allegation of *intentional* delay (Applicant sought no halt in construction, as here, but had only suffered involuntary delays in meeting its construction schedule), the Appeal Board affirmed the Licensing Board's dismissal of Intervenor's contentions.

In the instant case, Applicant has made a strong showing of not "intentionally" causing the halt in construction, with affidavit and documentary support of its position that the Bonneville Power Administration caused the delay by withholding its approval of bond issuances for further construction, the only avenue for financing available to Applicant. Intervenor makes no attempt to dispute BPA's power to control the pace of construction through its control over the financing of the project, but insists that it was Applicant, rather than BPA, who instigated the decision to defer construction and that BPA only concurred in it. Intervenor seeks the opportunity to prove that Applicant's decision to delay construction, not having been compelled by BPA, was also without a valid purpose.

Although we see little in Intervenor's transmittals to us in opposition to the motions for summary disposition to support its position that the recommendation of deferral was instigated by Applicant, rather than BPA, we would not grant the motions for summary disposition on that score. Corporate dealings and motivations are sufficiently arcane, notwithstanding the matters placed upon the public record in the form of corporate minutes, resolutions, and recommendations, to afford a litigant the right to go behind these records to seek the testimony of participants in the corporate transactions. Intervenor has not taken discovery depositions, possibly for lack of finances, but that would not preclude it from examining for the first time at an evidentiary hearing the appropriate officials of WPPSS and BPA to identify the actual decisionmaker. However, even if we could place the intention to delay on Applicant, rather than BPA, we would still have to hold for Applicant on the undisputed material facts relating to the purpose for the delay, on which we find very little disagreement among the parties.

Without dispute, what prompted the decision to delay construction was a lack of financial resources to complete the construction of WNP-1 and WNP-3, and the forecast of no electrical demand for the output of WNP-1, at the targeted completion date of July 1, 1986. Intervenor, in fact, posits that the situation is more precarious than given by Applicant — that there will be a lack of financing and a lack of demand for electrical power even after a 5-year hiatus in construction. Intervenor's Answer at 10-11, 14-16; Rosolie Affidavit at 3-4.

In ALAB-722, *supra*, the Appeal Board indicated that "an intentional slowing of construction because of a temporary lack of financial resources or a slower growth rate of electric power than had been originally projected would constitute delay for a valid business purpose." 17 NRC at 552 n.6. Since there is no dispute that the lack of financing and slower growth rate of electrical power caused the decision to defer construction, we should have little hesitation in deciding that Applicant's delay in construction met the Appeal Board's test of being for a valid business purpose. Intervenor, however, relies on further *dictum* in ALAB-722 (*id.* at 553) that the "ultimate 'good cause' determination is expected to encompass a judgment about why the plant should be completed and is not to rest solely upon a judgment as to the Applicant's fault for delay." Intervenor asserts that there is not merely a *temporary* lack of financial resources, but a permanent one, and a long-term lack of electrical demand that would negate any reasons for completing a plant. Intervenor's Answer at 10-11.

Intervenor's argument flies in the face of the Commission's directives to us in CLI-82-29, *supra*. There the Commission, in no uncertain terms, focused exclusively on the "reasons that have contributed to the delay in construction," rather than good cause for completing construction. 16 NRC at 1228; *see also id.* at 1229, 1230 and 1231. While ALAB-722, *supra*, appears to be at some variance with the Commission's directives to us to focus exclusively on causes for delay, rather than for completing construction, even that *dictum* would require a judgment about whether the plant should be completed only if Applicant has not first satisfied the test of either not being responsible for the delay or having delayed construction for a valid purpose. Since the Applicant, in this case, has halted construction, either intentionally or at the direction of BPA, for the valid reasons of a lack of financial resources and a slower growth of electric power, we need not reach a value judgment on the advisability of completing the plant.*

Intervenor also seeks a hearing on the other options it asserts were available to Applicant in place of its deferral of construction for the

*The Appeal Board has not illuminated the basis for its focus on the future, rather than on Applicant's past conduct, seemingly at variance with the Commission's directives to us, other than to conclude that this is called for by Section 185 of the Atomic Energy Act. 17 NRC at 553. Consequently, we offer no opinion on why the Appeal Board would permit an inquiry into the advisability of building a plant when it is for the benefit of an applicant that has failed the Commission's test of not being dilatory but would not permit such inquiry for the benefit of an intervenor wishing to scrap the plant. An applicant for a construction permit extension has, presumably, already satisfied its requirement of demonstrating the need for power at the construction permit stage and should not have to demonstrate that need again unless, under special circumstances, such a demonstration is deemed necessary *at the operating license stage. See* 10 C.F.R. §§ 51.21 and 51.23(e), and Statement of Consideration, 47 Fed. Reg. 12,940 (1982).

5-year projected period. These other asserted options of placing the project in indefinite mothball, terminating the project or negotiating with private utilities who own 30% of WNP-3 to delay WNP-3 instead, might have been more "prudent" according to Intervenor. Rosolie Affidavit at 2-3; Intervenor's Answer at 9. Nothing stated by Intervenor in its answer or submitted in support of it raises any question about the decision to delay construction being at least a rational business decision, albeit not the decision Intervenor might have made under the same circumstances.

We see no merit in the Board's seeking to substitute its own judgment for that of Applicant in selecting one of a number of rational alternatives available to Applicant. The one apparently favored by Intervenor (*ibid.*), of halting construction on WNP-3 rather than WNP-1, cannot support a denial of the requested extension. If the Applicant is attempting to salvage both nuclear plants by temporarily halting construction on one of them, that cessation of construction activities has a valid purpose regardless of which plant is chosen. We see no reason to attempt to force the cancellation of the plant chosen to be delayed (through a revocation of the construction permit) merely because some reasonable persons would have chosen to delay the other plant. Nor do we see any justification for the Board to question the reasonableness of Applicant's decision of deferral because Applicant did not choose, instead, either of the other two more extreme alternatives suggested by Intervenor of indefinite mothballing or termination.

We are not faced with an allegation that Applicant has actually decided to abandon the plant. Had Intervenor made such an allegation and offered some factual support for it we would not be so quick to grant summary disposition in favor of Applicant. A finding by us of abandonment might permit us to dismiss Applicant's application as being moot. *See Puerto Rico Electric Power Authority* (North Coast Nuclear Plant, Unit 1), ALAB-605, 12 NRC 153 (1980). Here, Intervenor has not gone beyond an attempt to prove that future power demands and lack of financing will cause an abandonment of the plant when Applicant is faced with resuming construction. If Intervenor were convinced that Applicant had irrevocably decided to abandon the plant, it is doubtful that it would continue to expend its resources on its interventions in this and the operating license proceedings.

## B. Reasonable Period of Time

Intervenor also challenges the reasonableness of the period of time requested for the extension. Intervenor asserts that the 5-year requested extension is unreasonable because it is insufficient. It would like the opportunity to prove that the plant could not be completed by 1991. Intervenor's Answer at 11-16.

We cannot fairly read into the Atomic Energy Act or the regulations thereunder any basis for challenging the reasonableness of the period of requested extension on grounds of insufficiency. Were there some overall time (rather than reasonableness) limitation on the total construction period or on the period that might be requested which Applicant is attempting to circumvent by requesting the needed time in increments, we might be persuaded otherwise. However, no such limitation is apparent to us. By requesting an insufficient period, Applicant could only injure itself because it would then be forced to apply for another extension and demonstrate good cause anew in order to complete the plant, when its original "good cause" demonstration could have supported an extension for the total period required.

Perhaps we would view differently Intervenor's arguments with regard to the insufficiency of the period requested if we could accept its further argument that the total period of extension must be examined with regard to the safety and environmental aspects of the deferral of construction. Indeed, Intervenor's argument that there may be equipment deterioration during a lengthy delay in construction that should be considered during a construction completion date extension proceeding (Intervenor's Answer at 17) has considerable superficial appeal. Certainly, one cannot easily disassociate the question of whether an extension should be granted from the realization that the granting of the extension might well lead to a deterioration in equipment. Similarly, one could postulate environmental effects from the prolongation of the construction period. However, were we to choose the most propitious moment for evaluating the effects of a prolonged or delayed construction period on safety and the environment, we would choose a time *after* the effects became apparent, namely, at the operating license stage. A hearing at this juncture would be mostly speculative. We note that the Licensing Board in the WNP-1-OL operating license proceeding, composed of the same members as here, has admitted a contention (Contention 20) that questions unnamed construction defects that might result from Applicant's method of preserving the construction during the period of deferral. *Washington Public Power Supply System* (WPPSS Nuclear Project No. 1), LBP-83-66, 18 NRC 780, 797-98 (1983).

A deferral of consideration of the safety and environmental effects of the delay in construction to the operating license stage not only makes the most sense, but it comports with the Commission's interpretation of

Section 185 of the Atomic Energy Act as not requiring the relitigation of health, safety or environmental questions between the time a construction permit is granted and the time the facility is seeking authorization to operate. CLI-82-29, *supra,* 16 NRC at 1228. And, since the health, safety and environmental effects of the prolonged construction are not to be questioned at this juncture, Applicant also can derive little benefit from understating the period needed for completion of construction, as alleged by Intervenor.

**C. Legal Standard**

Under 10 C.F.R. § 2.749, this proceeding should be dismissed if the filings indicate that there is no genuine issue as to any material fact. In deciding Applicant's and NRC Staff's motions for summary disposition we have construed all of the material facts in favor of Intervenor. We have assumed, notwithstanding the strong evidence offered to the contrary by Applicant, that the decision to halt construction was Applicant's, not BPA's. We have accepted Intervenor's assertions that there were more prudent alternatives to a temporary halt in construction, such as cancellation of the facility, placing it in mothball, or halting construction on WNP-1. We have also assumed for the purpose of deciding this motion that the period of extension requested isn't sufficient and that the economic situation will eventually cause an abandonment of the facility. We nevertheless reach the position that Applicant has demonstrated good cause for delaying construction by demonstrating valid reasons for doing so even though there may be more prudent alternatives and the option selected may prove fruitless. Having found good cause for the deferral of construction on the uncontroverted material facts, we must grant Applicant's and Staff's motions for summary disposition without inquiring further into the advisability of constructing the nuclear plant.

## Order

For all of the foregoing reasons and based upon a consideration of the entire record in this matter, it is, this 1st day of February 1984,

ORDERED

That Applicant's and NRC Staff's motions for summary disposition in favor of Applicant are granted and Intervenor's sole contention is dismissed, terminating the proceeding.

Within ten (10) days after service of this Memorandum and Order, which constitutes a final disposition of this proceeding before the Licensing Board, Intervenor may take an appeal to the Appeal Board by filing a notice of appeal pursuant to 10 C.F.R. §§ 2.762 and 2.785. A supporting brief would then be due within thirty (30) days after the notice of appeal is filed.

Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this Memorandum and Order will constitute the final decision of the Commission thirty (30) days from the date of issuance unless an appeal is taken in accordance with 10 C.F.R. § 2.762 or the Commission directs otherwise. *See also* 10 C.F.R. §§ 2.785 and 2.786.

THE ATOMIC SAFETY AND
LICENSING BOARD

Glenn O. Bright
ADMINISTRATIVE JUDGE

Jerry Harbour
ADMINISTRATIVE JUDGE

Herbert Grossman, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
February 1, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Peter B. Bloch, Chairman**
**Dr. Kenneth A. McCollom**
**Dr. Walter H. Jordan**

**In the Matter of**

**Docket Nos. 50-445**
**50-446**
**(Application for Operating License)**

**TEXAS UTILITIES ELECTRIC COMPANY,* *et al.***
**(Comanche Peak Steam Electric Station, Units 1 and 2)**

**February 8, 1984**

Based on a review of the history of the case, the Licensing Board concludes that Applicant had a fair opportunity to prove its case concerning quality assurance for design and that there is no reason to correct a previous decision to clarify that the Board's conclusions were based on the record.

## QUALITY ASSURANCE FOR DESIGN: APPENDIX B

Criterion XVI of Appendix B to Part 50 requires the prompt identification of design deficiencies, but it does not require that those deficiencies be called "nonconformances." No particular terminology is mandated.

*Previously, Texas Utilities Generating Company.

## QUALITY ASSURANCE: RELATIONSHIP TO § 50.55(e)

Criterion XVI of Appendix B to Part 50 is consonant with 10 C.F.R. § 50.55(e). The former requires a system for promptly identifying deficiencies, including design deficiencies. The latter requires the prompt reporting to the NRC of serious deficiencies.

## RULES OF PRACTICE: NEW ARGUMENTS

Absent some special procedural consideration, proposed findings of fact may make new arguments about record evidence. Allegedly contrary precedent is not persuasive.

## RULES OF PRACTICE: MOTION FOR RECONSIDERATION (NEW ARGUMENTS)

Motions for reconsideration are for the purpose of pointing out an error the Board has made. Unless the Board has relied on an unexpected ground, new factual evidence and new arguments are not relevant in such a motion.

## RULES OF PRACTICE: STANDARDS FOR APPLICANT TO REOPEN THE RECORD

Applicant is not subject to the same standards for reopening the record as are intervenors. It is neither logical nor proper to close down a multi-billion-dollar nuclear plant because of a deficiency of proof. However, repeated failures of proof would jeopardize intervenor's right to due process and would require the denial of a license.

## TECHNICAL ISSUES DISCUSSED

Pipe support stability
U-bolts cinched up around pipes
U-bolts made of SA-36 steel, clamping force
Local pipe stresses from pipe supports
U-bolts, overtensioning
Relationship of ASME Code and AWS Code, pipe supports
Richmond Inserts, axial torsion.

# MEMORANDUM AND ORDER
## (Reconsideration Concerning Quality Assurance for Design)

This Memorandum and Order considers the motions of the parties to reconsider our previous Memorandum and Order (Quality Assurance for Design).[1]

## I. APPLICANT'S RECONSIDERATION MOTION

Applicant requests that we revise our Design Decision so that we make it clear that

> the *evidentiary record* is presently not adequate to determine whether Applicant's pipe support design process satisfies Appendix B (a view which Applicant shares) and that further evidence will be required.[2]

Applicant's concern arises because it feels that it did not have adequate notice that this matter was being litigated and because we incorrectly interpreted Applicant's Findings. We disagree.

First, we note that our findings were explicitly related to the burden of proof as reflected in our record. We acknowledged our lack of confidence that our record reflected the real world; hence, we permitted Applicant to submit a plan to increase this Board's confidence in the plant's design. Thus, Applicant will have an opportunity to demonstrate its compliance with the requirements of Criterion XVI of Appendix B.[3] However, Applicant had not given us any basis for hedging our findings further. Our knowledge is limited to the evidentiary record, which is the basis for our findings, and we are required to make findings based on that record.[4] We have done so.

If Applicant did not have a fair opportunity to demonstrate the adequacy of its quality assurance program, then we might agree to hedge the language we use in finding a deficiency. However, Applicant had an

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[1] We adopt the terms we defined in our previous decision, LBP-83-81, 18 NRC 1410 (1983) (Design Decision). All three motions for reconsideration were filed on January 17, 1983 and shall be referred to as Applicant's Reconsideration, Staff's Reconsideration, and CASE's Reconsideration.

[2] Applicant's Reconsideration at 2.

[3] Although our language suggested that Applicant and Staff thought Appendix B did not apply to design, a more accurate statement of the position of these parties was that Criterion XVI of Appendix B, which requires the identification and prompt correction of deficiencies, did not apply to design. Our decision could have been clearer on this point, but we believe the discussion in the decision adequately stated our concerns.

[4] *See Commonwealth Edison Co.* (Byron Nuclear Power Station, Units 1 and 2), LBP-84-2, 19 NRC 36 (1984) (operating license denied based on the existing record).

abundance of opportunities to present its case and did not avail itself of them.

## A. Relevant Background

Our Design Decision sets forth the history of the Walsh/Doyle contention, but pertinent parts need be repeated to place Applicant's current claim in perspective. Based on testimony from Walsh and Doyle, CASE has argued that there were deficiencies in several design documents. CASE also argued that Applicant had not completed nonconformance reports related to design documents and that it had not filed 10 C.F.R. § 50.55(e) reports of significant design deficiencies.

Applicant answered that the deficiencies found by CASE were in preliminary design documents and were of no significance because they would be corrected before the plant was completed. It also argued that Appendix B did not require it to complete nonconformance reports for design deficiencies.

In another matter, which seemed unconnected to this question, Applicant has even argued that Appendix B does not require that reports called "nonconformance reports" or "NCRs" need be completed for *construction* deficiencies. This argument apparently is correct with respect to *all* deficiencies (including construction and design) because Appendix B, Criterion XVI, provides substantive criteria for identifying and correcting deficiencies but does not mandate any particular label for the reports concerning those deficiencies.

## B. Applicant's Initial Argument

With these contextual matters in mind, let us now set out in full the portion of Applicant's Findings that it would now have us interpret as arguing only that Appendix B does not require that any particular *label* be attached to nonconformances:

> *6. Documentation of "Nonconformances"*
>
> With respect to the allegation that Nonconformance Reports ("NCRs") should have been written against pipe support designs which were found to be inadequate, the NRC Staff testified, and the Board agrees, that 10 C.F.R. Part 50, Appendix B does not address inadequate designs but rather addresses the conformance of installed hardware and the inspection thereof to the design. With respect to 10 C.F.R. Part 50, Appendix B, Criterion III, concerning design control, that provision establishes review procedures, and does not involve reporting of nonconformances. (Tr. 6707-10.) Accordingly, we find there is no requirement for the identification of inadequate pipe support designs as nonconforming conditions. The iterative design process for pipe supports (including the internal checks in that process) discussed herein

*supra*, Section II.C.1, assures that inadequate designs or unstable supports are identified and corrected.[5]

## C. Analysis

We do not find this language to be consistent with Applicant's regulatory obligation. We consider the following language in Applicant's Findings to be clear: "Appendix B does not address inadequate designs but rather addresses the conformance of installed hardware and the inspection thereof to the design." The meaning of this passage, that the prompt identification of design deficiencies is not required by Appendix B, was echoed by a statement that "there is no requirement for the identification of inadequate pipe support designs as nonconforming conditions." This language concerns *requirements*, does not place "nonconforming" in quotes and is, simply, an unqualified statement that Criterion XVI is inapplicable.

Our conclusion that Applicant has not interpreted Appendix B, Criterion XVI, correctly in this proceeding also is related to the general conduct of the case. CASE has attempted to show deficiencies in particular design documents. Instead of demonstrating the existence of a system to identify and correct deficiencies, Applicant chose to show that:

> the designs raised by [CASE's] ... witnesses were taken from the initial stages of a carefully designed and comprehensive iterative design process and thus do not (nor were they intended to) reflect the quality of the final pipe support designs at Comanche Peak.[6]

We do not consider this to be isolated language. It represents Applicant's litigation approach, in which the Staff concurred. There has been no recognition that errors in design documents are an independent concern, regardless of whether they may be corrected before the plant is completed. Each design document must be a quality document. Although errors may be made, significant errors — particularly errors of which Applicant has been made aware through employee concerns and litigation — should be promptly identified, "documented," and corrected with reasonable speed.

We understand that Applicant now contends that it has such a system. However, the adequacy of this system for documenting and correcting design deficiencies (and construction deficiencies resulting from the

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[5] Applicant's Findings at 27-28.

[6] *Id.* at 18.

implementation of deficient designs[7]) has not yet been demonstrated[8] and CASE will have an opportunity to litigate both the adequacy of the system and the adequacy of its implementation.[9]

It also is not clear how Applicant's design program complies with the requirement of Criterion I that "persons . . . performing quality assurance functions [for design] . . . report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided."

It is difficult for us to sympathize with Applicant's surprise that its compliance with Appendix B was being litigated. The contention being litigated is a quality assurance contention and the Walsh/Doyle design concerns were admitted as a portion of that broad contention. Furthermore, these specific concerns about quality assurance for design were covered by Chapter XXV in CASE's Findings and Applicant had an opportunity to respond to those findings.

During the May 1983 hearings, both the Board and CASE asked questions concerning the Special Inspection Team's conclusion that Applicant would correct deficiencies before the plant was completed. Furthermore, as the Design Decision states, questions concerning the reporting of nonconformances were addressed in September 1982. At that time, Applicant did not argue that things labeled "nonconformance reports" were not required for design. It argued that, "[t]he item under consideration during design where you are going through an iterative process is not a nonconformance until you complete the design."[10] Furthermore,

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[7] *See* CASE's Answer to Motions for Reconsideration, February 1, 1984 (CASE's Answer) at 12, arguing that Criterion XVI of Appendix B requires a system for reporting all construction deficiencies, including those caused by faithful adherence to a deficient design.

[8] Criterion XVI requires that conditions adverse to quality be promptly identified and corrected. *Compare* Applicants' Reconsideration at 19, "*significant* conditions adverse to quality are identified." [Emphasis added.]

Although Criterion XVI restricts the requirements to identify the *cause* of a condition and to document that condition to *significant* deficiencies, the requirement to *identify* conditions is not restricted by use of the adjective, "significant." We anticipate receiving evidence concerning how Applicant's system actually handles specific deficiencies that have been detected.

[9] *See* CASE's Answer at 17-19. In this regard, we recognize the prodigious effort put in by CASE's unpaid volunteers, but we urge it to assist the Board (and the other parties) in being able to determine which aspects of prior exhibits bear on any new arguments presented by Applicant. In particular, we require CASE to make a good faith effort to see that new filings be susceptible of being understood without numerous cross-references. The cross-references are necessary to document what is in the record, but the Board and parties cannot readily undertake extensive tours through already-filed documents without an explanation in a filed pleading of what CASE believes those documents stand for.

We also note that the Board appears to be more ready to admit its mistakes than are the parties. We encourage others to be more ready to admit their mistakes and to concede points erroneously decided in their favor.

[10] Reedy, Tr. 5185. Finneran agreed that "until design of the installation is complete, there is no nonconformance condition." Tr. 5186. *See also* Taylor (Staff's Resident Inspector for Construction), Tr.

*(Continued)*

immediately after this question on nonconformances, Applicant elicited information from its witnesses concerning the use of CMCs, which it apparently considered to be a related question. At that time, Mr. Finneran's testimony about CMCs did not include the reporting of design deficiencies as a purpose of that document.[11]

Although Applicant will be permitted to show that it has an adequate quality assurance system for design, we do not consider it appropriate to modify any of our conclusions on this matter. Our conclusions fairly represent the state of the record.

## D. Applicant's Argument About § 50.55(e)

We found that the "need for prompt identification of deficiencies [pursuant to Appendix B, Criterion XVI] is consistent with 10 C.F.R. § 50.55(e)(1)" and that fulfillment of the § 50.55(e)(1) requirement to report significant deficiencies requires that the "ongoing quality assurance program for design . . . have the capacity to spot, track and resolve significant design deficiencies on an ongoing basis."[12] Applicant asks us to reconsider this position and to state that § 50.55(e) "does not impose any requirements concerning the timing of activities under Appendix B."[13] This we refuse to do. We have merely interpreted two sections of the regulations to be consonant with one another, a standard method of regulatory interpretation. The requirement for the "prompt" detection of deficiencies in Appendix B assures that significant deficiencies should be promptly detected and reported pursuant to § 50.55(e). We fail to understand what other position Applicant would have us adopt.

## E. New Arguments

Applicant would have us rule that new arguments presented in Findings are to be disregarded. However, its basis for this argument rests on two flimsy legs: (1) that it is a basic characteristic of administrative procedure that a party have the opportunity to know and meet the argument of the other party,[14] and (2) that *Union Electric Co.* (Callaway Plant, Unit 1), ALAB-740, 18 NRC 343, 350 (1983), contains language

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6707: "Appendix B, in dealing with nonconforming conditions, does not address nonconforming design. It only addresses the conformance of the installed hardware and the inspection thereof to the design."

[11] Tr. 5185-86.

[12] Design Decision, 18 NRC at 1414. *See* Applicant's Reconsideration at 9.

[13] Applicant's Reconsideration at 9.

[14] Applicant cites K. Davis, Administrative Law Treatise § 7.01 (1958). Applicant's Reconsideration at 17.

suggesting the need to present "an analytical disagreement" to an opposing witness for his consideration.

The first leg of this argument presents a truism that is inapplicable in this proceeding. Applicant has had an opportunity to learn about CASE's allegations through discovery. It could have asked for a prehearing conference to discuss in advance the parties' positions. It could have asked for the advance filing of findings (as we have ordered for subsequent hearings) or a trial brief. It did have an opportunity to file a reply to CASE's Findings. And if new arguments were made that required additional evidence, it could have moved to reopen the record for that purpose. We conclude that Applicant had an ample opportunity to know and respond to CASE's arguments.

As to the second leg of the argument, we find little factual support for the proposition that Applicant was prejudiced in any way by late-filed arguments and we do not interpret the *Callaway* case as barring new arguments in an intervenor's proposed findings.

Applicant has the following introductory remarks to make about its position that new arguments are barred:

> The reason to foreclose new arguments is that Applicant was not afforded the opportunity to meet the new argument with responsive evidence or cross-examination. In addition, we have identified below three instances in which the Board clearly relied on new arguments in reaching its conclusions.[15] Because the Board also relies on record material in deciding these questions, however, we do not ask the Board to reverse its conclusions but to revise them to reflect that a decision on these questions would be premature without affording Applicant an opportunity to respond.

In this passage, Applicant clarifies its position on new arguments. It does not claim prejudice from any arguments made by CASE in CASE's findings. Its sole claim to prejudice is that it was not permitted to respond to arguments made by CASE in its reply to the affidavits filed by the Staff concerning open items left after our hearing session.

We note that both of the arguments to which Applicant alleges that it had no opportunity to respond were, as Applicant admits, based on record evidence. The arguments were clearly set forth by witnesses. They related to open items that were addressed by Applicant in its findings. Furthermore, Applicant was under a clear directive by this Board to address all (potentially significant) evidence, including adverse evidence, relevant to its proposed findings and conclusions. Applicant had both the opportunity and the obligation to explain the relevance of

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[15] [Footnote from Applicant's Findings:] *See* Memorandum and Order at 13, n.35 [18 NRC 1419 n.35] (torsional moments in Richmond Inserts and shield wall thickness near the upper lateral restraint).

the underlying evidence when it filed its findings. There was no lack of opportunity there.

In Applicant's Reconsideration, there was another opportunity to file arguments concerning these matters. With respect to axial torsion in the Richmond Inserts, Applicant eschewed this opportunity to present any new arguments. It merely states that it has "obtained the independent opinions of outside experts on this point" and it asks the Board to reconsider the record on this matter.[16] With respect to the thickness of the wall near the upper lateral restraint, Applicant does not make any arguments at this time either. We find no prejudice to Applicant from this alleged lack of an opportunity to respond.

Nor do we find the citation of the *Callaway* case to be persuasive. In that case, the Appeal Board was considering an argument made in an intervenor's proposed findings, based on a citation to extra-record scientific material that could have been officially-noticed.[17] We applied the *Callaway* principle by refusing to rely on similar citations to scientific material in this case.[18] The Appeal Board's language in *Callaway* related to a situation in which intervenors had presented no witnesses and had not even conducted cross-examination. *Callaway* does not support Applicant's argument: that we must refuse to consider new arguments concerning evidence that is already in the record.

We do not change our conclusion that, absent some procedural consideration not present in this case, proposed findings may make new arguments about record evidence.

## F. Specific Factual Findings

Applicant's Reconsideration requests changes in factual findings, based at times on entirely new arguments and evidence concerning matters that have been litigated. This is not proper in a Motion for Reconsideration, which is an extraordinary filing alleging error in a decision of the Board. A motion for reconsideration should not include new arguments or evidence unless a party demonstrates that its new material

[16] Applicant's Reconsideration at 41.

[17] *Callaway*, 18 NRC at 349-50.

[18] LBP-83-55, 18 NRC 415 (1983). We note that the Staff "supports" Applicant's argument concerning *Callaway* but that its argument agrees with our interpretation of that case. NRC Staff Response to Applicant's Reconsideration, January 27, 1984, at 7. We note, additionally, that the Staff does not demonstrate prejudice to Applicant resulting from our interpretation. (*See also Tennessee Valley Authority* (Hartsville Nuclear Plant, Units 1A, 2A, 1B and 2B), ALAB-463, 7 NRC 341, 352 (1978), in which late-filed documents were considered by the Appeal Board because of their possible importance to public health.)

relates to a Board concern that could not reasonably have been anticipated.

Although we need not address in this decision any improper new arguments or evidence, we have chosen to address some of those arguments in order to communicate the Board's understanding of these matters and to facilitate the efficient progress of this case. There will be time to address these arguments more fully after new evidence is taken with respect to the Plan the Applicant is filing at the Board's request.

### *1. Mr. Michael A. Vivirito*

Applicant requests that we revise our decision to be less critical of Mr. Michael Vivirito than we were in the Design Decision, 18 NRC at 1420 n.37. After reading Applicant's comments and reviewing our decision, we conclude that some softening of our language is appropriate.

Mr. Vivirito was in many ways an impressive witness, with good control of technical matters and an ability to explain complex matters to us in a way that we could understand. His testimony concerning thermal expansion was particularly helpful to us.

Our concern with Mr. Vivirito's testimony is that he seemed at times to be too ready to dismiss matters as falling within his engineering judgment, without providing the Board an adequate explanation. He also presented to us some testimony that, while carefully described as "only background," nevertheless could have implied to Gibbs & Hill employees that Mr. Vivirito has some feeling that regulatory requirements for seismic analysis are unnecessarily strict. Since we are aware of the importance of compliance with regulatory criteria and of the tendency of the industry to feel that it is over-regulated, we became uncomfortable with the statement Mr. Vivirito made to us. The statement bore the possible meaning that Mr. Vivirito did not feel that rigorous compliance with seismic requirements was necessary to the safety of the plant and we were concerned that members of his organization could adopt this attitude, apparently held by a senior official of their company.

Although we continue to be sensitive to this issue, we think we were overly critical of an isolated comment made in one portion of a lengthy regulatory proceeding. We do not have reason to believe that this single passage of testimony reflects an attitude that prevails at Gibbs & Hill. We expect that Mr. Vivirito's sincere efforts to listen to the Board's concerns and to assist us in our decision process is more reflective of Gibbs & Hill's attitudes toward regulation than was this one remark. We apologize for making too much of this one statement.

We are hopeful that this discussion will clarify the nature of our concerns. Accordingly, we will delete Footnote 37 from our published decision.

## 2. *Mr. Kerlin*

Applicant relies on extra-record materials to rebut the Board's finding that Mr. Kerlin had some supervisory responsibility. Although Applicant has not yet presented evidence on this point, we are confident that it will do so in order to establish its point. Similarly, Applicant has pointed out to us that we mistakenly attributed an incident that occurred at the Fast Flux Test Facility to the Comanche Peak facility.[19]

These factual errors occurred in a portion of our decision where we were trying to ascertain the first date on which Applicant was aware of possible instability problems. The result of this change of facts is, however, inconsequential. Our current best information about the first date of Applicant's knowledge is some time in 1981.[20] Since there are no data in our record concerning how Applicant dealt with this deficiency,[21] and since the burden of proof is on the Applicant, we have no basis for concluding that it was handled in a reasonably prompt manner. We will have to await further evidence to reach a conclusion on the adequacy of Applicant's system for promptly resolving design deficiencies.

In deliberating about this point, however, the Board has become aware that the entire matter of instability has been handled in an incomplete manner in our record. There are abstract discussions of the nature of pipe support instability, including hard-to-understand descriptions of a model that is not in our record, discussions about a pen standing on end, language about instabilities that exist only when a pipe is "missing" and other abstract discussions. What is needed is a review of a detailed, worst-case sample of about five of the thirty cases of instability investigated by the Staff. Then the Board will become informed in detail of the relationship between the design process and the stability of pipe supports. Some of the relevant issues are: (1) whether the forces and moments indicated by the initial pipe run analysis were met by the pipe design groups at the node points to which these supports were attached, (2) whether all required static and dynamic forces were considered, (3)

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[19] Applicant's Reconsideration at 20-21.

[20] Design Decision, 18 NRC at 1425 n.57.

[21] Testimony of a Staff witness that the problem was "identified during the normal design review process" does not establish that the problem was identified and resolved with reasonable promptness, particularly in light of the Board's findings concerning the adequacy of cinching-up U-bolts to prevent rotation. *See* Applicant's Findings at 46.

the nature of the instability, including the conditions under which it would exist and the likelihood of those conditions occurring, (4) the extent to which Gibbs & Hill was provided with all the information about the performance of the support that they needed for the purpose of doing a revised pipe run analysis and a local pipe stress analysis, (5) the reason that these supports were unstable, (6) how Applicant identified these instabilities and the process by which it resolved (or is resolving) them, including the paper trail of that process, and (7) the potential safety significance of these deficiencies.

The Board acknowledges that its conclusions about the adequacy of Applicant's program to identify analogous problems or to promptly correct design deficiencies was a conclusion based on a record that may have been incomplete. The Design Decision should be interpreted to be consistent with this statement.[22]

### 3. *Walsh and Doyle*

Applicant states that it never intended to impugn the veracity of Walsh and Doyle[23] and has asked that we clarify that fact for our record, which we gladly do. When the Board stated that Applicant had used the limited role of the STRUDL Group to question the credibility of Mr. Walsh and Mr. Doyle we might have more correctly stated that they used the limited role of that group in the total design sequence as a way of arguing that the testimony was entitled to less weight.

We also accept Applicant's request for clarification concerning the breadth of knowledge of Mr. Walsh and Mr. Doyle. It accords with our understanding that

> Walsh and Doyle had a limited *vertical* view of the entire support design *process*, by virtue of the function of the group they worked in[, but] they had a broad *horizontal* view from which to observe a large number of support *designs* in the combined year and one-half they were employed in the STRUDL group.[24]

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[22] Our summary of our own conclusion was that Applicant has "not demonstrated the existence of a system that promptly corrects design deficiencies . . . ." Design Decision, 18 NRC at 1412. Our conclusions were based on the evidentiary record as it existed. *See also* Design Decision, 18 NRC at 1453 (acknowledging that further proof and analysis could cure the Board's difficulties).

[23] Applicant's Findings at 23-24. *Compare*, however, CASE's Answer in Doyle Deposition at 3. (We are not aware of the issue of the *Circuit Breaker* to which CASE refers, but we are confident that it will be brought to our attention when intimidation matters are litigated.)

[24] Applicant's Reconsideration at 26.

We do not think that Mr. Walsh or Mr. Doyle disagree with this characterization.[25]

On the other hand, we continue to believe that there would be serious repercussions for our confidence in the design of other portions of Comanche Peak were we to continue to be uncertain as to whether there were serious deficiencies in the design process for pipe supports or in specific designs for pipe supports.

### 4. *Specific Stability Questions*

Applicant urges that we reconsider our finding[26] in which we questioned whether the rate of unstable NPSI supports would be similar to the rate of unstable supports by the other two pipe support design groups. Applicant's request is based on the Affidavit of Mr. Finneran, dated June 3, 1983, and apparently not relied on either in Applicant's Findings or Applicant's Reply. Our review of that document, which was submitted to the Board at its request and should be considered to be in evidence, persuades us that the design review had progressed further than we had thought.[27] Consequently, if evidence persuades us of the adequacy of that review, including the appropriateness of Applicant's definition of instability (which has not been discussed) and the thoroughness of its survey examination, we will at that time accept its conclusion that only 21 of 13,681 supports, drawn from all design groups, were unstable.[28] Such a finding would, of course, go a long way toward giving the Board confidence in the stability of supports.

However, we decline to accept Applicant's suggestion that we may have inadvertently relied on a SIT Report discussion regarding "piping systems" in drawing conclusions about piping supports. The full quote from page 28 of the SIT Report is [emphasis added]:

> It is not general industry practice to explicitly address the overall stability of piping systems *together with their supports* in design guidelines. Rather, it is standard industry design practice to address only the structural integrity of supports in design guidelines. The Applicant's practice corresponds to this industry practice. Thus, no explicit design guidelines address overall stability. Functional adequacy, including

[25] It would appear to be time for the Staff and Applicant to confer in detail with Mr. Walsh and Mr. Doyle about *all* the deficiencies they allege. *See* CASE's Answer in Doyle Affidavit at 4 (there appear to be further problems that Mr. Doyle and Mr. Walsh have not yet brought up).

[26] Design Decision at 1426 n.68. Note that the reference should be to Applicant's Reply rather than to Applicant's Findings.

[27] We interpreted Applicant's citation of earlier testimony to have been a representation that the review had *not* progressed as far as it apparently has progressed. Applicant's Reply at 13 n.6, relied on earlier testimony and did not cite the Finneran Affidavit, which was filed 3 months previously.

[28] Finneran Affidavit at 3-5.

stability, of the overall piping system is typically a result of the normal iterative design and review process.

We relied on this passage for a finding that there were no design guidelines that address stability of pipe supports.[29] We do not understand how the iterative design process would substitute for such guidelines, although we may be persuaded of that through further proof. Furthermore, as we explained, we rejected the SIT's conclusions, found in the unquoted remainder of the paragraph we have cited, that stability problems may be avoided by cinching up U-bolts around pipes.

We note that this discussion appears within a section of the SIT Report devoted to "Stability of Pipe Supports Designed for CPSES."[30] Immediately following the paragraph we discussed above, there is a paragraph about the identification of unstable, nonrigid supports in Applicant's design process.[31] This discussion does not, however, track Applicant's review process from the time Applicant became aware of instability problems, probably because the SIT was not concerned about the question of whether or not deficiencies were being cured promptly.

A consequence of the SIT's approach, as explained in our record, is that the Board was left without a reasonable explanation of: (1) why design guidelines concerning stability were not necessary, and (2) whether design deficiencies are corrected promptly. Our conclusion is that this aspect of our decision is correct.

On another matter, we find that we properly construed the SIT Report's statement that "[d]esign modifications under *consideration* [emphasis added] by the Applicant are intended to prevent rotation of the box frame around the axis of the supported piping."[32] If the SIT meant to indicate that this problem had been resolved, the word "consideration" was ill-chosen. If the SIT would like to clarify its testimony or the Applicant would like to document its resolution of this problem, this aspect of our record might then be resolved to Applicant's satisfaction, but we do not think the SIT Report bears the meaning Applicant urges.[33]

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[29] Design Decision, 18 NRC at 1426.

[30] SIT Report at 27.

[31] *Id.* at 28.

[32] *Id.*

[33] Applicant's Reconsideration at 24.

### 5. *Friction on Pipes Attributable to U-Bolts*

We accept Applicant's clarification that it uses SA-36 steel in U-bolts, rather than the equivalent SA-307 steel we said it used.[34] However, we decline to rule on Applicant's new argument concerning the interpretation of ASME Code Section XVII-2462.1-31. In particular, we do not know whether the quoted section applies by analogy to the use of SA-36 steel to produce clamping forces that will restrain rotation of a pipe[35] and we have no evidence either about how great those clamping forces are or how great they need to be.

We do not consider it essential to our findings that Applicant may have initially designed its U-bolts to be cinched down. Although we consider the SIT Report, on rereading, to be somewhat ambiguous on this point, our finding on this subject merely helped us to feel that we understood how this possible problem of improper use of U-bolts arose. Should we be convinced that U-bolts were designed to be cinched down,[36] we would still need to be convinced that they exert sufficient clamping force to prevent rotation. If they do not exert sufficient force, the argument about the initial concept of U-bolts will only deprive us of an explanation that helped us to understand how this might have arisen. Applicant's argument does not persuade us that the U-bolts will exert sufficient force to restrain rotation.

In concluding our discussion of this point, we would note that the systematic discussion of instability which we have asked for, above, could help us to understand the nature of the stability problem and relate it to this question of clamping force. There is nothing in our record that quantifies in any way the amount of clamping force necessary to avert instability.

### 6. *Clamping Force*

The Board agrees with Applicant's statement that ASME Code Section XVII-2461.1-1 does not state that local stresses from SA-307 steel are too great, but we never gave that section that interpretation. The only purpose of our mention of this section in the context of local pipe

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[34] Applicant's Reconsideration at 28. However, the label attached to this steel does not seem to be significant since the different labels apparently refer to the same material applied to different uses. *See* CASE's Answer in Doyle Affidavit at 4.

[35] Although his statement is not yet in evidence, Mr. Doyle believes that ASME XVII-2462 applies and that Applicant is not in compliance with it. CASE's Answer in Doyle Affidavit at 4. This matter may be litigated.

[36] Mr. Doyle apparently will testify (and produce evidence) that the manufacturer did not intend these U-bolts to be cinched down. CASE's Answer in Doyle Deposition at 5.

stresses was to negate the possible inference from that section that SA-307 steel could not induce excess stresses. As we said, that section does not, however, exclude that possibility.[37]

With respect to clamping forces, we admit that there is substantial persuasive force to Applicant's new argument that we have erroneously equated forces in pounds with stresses in psi.[38] However, we are still without any explanation of the magnitude of the local stresses caused by the "soft" pipe clamps and we are confident that such an explanation should be easy to provide in the course of Applicant's forthcoming explanation of its treatment of local stresses from stiff pipe clamps.

At Applicant's request, we have also reexamined our discussion of the Staff's testimony about inspections of U-bolts.[39] We find no error. The Staff relied on the inspection as a way of assuring that the U-bolts have not been overtensioned.[40] However, "overtensioning" should be understood in the context of the combined load to be faced by the U-bolts, including subsequent thermal and seismic stresses that are not observed during the walkdown. We conclude that Staff was incorrect in placing any substantial reliance on walkdown inspections as a method for determining that the preloading stresses are acceptable.

A further concern of Applicant is that we should not have stated that its engineers may not have been "sufficiently sensitive to plant safety."[41] However, our statement came in the context of a discussion of whether localized stresses have been adequately considered with respect to *stiff pipe supports*. In that context, it is our understanding that the stresses exceed a reasonable margin of safety but that Comanche Peak's engineers did not attend to that problem, even though an analogous problem concerning "soft" supports had been called to their attention by CASE. If we should subsequently receive evidence that reasonable consideration was given to localized stresses from stiff pipe supports, we would then find it appropriate to rescind our characterization of the engineers.

With respect to whether or not Mr. Doyle presented "detailed calculations" of thermal stresses on U-bolts, we may have made a semantic error in so characterizing his testimony, but Mr. Doyle discussed test data that he used to extrapolate data he considered relevant to the U-bolt problem.[42] CASE's findings discuss the precise amount of thermal

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[37] Design Decision, 18 NRC at 1431.
[38] Applicant's Reconsideration at 31.
[39] *Id.* at 30.
[40] SIT Report at 32.
[41] Applicant's Reconsideration at 32, *citing* Design Decision, 18 NRC at 1434.
[42] CASE's Findings at IV-16.

expansion that would be expected for a pipe/U-bolt assembly covered with 900° insulation and also calculates the portion of the U-bolt that would not be in contact with the pipe at all. Given Mr. Doyle's earlier calculations of stresses from pretensioning, which equal or exceed the total allowable, these "calculations" or "extrapolations" from experimental results required that Applicant answer.[43]

Applicant also asks that we acknowledge that the responsibility for local pipe stress analysis has been assigned to Gibbs & Hill; however, the evidentiary support offered for this statement is a weak reed. Applicant points to a portion of the SIT Report dealing with Welded Stepped Connections.[44] That section states that Gibbs & Hill analyzes "local effects due to integral attachments." However, it does not discuss any responsibility to analyze local effects from nonwelded attachments and it is our understanding of the iterative design process, based on a portion of the record made subsequent to the filing of Walsh/Doyle Findings, that level of detail usually provided to Gibbs & Hill is insufficient to make local stress analysis possible.[45] We are also not aware of any local stress analysis performed on nonwelded attachments or of any analysis that demonstrated that such an analysis was not necessary. With respect to "stiff" supports, at least, it appears to be necessary but not to have been done.

### 7. *AWS Code*

In its request for us to reconsider our findings on the AWS Code, Applicant does not appear to have understood the basis for our conclusions, so we will attempt to state them in different terms. Applicant claims to comply with the ASME Code by performing weld qualification tests. However, it has not described those tests to us so we do not know the extent to which compliance with those tests would satisfy other industry standards found in the AWS Code. Applicant has admitted that some AWS Code standards are applied by reference despite the ASME Code standards. We want to have a basis for determining whether Applicant has correctly defined the standards that should be applied by reference and those that need not be applied because they are obviated by compliance with the ASME Code.

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[43] This argument, which we consider to be largely semantic, does not seem sufficiently serious to have found its way into Applicant's motion.

[44] SIT Report at 49.

[45] Taylor, Tr. 8922-25.

Applicant also has questioned our findings about when Mr. Doyle informed it that AWS Code provisions should be applied to Comanche Peak. Applicant appears to be correct that the finding is based on a CASE finding that was not supported by the record.[46] However, this error is not relevant to our basic concern about whether AWS Code provisions are being applied to Comanche Peak. It is relevant to the question of whether Applicant has promptly corrected welding deficiencies brought to its attention. In the instance of the Beta provisions, adopted on May 11, 1982,[47] it would be helpful if Applicant explains and documents how its quality assurance program for design handled this problem with respect to each of the design groups, including how the problem was detected and what was done to assure the acceptability of previously made welds.[48] With respect to other AWS provisions, the operation of the quality assurance program need not be explained unless we first find that there were deficiencies in not applying those other AWS provisions.

With respect to the application of Korol and Mirza criteria to NPSI rear brackets,[49] we accept Applicant's clarification that it has not adopted those criteria. However, we still wish to know whether the particular rear brackets are adequately designed.

Concerning repair of welds by "capping," we disagree with Applicant that Mr. Doyle did not submit relevant testimony.[50] CASE's findings argue that complete fusion is needed for an adequate weld and it cites Mr. Doyle's testimony at Tr. 6262 in support of that proposition. Applicant never answered this argument and has not shown a basis for believing that its repair procedures are properly qualified or are acceptable. We agree with Applicant that Mr. Compton supported its position and not CASE's,[51] but we are unwilling to accept Mr. Compton's unexplained acceptance of cap welding as "customary" as assurance that the welding repair procedure is adequate.[52]

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[46] Applicant's Reconsideration at 35.

[47] CASE Exhibit 716 at 4 (page 3 of guidelines).

[48] *See* CASE's Answer in Doyle Deposition at 7.

[49] Design Decision, 18 NRC at 1436.

[50] Applicant's Reconsideration at 36.

[51] Tr. 7957-58.

[52] It would have been helpful to us to have had Applicant's comment on this point prior to reaching our decision. Inevitably, review of one party's findings without the benefit of an adversary comment will lead to too-ready acceptance of that party's point of view. In this instance, we too-readily accepted CASE's characterization of the Compton testimony.

### *8. Generic Stiffness Values*

Applicant correctly perceives that our problem with generic stiffness values is not with the study submitted to justify those values but with Applicant's initial justification.[53] In this instance, the SIT made an adverse finding and Applicant never explained why its design had the alleged deficiency. This apparently was part of the Applicant's and Staff's approach, which was to show that deficiencies had no consequence but not to address how deficiencies had arisen or whether they were adequately addressed by quality assurance.

We also agree that the one specific design problem mentioned at 1443 of the Design Decision was not related to the generic stiffness problem. This might more properly have been discussed in a separate section of our decision, called "Potential Rotation of the Plate in One Support."

### *9. Differential Seismic Displacement*

Applicant's current explanation, which was not available to us prior to issuance of the Design Decision, persuades us that Applicant may be able to explain this problem to our satisfaction. However, our record is still devoid of evidence concerning how it came about that PSE violated its own design guidelines, how this event came to be reflected in the design quality assurance system, and whether this problem was resolved promptly, as required by 10 C.F.R. Part 50, Appendix B, Criterion XVI.

### *10. Testing of Richmond Inserts*

We fail to understand from Applicant's argument why the Board may have been incorrect in its Richmond Insert findings. Although it is true that the Staff's findings, adopted by the Board,[54] failed to mention shear cone analysis done by the Applicant,[55] Applicant has not persuaded us that this omission is relevant to the Staff's findings concerning "allowable tension loads."[56] The SIT Report concluded that, "[a]s a result of the Applicant's assumptions as to shear load capability [in Applicant's calculation of allowable tension loads], the specified shear load allowables are 50 percent higher for the 1½-inch insert than the value

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[53] Applicant's Reconsideration at 37-38. However, CASE intends to challenge the appropriateness of using the stiffness study to generalize to other plant systems. This matter should be covered by the Plan Applicant intends to submit. CASE's Answer in Doyle Affidavit at 8.

[54] Design Decision, 18 NRC at 1445-46.

[55] Applicant's Reconsideration at 39-40.

[56] Design Decision, 18 NRC at 1445. NRC Staff Response to Applicant's Reconsideration, January 27, 1984, at 6-7.

recommended by the manufacturer."[57] The SIT Report found this to be a deficiency both because this was an inadequate safety margin, in the absence of further testing, and because "standard industry practice requires that testing be done to confirm the [published allowable shear] values."[58]

Applicant correctly states that the ultimate question is whether "the plant, *as built,* can and will be operated without endangering the public health and safety."[59] However, we wish to be assured that design quality assurance for pipe supports (including Richmond Inserts) has been adequate. If it has not been adequate, then we will examine other design issues before reaching a conclusion about the ultimate question of the safety of the plant.

### *11. Axial Torsion*

This is a part of our decision to which we addressed unusual attention. Our reasoning was set forth in the Design Decision, 18 NRC at 1446-49. Of the two principal analyses set forth in our record, by Chen and Doyle, we prefer the view expressed by Mr. Doyle, and Applicant has not even attempted to explain why we have erred. The fact that Applicant has had "independent opinions of outside experts" corroborating its view is certainly not even entitled to our attention.[60]

## II. STAFF'S RECONSIDERATION MOTION

The Staff requests us to rescind that portion of our decision in which we state that the Staff argued that Appendix B did not apply to design. On one issue we consider that the Staff's point is valid, and an analogous point made by Applicant is also valid.

Obviously, both the Staff and Applicant have always believed that Appendix B, Criterion III, which addresses design of a plant explicitly, applies to the design of a nuclear power plant. To this extent, both have acknowledged the applicability of Appendix B. However, both Applicant and Staff have taken an approach to this litigation that seems inconsistent with the realization that Criterion XVI, "Corrective Action," applies to the design of a plant. That is what we think Mr. Taylor meant when

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[57] SIT Report at 19.
[58] *Id.*
[59] [Emphasis added by Applicant in Applicant's Reconsideration at 40.] *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-756, 18 NRC 1340, 1345 (1983).
[60] Applicant's Reconsideration at 41. CASE correctly points out that these are "phantom" experts who, "[h]aving struck . . . move on without cross-examination or rebuttal." CASE's Answer at 25.

he said "Appendix B . . . does not address nonconforming design. It only addresses the conformance of the installed hardware and the inspection thereof to the design."[61] Because Criterion XVI does not require reports called "nonconformance reports" for construction or for design, we can think of no other appropriate interpretation of these remarks than that Criterion XVI does not apply to design.

We are pleased that both Applicant and Staff now agree that Appendix B is applicable to design. In particular, Applicant seems to agree that Criterion XVI is applicable to design. We infer that the Staff also agrees with that position.

## III. CASE'S RECONSIDERATION MOTION

In general, we do not interpret CASE's Reconsideration as a challenge to our decision. It is more in the nature of anticipatory objections to the Plan that Applicant will file in response to our decision. To the extent that we suggested criteria for such a Plan, these were just suggestions, not binding on either party. It will be open to CASE to attack CYGNA as an inappropriate design review organization, providing that it has the evidence to do so.[62] It will also be open to CASE to attempt to diminish the credibility of the CYGNA report, should one be submitted, should it be able to establish a legitimate conflict of interest concerning the relationship between Texas Utilities Electric Company and CYGNA.

## IV. REOPENING THE RECORD CONCERNING APPENDIX B

CASE argues, quite forcibly, that Applicant should not be permitted to submit evidence concerning its compliance with Appendix B, Criterion XVI. CASE believes that Applicant already had its opportunity to present the evidence and that it did not do so. We believe CASE's point is a serious one and set forth the following extensive quotation from its filing:

> Applicant has had more than ample time and occasion to propose additional hearings if at any time they felt they were warranted. Applicant *chose* not to do this. Instead,

[61] Taylor, Tr. 6707.

[62] We will not determine the merits of the conflict-of-interest controversy at this time because the matter has not been fully litigated. However, the current state of the record tends to minimize the importance of the conflict-of-interest allegation. Applicant's Answer to Case's Motion for Reconsideration of Board's 12/28/83 Memorandum and Order (Quality Assurance for Design), February 1, 1984, Affidavit of David H. Wade (attached).

Applicant has subjected the Licensing Board and parties to a constant barrage of pleadings and arguments to hurry up and close the record because "delay" by the Board could adversely impact Applicant's phony fuel load date.

> Applicant was arguing as far back as September *16, 1982,* that "the record as it stands right now is more than adequate for the Board to make findings on the allegations raised by Mr. Walsh and Mr. Doyle." (Tr. 5416/11-14.) Applicant's constant haranguing to *close the record* has continued right up until the Board's 12/28/83 Order when Applicant finally perceived that it had *had* its chance and had blown it....
>
> Throughout their pleading, Applicant *admits* that the Board *cannot* find that Applicant's pipe support design process satisfies the requirements of 10 C.F.R. Part 50, Appendix B. It argues that the Board should not find it in violation of Appendix B but should instead, without any basis in the record, allow Applicant to basically go back and start over at this late date. CASE can just imagine the response of the Applicant and NRC Staff had CASE made such a suggestion! In fact, the Board has refused to allow CASE to supplement the record in some instances already.... The Board cannot use a double standard in these proceedings.[63]

Regrettably, we are unable to accept CASE's suggestion because we do not consider reopening by either party to be entirely symmetrical.[64] We are permitting Applicant to reopen the record without a showing of good cause because it does not seem to us logical or proper to close down a multi-billion-dollar nuclear plant because of a deficiency of proof. While there would be some "justice" to such a proposition, there would be no sense to it.

Furthermore, we note that intervenors receive several procedural advantages in our proceedings that also are not fully symmetrical and that compensate for the application of different standards for reopening the record. First, the Board has the authority to raise important issues *sua sponte,* thereby protecting public safety and the environment even when intervenors may not have raised the issues. Second, the Board has the responsibility to assure the adequacy of the record, thereby causing it to pursue more fully matters of public safety that may not have been fully

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63 CASE's Answer at 5-6.

64 We have considered whether CASE's point about reopening the record is irrelevant because the record has never been closed. However, there is no clear guidance concerning whether the record should have been closed. We conclude that the close relationship between the questions of leaving the record open for inadequacy or closing the record and entertaining a motion for reconsideration requires the use of similar standards in these two situations.

In this case, there is a special reason to consider these two questions to be similar. Prior to our decision to leave the record open we had already given the parties a chance to file supplemental briefs, accompanied by affidavits, on two issues — the AWS Code and Pipe Clamp Stresses — that we still consider to be inadequately addressed in our record.

We conclude that it is appropriate to consider the posture of this case to be similar to the posture of a case in which applicant has filed a motion to reopen the record. Consequently, we have chosen to address the applicability to this case of the previously enunciated standards for reopening the record.

pursued by intervenors. (For example, the Board has considered certain construction deficiency questions even though CASE failed to file findings on those issues.) Third, the burden of proof generally falls on applicants, who must therefore attempt to appreciate and rebut, by a preponderance of the evidence, all the implications of all issues raised by intervenors.

In one sense, the reopening of the record does not seem fair. CASE has been put to unnecessary expense because it will have to prove its case twice. In addition, the need to continue disputing an already closed issue is an unnecessary tax on its volunteer resources. Because of the burden imposed by our decision and the lack of precedent for failing to apply the standard for reopening the record to Applicant, we have extended to the parties, including CASE, an invitation to request that we refer the Design Decision for review by the Appeal Board.[65]

## V. THE ITERATIVE DECISION PROCESS

We are hopeful that the Board's response to the pending motions for reconsideration will serve two purposes. First, to correct errors that have been brought to our attention. Second, to help to clarify matters in our decision that the parties had difficulty interpreting or that they considered to be in error.

Our efforts to encourage the filing of motions to reconsider are, we realize, somewhat unusual. However, we consider the exercise to be a constructive way to refine issues and manage the remainder of the proceeding.

We anticipate that the next round of hearings should be the last. At some point, prolongation of hearings would represent a denial of due process to one or more of the parties. We encourage the parties to present their evidence and to prepare their required Proposed Findings with care, being sure to present a reasoned basis for the decision sought from the Board.

# ORDER

For all the foregoing reasons and based on consideration of the entire record in this matter, it is, this 8th day of February 1984,

ORDERED

[65] Design Decision, 18 NRC at 1456.

That Footnote 37 be struck from our Memorandum and Order (Quality Assurance for Design), LBP-83-81, prior to publication.

That LBP-83-81 shall in other respects be unmodified but that it shall be interpreted in light of the Memorandum accompanying this Order.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Walter H. Jordan (by PBB)
ADMINISTRATIVE JUDGE

Kenneth A. McCollom (by PBB)
ADMINISTRATIVE JUDGE

Bethesda, Maryland

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Morton B. Margulies,** Chairman
**Dr. Jerry R. Kline**
**Dr. David L. Hetrick**

**In the Matter of** **Docket No. 50-261-OLA**
**(ASLBP No. 83-484-03-LA)**

**CAROLINA POWER & LIGHT COMPANY**
**(H.B. Robinson Steam Electric Plant, Unit 2)** **February 10, 1984**

The Licensing Board dismisses this proceeding finding that the withdrawal of all remaining contentions by the sole intervenor has eliminated the basis for which the adjudicatory hearing was ordered.

# ORDER DISMISSING PROCEEDING

We ordered the holding of an adjudicatory hearing on the application of Carolina Power & Light Company to amend its license for operation of the H.B. Robinson Steam Electric Plant, Unit 2, to permit repair of the steam generators by replacement of major components. The decision was based on four contentions that were submitted by the Hartsville Group, a party intervenor.

Prior to the commencement of the adjudicatory hearing on February 7, 1984, the Hartsville Group withdrew one of the contentions and on motion of the Applicant we ordered the dismissal of another. During the course of the hearing the Hartsville Group withdrew its two remaining contentions thereby eliminating the entire basis for which the adjudicatory hearing was ordered. The need for a hearing no longer exists and therefore the adjudicatory proceeding is dismissed.

The matter of the amendment of the license may be handled by the Nuclear Regulatory Commission Staff.

It is so *Ordered.*

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Morton B. Margulies, Chairman
ADMINISTRATIVE LAW JUDGE

Dated at Bethesda, Maryland,
this 10th day of February 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF NUCLEAR REACTOR REGULATION**

**Harold R. Denton,** Director

| | |
|---|---|
| **In the Matter of** | **Docket No. 50-320 (10 C.F.R. § 2.206)** |
| **GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION (Three Mile Island Nuclear Station, Unit 2)** | **February 17, 1984** |

The Director of the Office of Nuclear Reactor Regulation denies a petition submitted by Marvin Lewis requesting that the Commission postpone the lifting of the reactor pressure vessel head at the Three Mile Island Nuclear Station, Unit 2.

## TECHNICAL ISSUE DISCUSSED: PYROPHORIC CONDITIONS

Based upon the staff's reviews and experience to date, there does not appear to be an undue risk to public health and safety from the possible formation of pyrophoric materials in the pressure vessel.

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

By letter dated September 13, 1983 to the Secretary of the Commission, Mr. Marvin Lewis requested that the Commission postpone the lifting of the reactor pressure vessel head at Three Mile Island Nuclear Station, Unit 2 (TMI-2). Mr. Lewis' letter was supported by a letter dated November 1, 1983, from Professor Earl Gulbransen of the

University of Pittsburgh to the Secretary of the Commission. Attached to Professor Gulbransen's letter was a paper on the effects of oxygen, nitrogen and hydrogen on the mechanical properties of zirconium. Mr. Lewis' letter and the supporting letter from Professor Gulbransen were referred to the Office of Nuclear Reactor Regulation for treatment as a petition pursuant to 10 C.F.R. § 2.206 of the Commission's regulations.

I have reviewed the information contained in Mr. Lewis' petition, the information in Professor Gulbransen's letter of November 1, 1983, and other information pertinent to the issues raised by the petition. For the reasons stated in this decision, Mr. Lewis' request is denied.

## PETITIONER'S ASSERTION AND REQUEST

Mr. Lewis contends that pyrophoric materials[1] may well exist within the reactor pressure vessel (RPV) and that the quantity of these materials is unknown. As a consequence, Mr. Lewis believes that the lifting of the RPV head is a "dangerous maneuver" which could result in a pyrophoric event. Mr. Lewis bases the likely existence of pyrophoric materials within the RPV on the conditions which existed within the vessel during the TMI accident. Mr. Lewis contends those conditions were favorable for the formation of pyrophoric zirconium or zirconium hydride, which can react violently when exposed to air. Consequently, Mr. Lewis requests that the RPV head lift be postponed pending a "public review" of the pyrophoricity issue. Mr. Lewis' contentions are supported by Professor Gulbransen, who also asserts that finely divided zirconium or zirconium hydride may well have been formed during the accident. Given the potential pyrophoricity of these materials, Professor Gulbransen warns that these materials must be kept under water pending further characterization of their pyrophoric nature. He urges that the greatest caution be exercised before proceeding with the RPV head lift.

## STAFF REVIEW OF THE PYROPHORICITY ISSUE

By letters dated May 25, May 26, and July 20, 1983, General Public Utilities Nuclear Corporation, the TMI Unit 2 licensee, forwarded to the NRC safety evaluation reports to support the planned reactor vessel Un-

[1] Pyrophoric materials are those which are capable of igniting spontaneously in air.

derhead Characterization Study.[2] This study was conducted during the months of August through October 1983 to gather data for the RPV head lift and involved a number of different activities. These activities included the lowering of the water in the reactor vessel to a level approximately 1 foot below the top of the plenum (*see* Figure 1), the measurement of the radiation fields underneath the RPV head, the measurement of the radiation fields around the RPV head and service structure, the visual inspection under the RPV head with a TV camera, the measurement of the topography of the core cavity with an ultrasonic device, and the removal of six samples from the core debris bed. Inasmuch as these activities, specifically the lowering of the water level in the reactor vessel, involved the uncovering of equipment (the plenum cover) which was previously covered with water, it was necessary to address in advance the issue of exposing potentially pyrophoric material to air. Accordingly, the issue of pyrophoricity was addressed by the licensee as part of its Underhead Characterization Study. Thereafter, the hazard posed by pyrophoric materials in the TMI-2 reactor vessel was extensively evaluated by the NRC staff in its review and approval of the Underhead Characterization Study.[3] The staff was particularly concerned with the potential for pyrophoric reactions of materials on the plenum cover and of samples removed from the core debris bed. The staff determined in its safety evaluation that:

(1) the presence of steam (*i.e.*, an oxidizing agent) and the temperature conditions during the accident would make it unlikely that significant quantities of zirconium hydride in a pyrophoric condition were produced during the accident,

(2) the primary system flow dynamics during the TMI-2 accident would not likely have transported large quantities of pyrophoric material, if formed, to the top of the plenum, and

(3) any pyrophoric materials in finely divided form would be dispersed and mixed with inert materials of core debris which would prevent the development of pyrophoric conditions.

Following the staff's approval, the Underhead Characterization Study was conducted by the licensee. As described below, all of the visual observations of the reactor vessel underhead conditions and laboratory

[2] *See* Letter from B.K. Kanga to L.H. Barrett, 4410-83-L-0098, Underhead Characterization Study (May 25, 1983); Letter from B.K. Kanga to L.H. Barrett, 4410-83-L-0100, Underhead Characterization SER, Core Topography Addendum (May 26, 1983); Letter from B.K. Kanga to L.H. Barrett, 4410-83-L-0133, Underhead Characterization SER, Core Sampling Addendum (July 20, 1983).

[3] Details concerning the staff's review are found in the following letters: Letter from L.H. Barrett to B.K. Kanga, NRC/TMI-83-043, *Reactor Vessel Underhead Characterization Safety Evaluation* (July 13, 1983); Letter from L.H. Barrett to B.K. Kanga, NRC/TMI-83-053, Response to Core Debris Safety Evaluation Report (SER) (August 19, 1983).

# REACTOR & SERVICE STRUCTURE

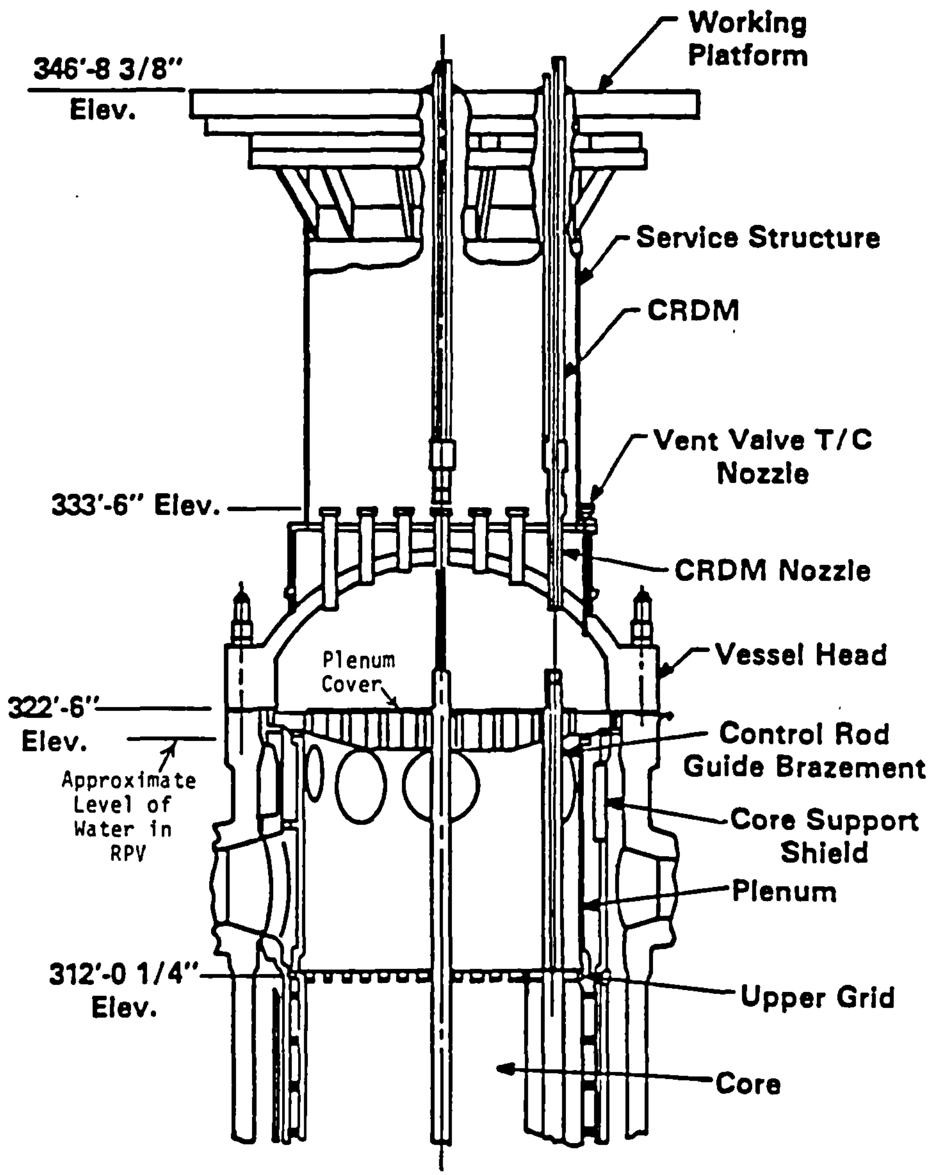


FIGURE 1

analyses of the chemical and pyrophoric properties of samples obtained from components within the reactor vessel and from solids filtered from the reactor coolant support the conclusions reached by the staff in its safety evaluation report.

The activities undertaken during underhead characterization to address pyrophoricity concerns were as follows. As a precaution prior to the lowering of the water level below the top of the plenum, the licensee conducted a closed-circuit television underwater inspection of portions of the plenum cover and observed that only an insignificant layer of material, approximately 1 millimeter in depth, was present on some of the plenum surfaces inspected. This observation verified the staff's conclusion that it was not likely that significant quantities of materials had been transported to the top of the plenum during the accident. Following the visual inspection, the licensee obtained two samples of the material from the plenum surface and the samples were tested for pyrophoricity by various attempts to initiate a pyrophoric reaction. The tests included a spark test (*i.e.*, an attempt to ignite the material with an electrically generated spark) and a flame test (*i.e.*, an attempt to ignite the material with a propane torch with approximate flame temperature of 2300°F). The spark test is perhaps the most reliable test for establishing the pyrophoric characteristics of a material in question as it provides an initiator (*i.e.*, the spark) for a reaction, if one can occur. The flame test is an extreme test that would show whether the material in question has any tendency to ignite at all or whether the material is completely inert.

For comparison with the tests on the plenum samples, the spark and flame tests were performed with some "cold" (*i.e.*, commercially available, nonradioactive elements and compounds) materials in powdered form, including iron, zirconium, and zirconium oxide. The particle size for the iron and zirconium powders was 62 microns or less and the particle size for the zirconium oxide was 125 microns or less. The cold tests demonstrated that the zirconium powder would ignite for both the spark and flame test; however, the material did not ignite spontaneously in the laboratory at atmospheric pressure and ambient temperature (*i.e.*, approximately 70°F). The powdered iron and zirconium oxide failed to ignite in either the spark or flame test.

The spark and flame tests on the samples removed from the plenum also failed to ignite the material, indicating the presence of little, if any, pyrophoric material and the absence of any pyrophoric characteristics. In fact, the plenum samples showed no more tendency to ignite than the "cold" iron and zirconium oxide samples. Both the "cold" laboratory tests and the tests on the plenum samples were videotaped by the licensee and the videotapes were reviewed by the NRC staff.

In addition to the pyrophoricity tests described above, the licensee performed chemical analyses of solids filtered from the reactor coolant system and of the thin films scraped from the surfaces of the control rod drive mechanism (CRDM) leadscrews removed from the reactor vessel head. *See* Figure 1. These analyses indicated the absence of zirconium metal and hydride particles. Based on the visual examinations, analyses and tests which indicate the probable absence of pyrophoric materials on the plenum cover, the NRC approved the lowering of the RPV water level to approximately 1 foot below the plenum surface, which enabled the licensee to proceed with the underhead characterization effort. The water was lowered to this level to simulate the radiological conditions that will exist for the RPV head lift. As a result, the plenum cover has been exposed to air since August 20, 1983, without any adverse impact. This condition has been visually confirmed by closed-circuit television inspection conducted subsequent to the lowering of the water level. Additionally, the six samples which were removed from the core debris bed have been exposed to air for several months with no indication of pyrophoric reactions.

The information resulting from the visual observation of the plenum and the analyses and tests on materials removed from within the RPV indicates that: (1) little material is present on the plenum surface, (2) the material on the plenum surface is not pyrophoric, (3) material filtered from the reactor coolant system during the accident lacks any pyrophoric content, (4) material scraped from CRDM leadscrews lacks any pyrophoric content, and (5) samples of material removed from the damaged core have not shown any tendency to undergo a pyrophoric reaction. Accordingly, the staff concludes that there is little potential for a pyrophoric event with the plenum cover exposed to air. The information provided by Mr. Lewis and Professor Gulbransen is of a general nature concerning pyrophoricity and the dangers that phenomenon poses for the head lift. The staff does not disagree with the petitioner that pyrophoric conditions could have developed in the RPV following the TMI accident. For that reason, prior to the receipt of the petition, the staff considered the issue of pyrophoricity as it relates to the licensee's proposed Underhead Characterization Study. Based upon the staff's reviews and the experience to date as described above, there does not appear to be an undue risk to public health and safety from the possible formation of pyrophoric materials in the pressure vessel.

With regard to Mr. Lewis' and Professor Gulbransen's cautions about proceeding with the RPV head lift on the basis of pyrophoricity concerns, it should be noted that the water level in the reactor vessel is presently at 1 foot below the plenum cover. This level is precisely that planned for the RPV head lift. No further lowering of the water level is contemplated for the RPV head lift. Thus, no further safety review of pyrophoric issues as related to the head lift is warranted. Moreover this issue has been addressed by actual experience along with evaluations, analyses, tests, and activities performed in connection with the Underhead Characterization Study.

Inasmuch as potential pyrophoric conditions have been given appropriate consideration and do not pose a significant hazard to the head lift, I have determined that no adequate basis exists for postponing the planned lift of the reactor vessel head or initiating proceedings to review the issue of pyrophoricity. Consequently, the petitioner's request is denied.

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

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland,
this 17th day of February 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung, Director**

**In the Matter of**

**Docket No. 50-293**
**(10 C.F.R. § 2.206)**

**BOSTON EDISON COMPANY**
**(Pilgrim Nuclear Power Station)**

**February 27, 1984**

The Director of the Office of Inspection and Enforcement grants in part and denies in part a petition submitted by the Massachusetts Public Interest Research Group requesting that the NRC take action with respect to the state of emergency planning at Pilgrim facility. Among the specific relief requested was the initiation of the 4-month period specified by the Commission's regulations within which to correct the alleged deficiencies at the Pilgrim facility and consideration by the Commission as to whether the state of emergency preparedness in conjunction with the alleged poor safety record at the Pilgrim facility warrants immediate shutdown or operation of the facility at reduced power.

## TECHNICAL ISSUE DISCUSSED: EMERGENCY PLANNING

The Federal Emergency Management Agency takes the lead in offsite emergency planning and reviews and assesses State and local emergency plans for adequacy. The NRC assesses the licensee's site emergency plans for adequacy and makes decisions with regard to the overall state of emergency preparedness.

## EMERGENCY PLAN: EMERGENCY PLANNING ZONE

The Commission's regulations preclude an Emergency Planning Zone (EPZ) radius significantly in excess of 10 miles. An EPZ of about 10 miles is considered large enough to provide a response base which would support activity outside the planning zone should this ever be needed.

## EMERGENCY PLAN: EVACUATION PLAN

The Commission has adopted an approach to emergency planning in which evacuation is only one of several possible responses to an emergency. It is unlikely that evacuation of the entire plume EPZ would be required in the event of an accident. Pending a final determination regarding the adequacy of evacuation time estimates, it is reasonable to conclude that the public health and safety will be reasonably assured in the interim by continued licensee compliance with Commission requirements regarding emergency planning and other health and safety requirements aimed at keeping the probability of serious accidents very low.

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

## INTRODUCTION

In its "Petition of the Massachusetts Public Interest Research Group for Emergency and Remedial Action" (Petition) dated July 20, 1983, the Massachusetts Public Interest Research Group (hereinafter referred to as Petitioner) requested that the Nuclear Regulatory Commission (NRC) take action to remedy alleged serious deficiencies in the offsite emergency response plans for the Pilgrim Nuclear Power Station in Plymouth, Massachusetts. Among the specific relief requested was the initiation of the 4-month period specified by the Commission's regulations, specifically 10 C.F.R. § 50.54(s)(2)(ii), within which to correct the alleged deficiencies at the Pilgrim facility and consideration by the Commission as to whether the state of emergency preparedness in conjunction with the alleged poor safety record[1] at the Pilgrim facility

[1] The Petition, in the relief it requested, made reference to the poor safety record at the Pilgrim facility as a reason for granting the relief. As stated in the September 6, 1983 letter to the Petitioner, with regard to Pilgrim's safety record since 1981, in mid 1982 the licensee initiated a Performance Improvement Plan pursuant to an NRC Order (47 Fed. Reg. 4171 (1982)) to improve the plant's performance. This plan, which was submitted to the NRC on July 30, 1982, has senior utility management involvement in assuring quality and has resulted in marked improvement in Pilgrim's operating record over the *(Continued)*

warrants immediate shutdown or operation of the facility at reduced power.

The Petitioner's request is based upon a report by the Petitioner entitled "Blueprint for Chaos II: Pilgrim Disaster Plans Still a Disaster" (hereinafter referred to as the Chaos II Report), the "Comments of Attorney General Francis X. Bellotti Relative to Off-Site Emergency Planning for the Pilgrim Nuclear Power Station" (hereinafter referred to as the Comments of the Attorney General), and upon two reports by the Federal Emergency Management Agency (FEMA) – "Interim Findings: Joint State and Local Radiological Emergency Response Capabilities for the Pilgrim Nuclear Power Station, Plymouth, Massachusetts," dated September 29, 1982, and "Report on the Pilgrim Nuclear Power Station Siren Test, June 19, 1982," dated January 1983.

In its Chaos II Report, the Petitioner has reviewed offsite emergency planning for the Pilgrim facility and claims to have identified certain deficiencies with regard to the size of the plume exposure pathway Emergency Planning Zone (EPZ), advance information provided to the public on what actions to take in the event of an emergency, required notifications during an accident itself, and evacuation planning and sheltering including the adequacy of reception and medical facilities. In each of these areas, the Petitioner makes various recommendations as to actions which it believes are required to improve the state of preparedness at the Pilgrim facility. The Petition states that the findings of the Chaos II Report are supported in part by a telephone survey of 100 residents of the EPZ conducted by the Petitioner. The survey was conducted between February and May of 1983.

In further support of its Petition, Petitioner references the Comments of the Attorney General which also question the adequacy of emergency planning for the Pilgrim facility. Specifically, Petitioner argues that the Comments of the Attorney General support Petitioner's claims that the EPZ has been drawn too small and that evacuation plans are inadequate.[2] The Comments of the Attorney General are based in part upon a study prepared for the Attorney General by MHB Technical Associates of San Jose, California.

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past 2 years. The last Systematic Assessment of Licensee Performance report, for the period July 1, 1982 to June 30, 1983, gave Pilgrim a Category 1 ("high-level performance") rating in emergency planning, a Category 2 ("satisfactory performance") rating in plant operations, and an overall Category 2 rating in the eight functional areas assessed. Since late 1981, there has been continued improvement in Pilgrim's performance with respect to operational safety. A satisfactory level of management attention and involvement in plant safety matters now exists.

[2] The Comments of the Attorney General were forwarded to FEMA on August 25, 1982. While the Comments of the Attorney General raise other issues related to the Pilgrim facility, the Comments are relied upon by the Petitioner only to support its claims regarding the adequacy of the current EPZ size and evacuation planning. *See* Petition at 6; Chaos II Report at 26.

## DISCUSSION

Emergency preparedness at the Pilgrim facility has been reviewed by both the NRC and FEMA. The NRC Final Rule on Emergency Planning (45 Fed. Reg. 55,402) became effective on November 3, 1980. FEMA and the NRC have jointly developed criteria for implementing these regulations; specifically the agencies have developed a guidance document entitled, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/FEMA-REP-1, Rev. 1 (NUREG-0654). The cooperative relationship between NRC and FEMA is described in a "Memorandum of Understanding Between NRC and FEMA Relating to Radiological Emergency Planning and Preparedness" of January 1, 1980 (45 Fed. Reg. 5847). Under the Memorandum of Understanding, FEMA takes the lead in offsite emergency planning and reviews and assesses State and local emergency plans for adequacy. The NRC assesses the licensee's site emergency plans for adequacy and makes decisions with regard to the overall state of emergency preparedness. The NRC and FEMA undertook a review of the state of emergency preparedness at the Pilgrim facility in accordance with the requirements of the final rule in emergency planning.

## NRC REVIEW

The NRC initiated the process of reviewing the licensee's emergency plan in 1979 in connection with its review of the construction permit application for Pilgrim Unit 2. Following the rule change in November 1980, an upgraded site emergency plan was submitted for the Pilgrim facility. The results of the NRC's evaluation of the licensee's upgraded emergency plan and an examination of the implementation of the plan, conducted during an Emergency Preparedness Implementation Appraisal (EPIA) on July 13-24, 1981, are summarized in Inspection Report 50-293/81-15 dated June 22, 1982. The findings of the EPIA indicated that certain corrective actions were required by the licensee in the emergency plan and in the implementation of its emergency plan in order to achieve an effective emergency preparedness program. The EPIA also identified areas of lesser significance where the licensee could improve its emergency preparedness. The licensee responded to the concerns identified by the NRC in a letter dated July 28, 1982, wherein the licensee concluded that the significant findings which had been identified in the EPIA report had been adequately addressed. Following the receipt of the licensee's response to the EPIA report, on August 5, 1982, the

NRC met with the licensee to discuss the status of EPIA findings. The NRC agreed with the licensee's actions on sixteen of the twenty significant findings, and only four of the twenty significant findings required further discussion. These four areas were dose assessment, recommended protective actions, in-plant surveys, and procedures related to emergency repair and corrective actions. After discussion of these four items, it was resolved that the licensee would take the necessary corrective actions. In its November 1, 1982 correspondence, the licensee reported that all planned actions relevant to the significant findings had been completed, informed the NRC of the progress on actions planned pertaining to the improvement items, and transmitted its response to the emergency plan evaluation findings. The licensee's response addressed each item identified in the EPIA. On December 29, 1982, the NRC Region I Office acknowledged the corrective actions that had already been taken and those planned by the licensee and informed the licensee that all corrective actions would be examined during a future inspection.

The licensee's action on the significant findings was verified during follow-up inspections conducted by Region I of the NRC on March 1-4, 1983, and June 21-August 15, 1983, and summarized in Inspection Reports 50-293/83-05 dated April 20, 1983 and 50-293/83-17 dated September 8, 1983. Within the scope of the follow-up inspections, no violations were observed and only one inspector follow-up item was identified.

In addition, on March 3, 1982 and June 29, 1983, the licensee conducted full-scale emergency exercises which were observed by both the NRC and FEMA. The NRC's findings are presented in Inspection Reports 50-293/82-09 dated March 24, 1982 and 50-293/83-16 dated July 29, 1983, in which it was determined that the emergency response actions taken by licensee personnel were adequate to provide protective measures for public health and safety. As a result of these review activities, there continues to be reasonable assurance that onsite emergency preparedness is adequate to protect the public health and safety.

## FEMA REVIEW

FEMA, in accordance with the Memorandum of Understanding, has reviewed the adequacy of offsite emergency preparedness at the Pilgrim facility. A preliminary review of the Massachusetts State Radiological Plan was conducted in October 1981 by the Regional Assistance Committee (RAC).[3] Based on the preliminary review, the RAC concluded that the plan was in an advanced but incomplete stage and that further revision to the plan was required in order to conform to the guidance criteria of NUREG-0654. The NRC requested that FEMA review the process for prompt protective action decisionmaking in Massachusetts based on draft State plans and information submitted to the RAC in early 1982. On June 11, 1982, FEMA issued an interim finding that the current protective action decisionmaking process in Massachusetts was adequate to provide for public protection. Formal submission of emergency plans to the RAC by State and relevant local jurisdictions was followed by the first joint radiological emergency response exercise on March 3, 1982. The exercise involved emergency preparedness organizations at both the State and local levels. The performance of these organizations in implementing their radiological emergency response plans was observed. Deficiencies were identified as a result of this exercise and corrective actions initiated by the parties involved. On September 10, 1982, FEMA Region I issued its "Exercise Report — Joint State and Local Radiological Emergency Response Exercise for the Pilgrim Nuclear Power Station, Plymouth, Massachusetts, March 3, 1982." By memorandum dated November 2, 1983, FEMA provided to the NRC its "Interim Findings — Joint State and Local Radiological Emergency Response Capabilities for the Pilgrim Nuclear Power Station, Plymouth, Massachusetts" dated September 29, 1982. The interim findings were based on a summary evaluation of the Massachusetts Radiological Emergency Response Plan and the exercise of the State and local emergency response plans held on March 3, 1982. Although deficiencies were identified which required corrective action, FEMA found that the Massachusetts State and local emergency plans and preparedness for coping with the offsite effects of radiological emergencies that may occur at the Pilgrim Nuclear Power Station were adequate to protect the public.

The second joint radiological emergency response exercise at Pilgrim was held on June 29, 1983. A seventeen-member Federal team was assigned to evaluate State, local and field activities. By memorandum dated November 29, 1983, FEMA transmitted to NRC its "Final Report of the Joint State and Local Radiological Emergency Response Exercise

[3] There exists in each of the ten standard Federal Regions a Regional Assistance Committee (RAC) (formerly the Regional Advisory Committee) chaired by a FEMA Regional official and having members from the Nuclear Regulatory Commission, Department of Health and Human Services, Department of Energy, Department of Transportation, Environmental Protection Agency, the United States Department of Agriculture and Department of Commerce. The RACs assist State and local government officials in the development of their radiological emergency response plans, review plans, and observe exercises to evaluate the adequacy of these plans and related preparedness. A description of the RAC authority and responsibilities is found in 44 C.F.R. Part 350.

for the Pilgrim Nuclear Power Station, Plymouth, Massachusetts," dated September 26, 1983 (1983 Exercise Report). The 1983 Exercise Report identifies no deficiencies that would lead to a negative finding.[4] Deficiencies requiring corrective action were identified by FEMA in two areas — the State police radio notification system and the transmission of meteorological information. FEMA also identified other deficiencies and additional areas of improvement for consideration by the State and local authorities regarding their offsite emergency preparedness program. FEMA will furnish a copy of the 1983 Exercise Report to the Commonwealth of Massachusetts and will request a schedule of actions for the correction of deficiencies. A copy of the 1983 Exercise Report was sent to NRC Region I on January 12, 1984 for its use in coordinating with FEMA Region I in ensuring that the identified deficiencies are addressed in a timely manner.

Following receipt of the Petition, the Petition and the supporting Chaos II Report were forwarded to FEMA for its evaluation and review since the Petition questioned the adequacy of offsite emergency preparedness at the Pilgrim facility. By memorandum dated November 9, 1983, FEMA provided to the NRC its final report entitled "Analysis of Emergency Preparedness Issues at Pilgrim Nuclear Power Station Raised by the Massachusetts Public Interest Research Group (MASSPIRG)," dated November 3, 1983, attached hereto as Appendix A. The November 3, 1983 report indicates that FEMA has reviewed the Petition and has also consulted with members of the RAC and officials of the Commonwealth of Massachusetts. This review resulted in FEMA confirming its interim finding referred to above that the Commonwealth of Massachusetts has demonstrated that there is reasonable assurance that the public would be adequately protected if there were an accident at the Pilgrim Nuclear Power Station. In addition, in its November 3, 1983 report, FEMA indicated that the results of the 1982 Exercise Report have been superceded by the results of the 1983 Exercise Report. In effect, the numerous deficiencies identified by FEMA in its 1982 Exercise Report have been corrected or otherwise resolved. Thus only two deficiencies requiring corrective action, as described above, remain outstanding.

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[4] On August 5, 1983, FEMA Headquarters revised their procedural policy on exercise observation and evaluation in order to provide a more uniform, workable approach for use by the ten FEMA regional offices in their exercise reporting process. The guidance provides for reporting deficiencies which would lead to a negative finding, deficiencies which require corrective action but otherwise would not lead to a negative finding, and other deficiencies where a correctable weakness is noted for which corrective action should be considered. Deficiencies that would lead to a negative finding would cause a finding that offsite emergency preparedness is not adequate to provide reasonable assurance that appropriate protective measures could be taken to protect the health and safety of the public.

The NRC has reviewed the November 9, 1983 FEMA response and concurs with the conclusions reached therein. However, further discussion is appropriate regarding the following issues raised by the Petitioner and addressed in the FEMA report.

## I. Capability of the Licensee to Make Accurate Release Estimates

The FEMA report notes at 6-7 that the role of the licensee in preparing release estimates upon which to make protective action determinations is more properly an NRC evaluation responsibility than that of FEMA. The NRC agrees that the licensee's capability is a proper area for NRC evaluation. During the EPIA, described previously, NRC inspectors conducted walk-through inspections with members of the licensee's onsite emergency organization. These inspections were conducted in the areas of control room dose projections, dose assessment, event classification, offsite notification, offsite monitoring and environmental assessment. The inspections identified deficiencies in the areas of the dose assessment scheme, basis for recommended protective actions and related procedures and training. The licensee took corrective actions on these deficiencies and, as mentioned above, follow-up inspection on the EPIA findings conducted by NRC Region I verified that corrective action had been taken by the licensee on all significant findings identified during the EPIA. Additionally, on March 3, 1982, a team of NRC observers was on hand to witness the full-scale exercise held at Pilgrim. During the conduct of the exercise, eleven NRC team members made detailed observations in various areas including: detection, classification and assessment; direction and coordination of the emergency response; notification; and dose projection and consideration of protective actions. The NRC team concluded that, while there was some room for improvement, there were no items which exhibited a potential for significant degradation of emergency response. Similar observations were made at the second full-scale exercise at Pilgrim on June 29, 1983. In this instance, the NRC team concluded that the licensee demonstrated the capability to implement its emergency plan and emergency plan implementing procedures in a manner which would adequately provide for the health and safety of the public.

## II. Size of the EPZ

The Petitioner suggests that the EPZ size may require considerable expansion. However, this is in effect an attack on the Commission's regulations, specifically 10 C.F.R. § 50.47(c)(2). The Commission's regulation sets EPZ size at "about 10 miles." While the regulation would allow leeway for a mile or two in either direction based upon local factors, it clearly precludes an EPZ radius significantly in excess of 10 miles as suggested by the Petitioner. *See Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), LBP-82-39, 15 NRC 1163, 1177-84 (1982), *aff'd,* ALAB-717, 17 NRC 346 (1983). However, even considering the Petitioner's assertion on its merits, the information provided by the Petitioner does not support enlargement of the EPZ.

The FEMA report of November 3, 1983 makes reference to the MHB Technical Associates Study used by Petitioner to support its request that the EPZ size for the Pilgrim facility should be enlarged. Petitioner's request is based in part on a review of preliminary Calculation of Reactor Accident Consequences (CRAC) results conducted by MHB Technical Associates for the Attorney General. The MHB Study is entitled "Review of Calculation of Reactor Accident Consequences (CRAC 2) Results and Liquid Pathways (NUREG-1596) Study: Implications for Emergency Planning in the Vicinity of the Pilgrim Nuclear Power Station." Under contract to the Department of the Attorney General for the Commonwealth of Massachusetts, MHB Technical Associates reviewed the CRAC computer code and its results for the Pilgrim Station and NUREG/CR-1596 "Consequences from Liquid Pathways After a Reactor Meltdown Accident," August 1981. The Petitioner argues that the MHB conclusions regarding the CRAC code require enlargement of the Pilgrim EPZ. The MHB study attempts to apply a generic study to a site-specific case. The CRAC calculations were carried out for a report which was written to support the formulation and comparison of possible siting criteria for nuclear power plants, and generic rather than site-specific parameters were used.[5] A realistic estimate of the risk from severe accidents at each plant was not attempted for that report.

The plume EPZ[6] for the Pilgrim facility is based upon NUREG-0654 guidance criteria.[7] The joint NRC/EPA Task Force that developed NUREG-0396 considered several possible rationales for establishing the

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[5] Technical Guidance for Siting Criteria Development, NUREG/CR-2239, December 1982. In NUREG/CR-2239, a generic rather than plant-specific power level was used; regional rather than site-specific assumptions regarding evacuation and relocation were used; and generic releases were assumed, as opposed to the design-specific release categories used for licensing.

[6] The plume exposure pathway Emergency Planning Zone (EPZ) established for the site is located entirely within the State of Massachusetts. Its boundary extends 9.5 to 12 miles from the site and includes portions of five townships.

[7] The guidance criteria of NUREG-0654 are derived from NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Reactors," December 1978, which provides the concept of generic Emergency Planning Zones.

size of the EPZs. These included risk, probability, cost effectiveness and an accident consequence spectrum. The Task Force chose to base EPZ size on a full spectrum of accidents and corresponding consequences tempered by probability considerations. It was the consensus of the Task Force that a plume EPZ of about 10 miles would provide an adequate planning base beyond which actions could be taken on an *ad hoc* basis using the same considerations that went into the initial action determinations. In its statement on "Planning Basis for Emergency Response to Nuclear Power Accidents," 44 Fed. Reg. 61,123 (1979), the Commission noted that an EPZ of about 10 miles is considered large enough to provide a response base which would support activity outside the planning zone should this ever be needed.

The Petitioner contends that, based upon the referenced CRAC code results, an enlargement of the current Pilgrim plume EPZ is warranted because the projected doses exceed the EPA Protective Action Guides (PAGs)[8] outside the 10-mile EPZ. Both NUREG-0654 and NUREG-0396 recognize, based upon CRAC code results, that the PAGs might be exceeded beyond the 10-mile plume exposure EPZ in the event of the worst possible accident and meteorological conditions. However, a 10-mile plume exposure EPZ was still chosen as a planning basis in NUREG-0654 because:

a. projected doses from the traditional design basis accidents would not exceed Protective Action Guide levels outside the zone;
b. projected doses from most severe fuel degradation sequences would not exceed Protective Action Guide levels outside the zone;
c. for the worst fuel degradation sequences, immediate life-threatening doses would generally not occur outside the zone; and
d. detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary.

On balance, the MHB Study referred to in the Comments of the Attorney General and used by Petitioner in support of its Petition does not

[8] The EPA has developed and the NRC has adopted a "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, revised February 1980, which provides guidance criteria for public health officials in determining the need for and in choosing the appropriate protective actions. The Protective Action Guide (PAG) is the projected dose to individuals in the population which warrants taking protective action, *e.g.*, sheltering or evacuation.

provide an adequate basis for reconsideration of the specific size of the Pilgrim plume EPZ.[9]

## III. Evacuation Time Estimates

In reviewing the Petition, the NRC staff considered information available to it concerning Evacuation Time Estimates (ETEs) and determined that, as Petitioner suggested, potential bottlenecks to effective evacuation of the EPZ may exist on the periphery of the EPZ. It would be important to control traffic beyond the EPZ so that such traffic, *e.g.*, on Route 3, did not lead to evacuation traffic congestion. Two notable points beyond the plume EPZ which could cause congestion are Route 3 at Route 128 and Route 3 at the Sagamore Bridge. These points could lead to larger ETEs than those now used. The NRC staff reviewed the ETEs now used while reviewing the construction permit application for Pilgrim, Unit 2, and has now determined that this matter should be specifically brought to the attention of FEMA for its consideration in the review of ETEs for the Pilgrim facility. Consequently, this matter was referred to FEMA on January 20, 1984 for consideration and my staff has requested a response from FEMA by March 30, 1984. Therefore I am deferring resolution of this part of the Petition until after I receive FEMA's response.

I see no adequate reason to suspend operation of the Pilgrim facility pending this response. The overall state of emergency preparedness is adequate. No deficiencies which would lead to a negative finding on preparedness have been identified by FEMA. The sole remaining issue is the adequacy of ETEs for planning an emergency evacuation. The Commission has adopted an approach to emergency planning in which evacuation is only one of several possible responses to an emergency. *See* NUREG-0654, NUREG-0396 and 10 C.F.R. § 50.47(b)(10). It is unlikely that evacuation of the entire plume EPZ would be required in the event of an accident. Pending a FEMA determination on the adequacy of the ETEs, it is reasonable to conclude that the public health and safety will be reasonably assured in the interim by continued licensee compliance with Commission requirements regarding emergency planning and other health and safety requirements aimed at keeping the

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[9] In its November 3, 1983 report, FEMA notes that current NRC studies related to accident source terms, probabilities, and consequences are expected to result in a revision to NUREG-0654, which could lead to reconsideration of existing EPZ requirements. Current NRC proposals include a graduated response capability within the present EPZ, involving additional requirements for predetermined prompt actions within the first few miles of the reactor. The NRC is not considering at this time altering the overall size of the EPZ.

probability of serious accidents very low.[10] *Cf. Consolidated Edison Co. of New York* (Indian Point, Unit No. 2), CLI-83-16, 17 NRC 1006 (1983).

In view of the overall adequacy of emergency preparedness for Pilgrim and the low likelihood that an evacuation would be required as a response in the event of a radiological emergency at Pilgrim, Petitioner's requests that the NRC (1) issue a finding that the state of emergency preparedness at Pilgrim does not provide reasonable assurance that protective measures can and will be taken in the event of a radiological emergency, (2) suspend operation of the plant or order operation at reduced power, or (3) start the 4-month time period for correction of deficiencies are denied at this time.

## CONCLUSION

In summary, both onsite and offsite emergency preparedness at the Pilgrim facility have been given continued review by both the NRC and FEMA. Onsite preparedness has been determined to be adequate based upon direct NRC evaluation of the licensee's emergency planning capabilities and based on the results of the continuing inspection program in this area conducted by Region I of the NRC. Offsite emergency preparedness has been reviewed by FEMA and it has been found that offsite plans are adequate and capable of being implemented. The most recent examination of offsite emergency preparedness by FEMA specifically considered the allegations raised by Petitioner and specifically found continued assurance of the adequacy of offsite emergency preparedness to protect the public health and safety. Consequently, I conclude that the overall state of emergency preparedness at the Pilgrim facility is sufficient to assure the public health and safety while the remaining issue of Evacuation Time Estimates is considered by FEMA.

Accordingly, the Petition's request for action pursuant to 10 C.F.R. § 2.206 has been denied in part and deferred in part as described in this decision. Once FEMA provides the Commission with its findings regarding Evacuation Time Estimates, the staff will provide the Petitioner with a copy of FEMA's evaluation and will inform the Petitioner of the staff's decision as to whether further action should be taken.

[10] On December 10, 1983, the Pilgrim facility was shut down for inspection of pipe cracking in the recirculation system and for replacement of defective pipes. It is anticipated that the facility will be shut down for approximately 6 months. This should enable the staff to resolve the issue of the adequacy of the ETEs prior to plant start-up.

As provided by 10 C.F.R. § 2.206(c), a copy of this decision will be filed with the Secretary for the Commission's review.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 27th day of February 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**COMMISSIONERS:**

**Nunzio J. Palladino, Chairman**
**Victor Gilinsky**
**Thomas M. Roberts**
**James K. Asselstine**
**Frederick M. Bernthal**

**In the Matter of** **Docket No. 50-289-SP**

**METROPOLITAN EDISON COMPANY**
**(Three Mile Island Nuclear Station, Unit No. 1)** **March 28, 1984**

In response to an Appeal Board memorandum (ALAB-724, 17 NRC 559 (1983)), concerning the treatment to be accorded the issues raised in a Board Notification (BN-83-47), the Commission decides that the issue of whether the power-operated relief valve should be safety-grade, because of the potential for using it to mitigate the consequences of design basis steam generator tube accidents, has no reasonable nexus to the TMI-2 accident and is, therefore, outside the scope of the proceeding. The Commission also decides that the information in the Board Notification is not significant enough to warrant reopening the record *sua sponte,* even if it were within the scope of the proceeding.

## TECHNICAL ISSUE DISCUSSED

Uses of power-operated relief valve in depressurization in the event of a steam generator tube rupture.

# MEMORANDUM AND ORDER

On August 5, 1983, the Commission took review of that aspect of ALAB-724 (17 NRC 559 (1983)) which concerned Board Notification (BN) 83-47[1] and whether the power-operated relief valve (PORV) should be safety-grade because of the potential for using it to mitigate the consequences of design basis steam generator tube rupture (SGTR) accidents.[2] In particular, the Commission requested the parties to brief (1) whether the concerns raised by BN-83-47 are outside the scope of the TMI-1 adjudicatory proceeding, and (2) whether, if they are within the scope of the proceeding, the information contained in BN-83-47 warrants reopening the record *sua sponte*. The Union of Concerned Scientists (UCS), the licensee and the NRC staff filed briefs addressing these two questions. As explained below, the Commission has concluded that the concerns raised by BN-83-47 are outside the scope of the restart proceeding, and, even if they were not, that they do not warrant reopening the record.

## I. WHETHER THE CONCERNS RAISED BY BN-83-47 ARE WITHIN THE SCOPE OF THE RESTART PROCEEDING

### A. The Scope of the Restart Proceeding

The Commission, in the order establishing the restart proceeding, set forth the subjects to be considered at the hearing as follows:

[1] In BN-83-47 the staff concluded that PWRs need "a capability for rapid primary system depressurization . . . in order to effectively mitigate the design basis steam generator tube rupture accident" by terminating long-term releases to the environment, and that "the components and systems to provide this depressurization capability should be safety-grade." As staff explained, where reactor coolant pump (RCP) flow is lost, Westinghouse and B&W plants rely on the pressurizer PORVs to accomplish depressurization, and the pressurizer PORVs in most plants are not safety-grade. Although staff further indicated that the depressurization function could be accomplished by the auxiliary pressurizer spray system, staff stated in BN-83-47 that the spray system is not safety-grade at TMI and, regardless, "the pressurizer PORV would be the preferred means of depressurizing the [reactor coolant system]." Staff qualified its analyses by indicating that these requirements only applied to reactors undergoing OL review and that the significance of BN-83-47 and possible corrective actions for operating reactors is yet to be determined.

Subsequently, in BN-83-110 staff clarified its position by indicating that there may be means short of meeting safety-grade design criteria for the PORV by which a licensee may justify the acceptability of its depressurization capability in the face of a design basis steam generator tube rupture event.

[2] The Appeal Board in ALAB-724 identified to the Commission two safety concerns which it believed were outside the scope of the proceeding: (1) whether the PORV should be safety-grade because of the potential for using it to mitigate the consequences of design basis SGTR accidents; and (2) a corrosion problem with the PORVs that could result in the PORV not functioning when needed.

> (1) Whether the "short-term actions" recommended by the Director of Nuclear Reactor Regulation (set forth in Section II of this Order) are necessary and sufficient to provide reasonable assurance that the Three Mile Island Unit 1 facility can be operated without endangering the health and safety of the public, and should be required before resumption of operation should be permitted.
> (2) Whether the "long-term actions" recommended by the Director of Nuclear Reactor Regulation (set forth in Section II of this Order) are necessary and sufficient to provide reasonable assurance that the facility can be operated for the long term without endangering the health and safety of the public, and should be required of the licensee as soon as practicable.

CLI-79-8, 10 NRC 141, 148 (1979).

The Licensing Board discussed the scope of the proceeding in its first special prehearing conference order, LBP-79-34, 10 NRC 828 (1979). In that order the Licensing Board rejected the licensee's argument that it could consider only the individual factual issues expressly stated in the Commission's August 9 order, or in the documents referenced in that order. The Licensing Board did agree with the licensee "that the Commission did not mean to encompass in this proceeding all of the lessons which have been, or some day may be learned from the TMI-2 accident." *Id.* at 830. The Licensing Board also rejected the argument of several intervenors that any issue pertaining to health and safety could be appropriately litigated in the hearing.

The Licensing Board found that it could accept either UCS' or the staff's view on the scope of the proceeding as reasonable. Staff had suggested that the scope should be governed by whether there was a clear and close analogue and/or some reasonable nexus between the issue sought to be raised and the TMI-2 accident. UCS had argued that the test was whether the issue raised can be related to both the TMI-2 accident and whether TMI-1 can be safely operated without posing an undue threat to the public health and safety.

Without explicitly setting forth a standard, the Board in its prehearing conference order noted that the problem was in applying the test once it is defined. The Board went on to discuss proposed contentions which started from an example related to the TMI-2 accident and from there sought to enlarge the scope of the example to embrace all possibilities in the class of events or circumstances represented by the example. For those contentions the Board stated the following:

> This class of contentions has been difficult to evaluate. On one hand we do not expect intervenors now to be able to specify each circumstance related to the TMI-2 accident which should be considered, nor do we believe that only these system components alleged to have contributed directly to the accident may now be considered. On the other hand, practical evidentiary considerations and due process require that there be some reasonable bounding of the example-type contentions.

Frequently we have permitted a broadening of the contention to include the class of system components in the major safety system involved, most often the core cooling system and the containment isolation system. However, intervenors must be aware that this broadening may not produce the showing sought by the contention. The specificity of the contention will necessarily shape the specificity of the evidence produced in response. The discovery process should be used to refine these contentions so that only those circumstances reasonably related to the accident are identified for hearing.

10 NRC at 832.

The Commission in subsequent orders confirmed that the restart proceeding was not limited to the issues set forth in the original order (*see, e.g.*, CLI-80-16, 11 NRC 674), and that the test for admissibility was whether "there is a reasonable nexus between the issue and the TMI-2 accident." Unpublished order of March 14, 1980. This standard that contentions must have a reasonable nexus to the TMI-2 accident was repeated throughout the proceeding (*see, e.g.*, 14 NRC at 394), and was interpreted to include safety questions having a nexus to a small-break LOCA or a loss of main feedwater. *See, e.g.*, ALAB-729, 17 NRC 814, 822 n.7 (1983). The Licensing Board explained that this standard "is based on the facts that TMI-1 was reviewed and approved at the operating license stage and that, but for the accident, we would not be involved in this particular proceeding." 14 NRC at 1730.

## B. Parties' Positions

All parties agree that the standard for whether the issues raised by BN-83-47 are within the proceeding is whether they have a reasonable "nexus" to the TMI-2 accident. The parties disagree, however, on what constitutes a reasonable nexus to the accident and on whether the information in BN-83-47 has such a nexus.

### 1. *Union of Concerned Scientists (UCS)*

UCS argues that the material in BN-83-47 is within the proceeding under two key interrelated theories.[3] First, UCS states that the nexus standard is not limited to small-break LOCAs and loss-of-main-feedwater transients, rather that the nexus standard includes the lessons learned from the accident. UCS maintains that one such lesson learned

[3] UCS in addition to its two main theories also argues: (1) It is not clear that the TMI-2 accident did not involve an SGTR; and (2) the scope of the restart proceeding included consideration of the capability to limit doses to ensure compliance with 10 C.F.R. Part 100 criteria, the subject of BN-83-47.

concerns the role of equipment previously treated as unrelated to safety in the causation and mitigation of accidents, and that under this lesson the PORV should be safety-grade. UCS argues that BN-83-47 falls within this lesson learned and thus is within the proceeding.

*Second*, UCS maintains that BN-83-47 is within the proceeding because it relates to the three contexts in the restart proceeding in which the issue of whether the PORV needs to be safety-grade was litigated. The first context cited by UCS is its Contention 5.[4] UCS argues that the nexus of Contention 5 to the accident was the lesson learned that systems previously considered unrelated to safety do perform safety functions. In this connection UCS argued there were six functions of the PORV, two used to control coolant pressure, that required that it be safety-grade. In UCS' view, the information in BN-83-47 relates to a seventh function of the PORV requiring that it be safety-grade.

The second context cited by UCS concerns whether "feed and bleed" is an acceptable means of cooling the core. UCS attempted in the proceeding to show the pressurizer safety valves cannot be used to "bleed" in the event of an SGTR, and therefore that the PORV must be safety-grade.

The third context discussed by UCS involves UCS Contention 14, which asserted that systems which can be called upon to mitigate accidents should be required to meet safety-grade criteria.[5] UCS states that staff's position in response to this contention was that only systems or components required to perform critical functions — *e.g.*, to shut down the reactor or mitigate accidents — need be safety-grade. UCS maintains that BN-83-47 discloses that the PORV must be safety-grade because it is required to perform both of these functions, and, consistent with staff's own testimony, the Licensing Board therefore would have had to require that the PORV be safety-grade if BN-83-47 had been before the Board.

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[4] UCS Contention 5 in part stated:

> Proper operation of power-operated relief valves, associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria.

[5] UCS Contention 14 in part read as follows:

> The accident demonstrated that there are systems and components presently classified as non-safety-related which can have an adverse effect on the integrity of the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity. . . . The Staff proposed to study the problem further. This is not a sufficient answer. All systems and components which can either cause or aggravate an accident or can be called upon to mitigate an accident must be identified and classified as components important to safety and required to meet all safety-grade criteria.

### 2. *Licensee*

Licensee maintains that the reasonable nexus standard means having a nexus to an accident involving a loss of main feedwater or small-break loss-of-coolant accident. Licensee argues that SGTR accidents and the need for rapid primary system depressurization capability, the subjects of BN-83-47, are unrelated to the accident. Licensee states that the only issues relating to the operability of the PORV raised in connection with the accident were concerned with the role of the PORV in causing, aggravating and mitigating a small-break LOCA. The capability to depressurize rapidly so as not to exceed Part 100 dose criteria is not among the PORV concerns which rose from the TMI-2 accident.

Licensee notes that no party raised a contention on the adequacy of the PORV or any other components or systems to mitigate SGTR events, and UCS did not advance SGTR events as a basis for its Contention 5. Licensee further notes that the Licensing Board accepted staff's analysis of event sequences as including all sequences with a nexus to the TMI-2 accident, and that analysis did not include SGTR events.

Finally, Licensee, citing *Rulemaking Hearing, Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors,* CLI-73-39, 6 AEC 1085, 1087 (1973), argues that SGTR accidents have not been treated as small-break loss-of-coolant accidents.

### 3. *Staff*

Staff asserts that no party in its contentions sought to raise any issue related to an SGTR accident. Staff further asserts that the restart proceeding included only issues related to the capability to cool the core adequately following a small-break LOCA and/or main feedwater transient, and that the concern of BN-83-47 at bottom is a concern about potential offsite doses that may result from failure to limit primary-to-secondary coolant loss through timely termination of the leakage.

## C. Analysis

The restart proceeding is an enforcement proceeding which is being held because of the TMI-2 accident. The purpose of this proceeding is not to litigate the overall safety of TMI-1, but rather to resolve questions arising from the accident. Thus, the issue presented here is whether the concerns raised by BN-83-47 have a reasonable nexus to the TMI-2 accident.

The basic concern in BN-83-47 is restricting primary-to-secondary leakage in the steam generator in order to avoid exceeding Part 100 dose criteria in the event of an SGTR; BN-83-47 does not address the possibility of uncovering the core or of leakage into the containment. Although the restart proceeding did not necessarily exclude all issues involving the potential for offsite leakage,[6] the Commission also does not believe it necessarily included every potential scenario for offsite releases which would result from any type of small-break LOCA or loss-of-main-feedwater transient. In each case the path potentially leading to offsite doses must have a reasonable nexus to the accident, and there is no reasonable nexus between the TMI-2 accident and primary-to-secondary leakage in the steam generator due to an SGTR.[7] The TMI-2 accident involved a main feedwater transient followed by a small-break LOCA, in particular, a stuck-open PORV, leading to leakage into the containment and partial uncovering of the core. The concerns in BN-83-47 are unrelated to this sequence of events.[8]

UCS nonetheless argues that BN-83-47 relates to contentions in the proceeding and therefore that it is within the scope of the proceeding. UCS argues that BN-83-47 relates to Contentions 5 and 14 and to the feed-and-bleed issue.[9] BN-83-47 on its face may appear to relate to UCS Contention 5, which concerned whether the PORV should be safety-grade. However, UCS did not argue that the PORV should be safety-

[6] For instance, one contention that was resolved by the Licensing Board concerned initiation of containment isolation. Prior to the accident, TMI-1 and -2 were designed so that containment isolation occurred following receipt of a containment building high-pressure signal. The accident raised concerns that significant fuel damage can occur in the absence of containment high pressure, and hence the NRC staff required diversity in the parameters sensed for the initiation of isolation. Therefore initiation of containment isolation clearly has a reasonable nexus to the accident.

[7] The NRC memorandum cited by UCS to show that there was primary-to-secondary leakage in Steam Generator B during the accident indicates that this leakage was not due to an SGTR.

[8] The Commission also notes that not all accidents generically labelled "small-break LOCAs" have similar characteristics. For instance, SGTRs should be distinguished from LOCAs in the bypass system, and both should be distinguished from LOCAs in the main primary system piping. The differences between an SGTR and other types of small-break LOCAs include the following: An SGTR does not involve loss of primary coolant directly to the containment; for the same break area an SGTR does not result in as rapid a depressurization; an SGTR is better mitigated with reactor coolant pumps running; and, within the design basis, single tube failure, an SGTR does not challenge the containment.

[9] UCS Contention 14 dealt with systems classification and interaction, and staff testified in that regard that components necessary for mitigation purposes should be safety-grade. The Licensing Board accepted this definition, thus resolving Contention 14. UCS is now apparently asserting that under the staff's definition the information in BN-83-47 requires that the PORV be safety-grade. Staff in BN-83-47 did not state that use of the PORV was necessary to mitigate an SGTR, and hence UCS' interpretation is incorrect. In addition, Contention 14 did not address how the safety-grade standard should be applied for each piece of equipment in the plant. To hold otherwise would mean that parties under Contention 14 could litigate whether each piece of equipment in the plant needed to be safety-grade, regardless of any nexus to the accident.

UCS also argues that BN-83-47 relates to the feed-and-bleed issue. The only arguable connection between the two is that both concern use of the PORV. However, the potential use of the PORV to "bleed" bears no relationship to using the PORV to depressurize in the event of an SGTR.

grade because of the potential for using it to depressurize in the event of an SGTR. That some uses of the PORV were litigated in the proceeding does not mean that all potential uses of the PORV are within the proceeding.[10] The issue in each instance is whether the postulated scenario has a reasonable nexus to the TMI-2 accident. As explained above, the information in BN-83-47 does not have such a nexus.[11]

## II. WHETHER INFORMATION IN BN-83-47 WARRANTS REOPENING THE RECORD

### A. Parties' Positions

Staff and licensee argue that the information in BN-83-47 does not warrant reopening the record; UCS argues that it does.

UCS argues that the information in BN-83-47 is new and significant, and therefore warrants reopening the record. UCS states that consideration of the information in BN-83-47 regarding the need for rapid depressurization of the primary system in the event of an SGTR would have led the Licensing Board to reach a different result relative to design requirements for the PORV in order to be consistent with the staff's testimony regarding UCS Contention 14.

Licensee argues that reopening based on BN-83-47 would inject a new issue into the restart proceeding, and that the standard for such action must be "that a serious new safety concern exists which would warrant the immediately effective suspension of this operating license." Licensee concludes that BN-83-47 does not raise a safety concern meeting this standard.

Staff maintains the criteria for reopening must be met by the information in BN-83-47 itself, that other items in the proceeding regarding the PORV do not affect whether the information in that Board Notification warrants reopening. Staff states that the Commission should not *sua sponte* reopen unless

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[10] A contrary interpretation could lead to a never-ending proceeding. Under UCS' view, use of the PORV to mitigate any type of accident would fall within Contention 5, even if it had no conceivable connection with the TMI-2 accident. This would be inconsistent with the purposes of this proceeding.

[11] The Commission recognizes that in some instances the Licensing Board allowed contentions based on an example related to the accident to be broadened to include the class of systems components in the major safety system involved, and that arguably under that rationale Contention 5 could be read to include the concerns in BN-83-47. However, the Licensing Board recognized that this type of expansion was not required, and the Commission does not believe Contention 5 should be considered to include uses of the PORV beyond the defined scope of the proceeding.

(1) it can make an affirmative finding that the information in BN-83-47 demonstrates the existence of a problem that presents a grave threat to public safety, or (2) it believes that omission of the information in BN-83-47 from the evidentiary record would leave the record materially inaccurate or incomplete on a contested issue admitted to the restart proceeding, or (3) the information in BN-83-47 has created serious doubts about the correctness of a decision below on a contested issue admitted to the restart proceeding and the Commission believes the taking of supplementary evidence is required to resolve those doubts.

Staff maintains that the information in BN-83-47 does not rise to this level.

## B. Analysis

In the Commission's view, the material in BN-83-47, as clarified in BN-83-110, does not warrant reopening the record *sua sponte*. BN-83-47 represents the preliminary view of the Division of Systems Integration (DSI) that pressurized water reactors undergoing licensing review should have a safety-grade means for depressurizing the primary system in the event of an SGTR. BN-83-47, which indicates that DSI is evaluating actions that may need to be taken on operating reactors is only a statement of a Division position. Thus, it does not necessarily represent even the view of the staff as a whole. Indeed, the issue is a generic one the resolution of which must therefore be reviewed by the Committee to Review Generic Requirements. The resolution must also be approved by the Executive Director of Operations. Further, BN-83-47 does not indicate that a safety-grade PORV is the only acceptable means for rapidly depressurizing the primary system, nor does it conclude that offsite doses would exceed Part 100 criteria in the event of an SGTR without such capability. Indeed, in BN-83-110 staff clarified that there may be means short of meeting safety-grade design criteria for the PORV by which a licensee may justify the acceptability of its depressurization capability in the face of a design basis SGTR event. It is the Commission's judgment that this preliminary viewpoint of the staff does not demonstrate an unacceptable risk to the public health and safety and does not raise an issue significant enough to warrant reopening the record *sua sponte*, even if it were within the scope of the proceeding.

Commissioner Gilinsky dissents from this decision. His views are attached.

It is so ORDERED.

For the Commission*

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 28th day of March 1984.

## SEPARATE VIEWS OF COMMISSIONER GILINSKY (SECY-84-48, REVIEW OF ALAB-724)

Whether to require the pressurizer power-operated relief valves to be safety-grade is an important safety question as these valves may be relied upon to depressurize the primary system in several types of accidents. The Commission should have decided the merits of this question, instead of dismissing it on legalistic jurisdictional grounds.

*Commissioner Gilinsky was not present when this order was affirmed, but had previously indicated his disapproval.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Alan S. Rosenthal,** Chairman
**Gary J. Edles**
**Howard A. Wilber**

**In the Matter of** **Docket Nos. 50-443-OL**
**50-444-OL**

**PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE,** ***et al.***
**(Seabrook Station, Units 1 and 2)** **March 16, 1984**

Finding the standard for interlocutory review of a licensing board ruling not met, the Appeal Board denies an intervenor's request for directed certification of the Licensing Board's denial of its motion for dismissal of the operating license application for Unit 2 of the Seabrook facility sought on the ground that that Unit is only 22 percent completed.

## RULES OF PRACTICE: DIRECTED CERTIFICATION

In the exercise of its directed certification authority conferred by 10 C.F.R. 2.718(i), an appeal board will step into a proceeding still pending below only upon a clear and convincing showing that the licensing board ruling under attack either (1) threatens the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated by a later appeal or (2) affects the basic structure of the proceeding in a pervasive or unusual manner. *Arizona Public Service Co.* (Palo Verde Nuclear Generating Station, Units 2 and 3), ALAB-742, 18 NRC 380, 383 (1983); *Public Service Co. of Indiana*

(Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977).

**OPERATING LICENSE PROCEEDINGS: TIMELINESS OF LICENSE APPLICATION**

The Commission's regulations are devoid of any specific requirement that the reactor reach a particular stage of completion before the filing of an operating license application.

## APPEARANCES

**Robert A. Backus**, Manchester, New Hampshire, for the intervenor, Seacoast Anti-Pollution League.

**Thomas G. Dignan, Jr.**, and **R.K. Gad III**, Boston, Massachusetts, for the applicants, Public Service Company of New Hampshire, *et al.*

**William F. Patterson, Jr.**, for the Nuclear Regulatory Commission staff.

# MEMORANDUM AND ORDER

1. This operating license proceeding involves Units 1 and 2 of the Seabrook nuclear facility. Two months ago, we upheld the Licensing Board's denial of an untimely petition for leave to intervene in the proceeding.[1] That petition had put forth a single claim: given the asserted fact that Unit 2 is only 22 percent completed, the application for an operating license for that unit is premature and, as a matter of law, should be turned down for that reason.

Our affirmance of the Licensing Board's rejection of the intervention petition rested not on the lateness of the hour *per se* but, rather, on another consideration. Examination of the record below revealed that, shortly after the petition surfaced, one of the existing parties to the proceeding — the Seacoast Anti-Pollution League (SAPL) — had filed a motion seeking the same relief respecting Unit 2 (dismissal of the

[1] ALAB-758, 19 NRC 7 (1984).

operating license application) on the strength of precisely the same prematurity claim. In common with the tardy petitioner, SAPL had relied upon 10 C.F.R. 50.57(a)(1), which requires, as a precondition to the issuance of an operating license, a finding that:

> Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the [Atomic Energy] Act, and the rules and regulations of the Commission.

In an unpublished memorandum and order issued on January 13, 1984, the Licensing Board ruled against SAPL's position and accordingly denied the motion.

In the totality of these circumstances, we concluded:

> events clearly have overtaken [the tardy petitioner's] intervention effort. As matters now stand, his objective of having the Unit 2 prematurity issue placed before the Licensing Board has been achieved — albeit through the endeavors of someone else. True, on the SAPL motion the Board determined the issue against [the tardy petitioner's] position. There is no reason to suppose, however, that the Board would have decided it any differently had it considered his claim rather than SAPL's.[2]

In this connection, we went on to note our belief that, "[a]t the very least, the Licensing Board's analysis of the Unit 2 prematurity question in its January 13 memorandum and order is not manifestly (or even probably) erroneous."[3] By way of elaboration, we said that "this much is clear":

> First, the Licensing Board correctly held that it is not its responsibility, but that of the Director of Nuclear Reactor Regulation, to make the finding required by Section 50.57(a)(1) as a precondition to the issuance by the Director of an operating license. *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-226, 8 AEC 381, 410-11 (1974). Second, there is nothing in the Commission's regulations specifically providing that a reactor must have reached a particular stage of completion before an operating license application may be filed. Third, just 16 months ago the Commission denied a petition for rulemaking that sought amendments to the Rules of Practice that would have, *inter alia*, limited the scope of each operating license hearing to a single reactor unit even if that unit were one of several similar units constructed on a multi-reactor site. 47 Fed. Reg. 46,524 (1982). In support of his proposal, the petitioner had noted that the "time lag between inservice dates for individual reactors at multi-reactor nuclear plants has been increasing for many years." *Ibid.* In the Commission's view, however, that consideration did not provide a sufficient basis for requiring "an exclusive hearing on each reactor unit." *Id.* at 46,525.[4]

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[2] *Id.* at 11.

[3] *Id.*

[4] *Id.* at 11 n.18.

With only the most fleeting acknowledgement of these determinations,[5] SAPL now asks that we review the denial of its motion to dismiss in the exercise of the discretionary directed certification authority conferred by 10 C.F.R. 2.718(i).[6] We are told by SAPL that the necessary effect of the rejection of the prematurity claim is that the public (including itself) will be denied a fair hearing with respect to Unit 2. This is said to be so because (1) it is not possible to put forth contentions at this juncture with respect to vital safety systems not as yet installed in Unit 2; and (2) by reason of the delay in the construction of the unit, this proceeding is likely to be at an end before issues pertaining to that construction could be raised and litigated.[7]

In response, the applicants and the NRC staff maintain that the established standards for directed certification are not met here and that, in any event, the Licensing Board's ruling was correct.[8]

2. We have often had occasion to stress that

> interlocutory appellate review of licensing board orders is disfavored and will be undertaken as a discretionary matter only in the most compelling circumstances. More specifically, in the exercise of our directed certification authority conferred by 10 C.F.R. 2.718(i), we will step into a proceeding still pending below only upon a clear and convincing showing that the licensing board ruling under attack either
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> > (1) threaten[s] the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated by a later appeal or (2) affect[s] the basic structure of the proceeding in a pervasive or unusual manner.[9]

SAPL does not assert here that it is threatened with serious immediate harm that could not be remedied on an appeal from the Licensing Board's eventual initial decision. Instead, it invokes the second criterion

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[5] SAPL Appeal of Denial of Motion to Dismiss Application for Seabrook Unit 2 (February 17, 1984) at 11.

[6] *See Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 482-83 (1975). Although the term "appeal" appears in the caption, it is plain from the body of its filing that SAPL is invoking Section 2.718(i). In light of the general proscription against interlocutory appeals found in 10 C.F.R. 2.730(f), any claim of a right to have the challenged Licensing Board ruling reviewed at this juncture would have been unavailing.

[7] SAPL Appeal, *supra* note 5, at 1-7. In this regard, SAPL asserts that construction of Unit 2 has been "effectively suspended," with the consequence that there is no current completion date for that unit. *Id.* at 2-3. On March 1, 1984, however, counsel for applicants sent a letter to the Licensing Board to the effect that a newly established "interim schedule" projects fuel loading of Unit 2 on July 31, 1990.

[8] Applicant's Response to "SAPL Appeal of Denial of Motion to Dismiss Application for Seabrook Unit 2" (March 5, 1984); NRC Staff Response in Opposition to "SAPL Appeal of Denial of Motion to Dismiss Application for Seabrook Unit 2" (March 8, 1984).

[9] *Arizona Public Service Co.* (Palo Verde Nuclear Generating Station, Units 2 and 3), ALAB-742, 18 NRC 380, 383 (1983) (footnotes omitted). As indicated therein, the genesis of the quoted two-prong test was *Public Service Co. of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977).

alone.[10] It is plain upon analysis, however, that the inclusion of Unit 2 does not affect the basic structure of this proceeding in a pervasive or unusual manner.

As we observed in ALAB-758, the Commission's regulations are devoid of any specific requirement that the reactor reach a particular stage of completion before the filing of an operating license application. *See* p. 567, *supra*. This being so, it is hardly surprising that, over the years, the Commission has instituted and carried forward numerous operating license proceedings encompassing two or more units in quite different stages of completion.[11]

Further, we find not objectionable the practice of considering in a single proceeding those issues common to all units of a multi-unit facility. Indeed, the practice seems to us to make very good sense. In the proceeding at bar, many common issues have already been tried or will be heard at a future evidentiary session: *e.g.*, control room design, equipment environmental qualification, and various aspects of onsite and offsite emergency planning.[12] We know of no useful purpose that would be served by now resolving these issues for Unit 1 alone and then replowing the exact same ground at some later date in the context of Unit 2.

If we apprehend its position correctly, SAPL does not suggest otherwise. There is not a word of complaint in its appellate papers respecting the scope of the issues that are currently being explored in the proceeding. Rather, SAPL's concern appears to lie in another direction. As previously noted, its focus is the obvious present lack of opportunity to advance contentions related to the quality of as yet uncompleted Unit 2 construction and the possibility that the proceeding might come to an end before such an opportunity became available. *See* p. 568, *supra*.

That may well be a legitimate concern. And, if so, SAPL might have some basis for insisting that, with respect to Unit 2, the evidentiary record in this proceeding not be closed until after construction of that unit is much further advanced than it is today. But we need not — and do not — pass judgment upon that question at this time. For one thing,

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[10] SAPL Appeal, *supra* note 5, at 1.

[11] Among the several such proceedings currently in progress (in addition to the one at bar) are the three involving the two-unit Limerick, Perry, and Vogtle facilities. Docket Nos. 50-352 and 50-353 (Limerick); 50-440 and 50-441 (Perry); and 50-424 and 50-425 (Vogtle).

It is our understanding that (1) the two Limerick units are approximately 90% and 30% built and that there is at least a five-year differential in their current projected completion dates; (2) the two Perry units are approximately 92% and 42% built with a several year differential in their projected completion dates; and (3) the two Vogtle units are approximately 62% and 25% built with a similar differential in their projected completion dates.

[12] As the applicants stress, the designs of Units 1 and 2 are, for all practical purposes, identical. Applicants' Response, *supra* note 8, at 6.

to date no such relief has been explicitly sought by SAPL. For another, should SAPL move below to hold the record open on Unit 2 to await the substantial completion of construction of that unit, there will be time enough for the Licensing Board to act upon the motion when it is ready to close the record on Unit 1. If aggrieved by the Board's determination, SAPL can register its dissatisfaction on an appeal from the initial decision.

In short, there simply is no reason for our intercession on SAPL's behalf at the current stage of the proceeding. Apart from SAPL's failure to counter our determinations in ALAB-758 (*see* p. 567, *supra*), it is manifest that the inclusion of both Seabrook units in this proceeding neither represents a departure from established Commission practice nor affects the basic structure of the proceeding in a pervasive or unusual manner. Moreover, as just seen, there is another, and considerably more appropriate, avenue available to SAPL for seeking to protect its ability to put forth at a later date additional contentions associated with Unit 2 construction.[13]

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The Seacoast Anti-Pollution League's motion for directed certification of the Licensing Board's January 13, 1984 memorandum and order is *denied.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

[13] In addition to its motion to dismiss the application for an operating license for Unit 2, SAPL submitted an untimely contention advancing the same prematurity claim found in the motion. In its January 13 memorandum and order denying the motion, the Licensing Board also rejected the late-filed contention. The reasons we have assigned for not undertaking an interlocutory review of the denial of the motion apply equally to the rejection of the contention.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Thomas S. Moore,** Chairman
**Dr. John H. Buck**
**Dr. W. Reed Johnson**

**In the Matter of** | **Docket Nos. 50-275**
**50-323**

**PACIFIC GAS AND ELECTRIC COMPANY**
**(Diablo Canyon Nuclear Power Plant, Units 1 and 2)**

**March 20, 1984**

Following the conduct of evidentiary hearings by the Appeal Board on the adequacy of the applicant's efforts to verify the design of the Diablo Canyon facility, the Appeal Board decides that the actions taken by the applicant provided adequate confidence that Unit 1's structures, systems and components are designed to perform satisfactorily in service and that any significant design deficiencies in that unit resulting from defects in the applicant's design quality assurance program have been remedied. The Appeal Board thus concludes that there is reasonable assurance that Unit 1 can be operated without endangering the health and safety of the public.

The Appeal Board withholds decision with respect to the adequacy of the design verification program for Unit 2.

## RULES OF PRACTICE: BURDEN OF PROOF

In order for the applicant to prevail on each factual issue, its position must be supported by a preponderance of the evidence. *See Tennessee*

*Valley Authority* (Hartsville Nuclear Plant, Units 1A, 2A, 1B, and 2B), ALAB-463, 7 NRC 341, 360 (1978), *reconsideration denied,* ALAB-467, 7 NRC 459 (1978); *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), ALAB-355, 4 NRC 397, 405 n.19 (1976).

## PLANT DESIGN: STANDARD FOR DETERMINING ADEQUACY

To determine that an applicant's verification programs are sufficient to verify the adequacy of a plant's design, the applicant's efforts must be measured against the same standard as that set forth in the Commission's quality assurance criteria, 10 C.F.R. Part 50, Appendix B: whether the verification program provides "adequate confidence that a [safety-related] structure, system or component will perform satisfactorily in service." If the applicant's verification efforts meet this standard, then there will be reasonable assurance with respect to the design of the facility that it can be operated without endangering the health and safety of the public.

## QUALITY ASSURANCE/QUALITY CONTROL: DOCUMENTS

The Commission's regulations do not require that all pertinent quality assurance or quality control documents be consolidated and integrated into a single manual or set of manuals.

## TECHNICAL ISSUES DISCUSSED

Sampling Techniques (statistical and judgmental) and Scope;
Instrument Tubing Supports;
Containment Uplifting;
Modeling for Seismic Analysis (including the use of soil springs, fixed-base analysis, response of one building as imput into model of another, lumped mass-spring model, finite element models, degrees of freedom);
Soil Analysis (Seismic Refraction Tests and Cross-hole; and Up-hole Testing Techniques);
Seismic Response Spectra;
Fire Protection;
Jet Impingement Analysis;
Circuit Breakers (nameplate rating);
Design Drawings and Analyses (conformance with plant as built);
Component Cooling Water System Heat Removal Capacity;

Small Bore Piping and Support Design (computer-based analysis and span criteria);
Design Error Rate (adequate confidence versus perfection);
Hosgri Fault;
Westinghouse Quality Assurance Program;
Causes of Quality Assurance Failures.

## APPEARANCES

**Joel R. Reynolds, John R. Phillips** and **Eric R. Havian,** Los Angeles, California, and **David S. Fleischaker,** Oklahoma City, Oklahoma, for the San Luis Obispo Mothers for Peace, *et al.,* joint intervenors.

**John K. Van De Kamp,** Attorney General of the State of California, **Andrea Sheridan Ordin, Michael J. Strumwasser, Susan L. Durbin** and **Peter H. Kaufman,** Los Angeles, California, for George Deukmejian, Governor of the State of California.

**Robert Ohlbach, Philip A. Crane, Jr., Richard F. Locke** and **Dan G. Lubbock,** San Francisco, California, and **Arthur C. Gehr, Bruce Norton** and **Thomas A. Scarduzio, Jr.,** Phoenix, Arizona, for Pacific Gas and Electric Company, applicant.

**Lawrence J. Chandler** and **Henry J. McGurren,** for the Nuclear Regulatory Commission staff.

# DECISION

On April 21, 1983, we granted the motions of the joint intervenors and the Governor of California to reopen the record in this operating license proceeding. Instead of remanding to the Licensing Board for that purpose, we acquiesced in the request of the parties that we hear the further evidence ourselves. This decision sets forth our findings of fact and conclusions of law based upon that evidence.

## I. HISTORY OF PROCEEDING

A. In July 1981, the Licensing Board issued a partial initial decision authorizing the Director of Nuclear Reactor Regulation to issue a license to the Pacific Gas and Electric Company (PG&E or applicant) to load fuel and to conduct low power tests up to five percent of rated power at its Diablo Canyon Nuclear Power Plant, Units 1 and 2.[1] After the Commission's favorable immediate effectiveness review for Unit 1 (conducted pursuant to 10 C.F.R. 2.764(f)),[2] the Director issued a low power license for that unit on September 22, 1981.[3]

Shortly thereafter, while preparing a response to an agency request for information, the applicant discovered errors in the assignment of seismic design spectra for equipment and piping in portions of the containment for Unit 1. These errors, combined with the identification by the NRC staff of serious weaknesses in the implementation of the applicant's quality assurance program, led the Commission to suspend conditionally the applicant's low power license. The license suspension was to remain in effect pending the applicant's satisfactory completion of an independent design verification program focusing upon the pre-1978 work of the service-related contractors utilized in the seismic design process of safety-related structures, systems and components for Unit 1.[4]

In addition to the Commission's enforcement action, the staff instructed the applicant to provide it with the results of a further independent verification program for Unit 1 to enable the staff to authorize operation above low power levels. This verification was to be aimed at the pre-June 1978 service-related contractors used by the applicant in the nonseismic design of safety-related structures, systems and components, the applicant's internal design activities, and the post-1977 service-related contractors utilized by the applicant for both seismic and nonseismic design of structures, systems and components.[5]

In order to secure reinstatement of its license and eventual authorization for full power operation, the applicant initiated a verification program to meet the Commission's order and the staff's directive. As subsequent events would reveal, the applicant's verification efforts expanded far beyond those originally envisioned and took more than two years to complete.

While the verification was ongoing, and while the joint intervenors' appeal from the Licensing Board's low power decision was pending before us, the joint intervenors, on June 8, 1982, filed a motion to reopen the record on the issue of the adequacy of the applicant's quality

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[1] *See* LBP-81-21, 14 NRC 107 (1981).
[2] *See* CLI-81-22, 14 NRC 598 (1981).
[3] License No. DPR-76.
[4] *See* CLI-81-30, 14 NRC 950 (1981).
[5] *See* Applicant Exhibit (App. Exh.) 87, letter from H. Denton, NRC, to M. Furbush, PG&E (Nov. 19, 1981).

assurance program. That motion was based essentially upon the same information that prompted the Commission's enforcement action and the various deficiencies identified by the verification program up to that time.

Besides opposing the joint intervenors' motion on the merits, the applicant claimed that the Commission's enforcement order conditionally suspending its license had divested us of jurisdiction to reopen the record. Although unpersuaded by this argument, we certified the jurisdictional question, among others, to the Commission in order to avoid any unnecessary delay in the licensing process were it ultimately to be accepted.[6] In due course, the Commission responded that it had not intended, and did not now intend, to divest the adjudicatory boards of jurisdiction to act on the motions and that they should be treated in accordance with applicable case law.[7]

We then directed the certification to us of a similar motion that had been filed by the Governor of California on August 2, 1982 with the Licensing Board.[8] After hearing argument on the motions, we concurred with the concessions of the applicant and the staff that, with respect to the issue of design quality assurance, the motions of the joint intervenors and the Governor met the standards for reopening the record.[9] Accordingly, we granted the motions on April 21, 1983.[10]

[6] *See* ALAB-681, 16 NRC 146 (1982).

In addition, we asked whether the Commission wished to relieve the adjudicatory boards of jurisdiction with regard to quality assurance issues at Diablo Canyon and whether the Commission had any other instructions with regard to the reopening motions.

[7] *See* CLI-82-39, 16 NRC 1712 (1982).

[8] *See* Order of January 5, 1983 (unpublished).

When the Governor filed his reopening motion with the Licensing Board, the Board had yet to issue its decision resolving all contested issues necessary for full power operation. Subsequently, on August 31, 1982, the Board issued its initial decision authorizing full power operation. *See* LBP-82-70, 16 NRC 756. There the Board noted that its action did not affect the applicant's license suspension and that it would hold the Governor's reopening motion in abeyance to await the answer to the jurisdictional questions previously certified to the Commission in ALAB-681. *Id.* at 760 and 763. All parties filed exceptions to the Licensing Board's initial decision and those appeals are currently pending before us. In addition, the Commission still must conduct its immediate effectiveness review of that Licensing Board decision.

[9] The motions of the joint intervenors and the Governor also sought reopening on the issue of the adequacy of the applicant's construction — as opposed to design — quality assurance program. Because of the manner in which the issue was presented, we deferred ruling on it. *See* Memorandum and Order of April 21, 1983 (unpublished). Thereafter, the joint intervenors and the Governor filed new motions to reopen the record on the construction quality assurance issue. In ALAB-756, 18 NRC 1340 (1983), we set out the reasons for denying these motions.

[10] *See* Memorandum and Order of April 21, 1983 (unpublished).

The granting of the motions to reopen the record had no effect on the Licensing Board's previously issued partial initial decision authorizing fuel loading and low power testing (LBP-81-21, 14 NRC 107 (1981)) or initial decision authorizing full power operation (LBP-82-70, 16 NRC 756 (1982)). Our action neither vacated nor stayed these decisions. We subsequently affirmed the Licensing Board's low power decision. *See* ALAB-728, 17 NRC 777 (1983). Similarly, the reopening of the record in the operating license proceeding had no effect on the Commission's enforcement action suspending the applicant's low power license.

Although the motions to reopen were predicated on deficiencies in the applicant's design quality assurance program and the applicant's failure to comply with 10 C.F.R. Part 50, Appendix B, the real issue in the proceeding quickly moved beyond that point.[11] As noted in our prehearing order of August 16, 1983,

> the history and nature of the design quality assurance issue at Diablo Canyon make this reopened proceeding unusual. Normally, an effectively functioning design quality assurance program ensures that the design of a nuclear power plant is in conformance with the design criteria and commitments set forth in an applicant's PSAR [Preliminary Safety Analysis Report] and FSAR [Final Safety Analysis Report]. In the case of Diablo Canyon, however, this confidence has been seriously eroded by the existence of significant evidence that the design quality assurance program was faulty (i.e. it failed to comply with 10 C.F.R. Part 50, Appendix B). Hence, there is now substantial uncertainty whether any particular structure, system or component was designed in accordance with stated criteria and commitments.[12]

The order then indicated we would take our lead from the Commission and permit the applicant's various verification efforts "to substitute for, and supplement, the applicant's design quality assurance program in order to demonstrate that the Diablo Canyon plant is correctly designed."[13] It concluded by stating that the "real issue . . . has, in effect, moved beyond the question of what deficiencies existed in the applicant's Diablo Canyon design quality assurance program to the question whether the applicant can demonstrate that [its verification efforts] verify the correctness of the Diablo Canyon design."[14]

Trial of the thirty-nine contested issues regarding the adequacy of the applicant's verification efforts commenced October 31, 1983 in Avila Beach, California near the reactor site and consumed fifteen hearing

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[11] Indeed, as the applicant's counsel stated at the argument on the motions to reopen,

> [w]e are willing to stipulate that there — that there are, may have been, and have been deficiencies in design QA [Quality Assurance]. That is behind us. There is no sense in litigating design QA. Where does that get anybody? It doesn't accomplish anything.

Transcript (Tr.) of April 14, 1983 oral argument at 215. *See* Order of August 16, 1983 (unpublished).

[12] Order of August 16, 1983 (unpublished) at 4-5. The analysis of the issues involved in the reopened proceeding outlined in the August 16 order was subsequently incorporated into our August 26, 1983 prehearing conference order.

[13] Order of August 16, 1983 (unpublished) at 5.

[14] *Id.* at 6.

days.[15] The applicant presented twenty-five witnesses,[16] the staff fourteen, the joint intervenors one, and the Governor three.[17] The hearing produced some 3700 pages of transcript and better than 6000 pages of exhibits. At the conclusion of the hearing, the parties were ordered, pursuant to 10 C.F.R. 2.754, to file proposed findings of fact and conclusions of law and were admonished that the failure to file proposed findings on any issue would be deemed a waiver of that issue.[18] The last of the parties' proposed findings was filed January 4, 1984. The joint intervenors and the Governor both failed to file proposed findings on sixteen issues.[19] In addition, the joint intervenors failed to file proposed findings on an issue that the Governor abandoned in his findings.[20] These issues are therefore waived, leaving twenty-two issues for resolution.[21]

In order to prevail on each of the remaining factual issues, the applicant's position must be supported by a preponderance of the evidence.[22] We do not decide, however, whether each element of the Commission's November 19, 1981 enforcement order (or other subsequent directives) has been met. That task is for the Commission itself.[23] Rather, we must

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[15] We accepted fifty-six issues of those originally sought to be litigated in the reopened proceeding by the joint intervenors and the Governor. *See* Orders of August 26 and October 7, 1983 (unpublished). Prior to the hearing, the joint intervenors and the Governor withdrew seventeen, leaving thirty-nine contested issues. *See* Withdrawal of Certain Contentions by Governor Deukmejian and Joint Intervenors (Oct. 24, 1983); Withdrawal of Certain Additional Contentions by Governor Deukmejian and Joint Intervenors (Oct. 31, 1983). As numbered in our August 26, 1983, prehearing conference order, the following issues remained at the time of the hearing: 1(a), (b), (c), (d), (e); 2(a), (b), (c), (d); 3(f)(i), (ii), (iii), (iv), (v), (o), (p), (q), (r), (s), (t); 4(a), (b), (h), (i)(1), (2), (j)(1), (2), (k), (l), (q), (r), (s), (t), (u); 5; 6; 7; 8; and 9. These issues are set forth in Appendix A of this decision.

[16] Seven of the applicant's witnesses were members of the Independent Design Verification Program (IDVP), *see* pp. 578-79, *infra*.

[17] The applicant and the staff witnesses testified as panels. Because of the number of issues in the proceeding, the issues were treated discretely and the composition of the panels varied accordingly. A list of the witnesses, their education and their present position appears in Appendix B of this decision.

[18] Tr. D-3239. *See Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-709, 17 NRC 17 (1983).

[19] Those issues are as follows: 2(d); 3(f)(ii), (p), (s), (t); 4(a), (b), (h), (i)(2), (j)(1),(2), (k), (q), (r), (s), (u).

[20] The joint intervenors failed to file proposed findings on issue 3(f)(i) dealing with the boundary motion inputs for the applicant's soil structure interaction analysis of the containment building. *See* Joint Intervenors' Proposed Findings of Fact and Conclusions of Law (JI PF) (Dec. 23, 1983). The Governor's proposed findings now accept the applicant's results. *See* Proposed Findings of Fact and Conclusions of Law of Governor Deukmejian (Gov. PF) at 39-40 (Dec. 24, 1983).

[21] The issues remaining for decision are as follows: 1(a), (b), (c), (d), (e); 2(a), (b), (c); 3(f)(iii), (iv), (v), (o), (q), (r); 4(i)(1), (l), (t); 5; 6; 7; 8; and 9.

[22] *See Tennessee Valley Authority* (Hartsville Nuclear Plant, Units 1A, 2A, 1B, and 2B), ALAB-463, 7 NRC 341, 360 (1978), *reconsideration denied*, ALAB-467, 7 NRC 459 (1978); *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), ALAB-355, 4 NRC 397, 405 n.19 (1976).

[23] The Commission's order suspending the applicant's low power license until the successful completion of a prescribed verification program was a Commission enforcement action. Because the applicant did not challenge that action, and the Commission did not otherwise direct, no enforcement proceeding was begun. Nor did the Commission, when responding to our certified questions, indicate that its en-

*(Continued)*

independently determine whether the verification programs and their results placed before us in the reopened operating license proceeding are sufficient to verify the adequacy of the Diablo Canyon design. To do this, the applicant's efforts must be measured against the same standard as that set forth in the Commission's quality assurance criteria, 10 C.F.R. Part 50, Appendix B: whether the verification program provides "adequate confidence that a [safety-related] structure, system, or component will perform satisfactorily in service." If the applicant's verification efforts meet this standard, then there will be reasonable assurance with respect to the design of the Diablo Canyon facility that it can be operated without endangering the health and safety of the public.

B. A summary of the development and content of the Diablo Canyon verification efforts is helpful to an understanding of our resolution of the issues in Part II, *infra*.

Immediately after the discovery of the seismic design errors at Diablo Canyon, the applicant retained Robert L. Cloud and Associates, Inc. (Cloud Associates) to develop and implement an internal verification program to assess the adequacy of the plant's seismic design.[24] The initial Cloud Associates' review indicated that the design problems were more pervasive than at first thought.

Subsequent to the issuance of the Commission order[25] calling for the establishment of an extensive and structured verification effort, the applicant, on December 4, 1981, proposed a program managed by Cloud Associates that would include the services of R.F. Reedy, Inc. (Reedy Inc.) for quality assurance verification and Teledyne Engineering Services (Teledyne) for overall review of the program and its implementation. This effort was to be directed at the seismic design work performed for the applicant under pre-June 1978 service-related contracts and was labeled the Phase I program.[26] Thereafter, in response to the broader matters raised in the staff letter, the applicant also submitted a Phase II program. This program included an examination of the

forcement action should become part of the operating license proceeding. *See* note 7, *supra* and accompanying text. Therefore, we believe it is clear that the Commission did not intend to leave enforcement of its order to the reopened licensing proceeding. Thus, the elements of the verification program contained in the Commission's enforcement order, like those contained in the November 19, 1981 staff letter to the applicant (*see* note 5, *supra*), may prove useful in assessing the overall adequacy of the applicant's verification program, but in these circumstances, they do not control our determination of the sufficiency of the applicant's verification efforts.

[24] App. Exh. 90, Diablo Canyon Nuclear Power Plant-Unit 1, Final Report, Independent Design Verification Program, Vol. I (1983) (hereinafter IDVP Final Report), at 1.2-1 to -2.

[25] *See* CLI-81-30, *supra*, 14 NRC at 950.

[26] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-1. *See also* letter of December 4, 1981 from M. Furbush, PG&E, to H. Denton, NRC.

nonseismic work performed for the applicant under pre-June 1978 service-related contracts, the applicant's own internal design activities, and all the nonseismic and seismic work performed for the applicant under post-1977 service-related contracts. The Phase II program also added the Stone and Webster Engineering Corporation (Stone and Webster) to the other organizations already proposed to conduct this review.[27]

The Commission's order required that the companies conducting the verification program possess the necessary technical competence and that they be independent of the applicant.[28] On March 4, 1982, the Commission approved the Phase I program but required that Teledyne be the program manager because Cloud Associates had previously done substantial work for the applicant.[29] In accordance with this Commission action, Teledyne prepared an Independent Design Verification Program (IDVP) Phase I Program Management Plan which integrated the earlier Cloud Associates' plan and included requirements for Teledyne's acceptance of work done prior to its takeover as program manager on March 25, 1982.[30] Under Teledyne's direction, Cloud Associates would perform the review of seismic, structural and mechanical design and Reedy Inc. would review quality assurance.[31] The Phase I Plan included only the safety-related (Diablo Canyon Design Class I) buildings, equipment, piping and components that had been requalified in consideration of the Hosgri 7.5M earthquake.[32] The plan described the initial sampling and the requirements for any additional verification and sampling.[33] In a

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[27] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-2. *See also* letter of January 13, 1982 from M. Furbush, PG&E, to H. Denton, NRC.

[28] CLI-81-30, *supra*, 14 NRC at 957.

[29] App. Exh. 156, SECY-82-89, and App. Exh. 158, Memorandum from W. Dircks to S. Chilk indicating Commission approval.

[30] App. Exh. 88, IDVP Program Management Plan, Phase I, Revision I (July 6, 1982) (hereinafter IDVP Phase I Management Plan); App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

[31] App. Exh. 88, IDVP Phase I Management Plan, at 17; App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

[32] App. Exh. 88, IDVP Phase I Management Plan, Appendix D at 2; App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-8.

[33] When a criterion used in the IDVP verification process was not met, the IDVP issued an Error or Open Item (EOI) File to track the resolution of the IDVP concern. Following further investigation, the IDVP would classify the item as either a deviation (*i.e.*, a departure from standard procedure but not a mistake) or one of four categories of error (*i.e.*, A, B, C, D). The safety significance, if any, of an error was not part of the classification scheme. Rather, an error was considered class A if design criteria or operating limits of safety-related equipment were not met and physical modifications or changes in operating procedures were required. An error was considered class B if it met the definition of class A but could be resolved by more realistic calculations or retesting, instead of physical modifications. A class C error was one in which incorrect engineering or incorrect installation of safety-related equipment was found, but no design criteria or operating limits were exceeded. An error was considered class D if safety-related equipment was not affected. An EOI file remained open until the IDVP determined that the item was in conformance with licensing criteria. App. Exh. 88, IDVP Phase I Management Plan, at 25, and Appendix E; App. Exh. 89, IDVP Program Management Plan, Phase II (June 18, 1982) (hereinafter IDVP Phase II Management Plan), at 24; App. Exh. 90, IDVP Final Report, Vol. I, at 3.6-2 to -6.

letter dated April 27, 1982, the NRC staff approved the IDVP Phase I Plan.[34]

Several months later, Teledyne developed an IDVP Phase II Management Plan and submitted it to the NRC.[35] This plan encompassed nonseismic, service-related contracts performed prior to June 1978, the applicant's internal design activities, and all service-related contracts after January 1978.[36] The participants and their general responsibilities were the same as those in the Phase I Plan but Stone and Webster was added to perform the review of nonseismic safety systems and analyses.[37] On December 9, 1982, the Commission approved the Phase II Plan.[38]

Shortly after receiving approval of the Phase I program, the applicant retained Bechtel Power Corporation to work with it and act as Completion Manager for the Diablo Canyon facility. To align the verification activities with this development, the applicant developed an Overall Management Plan that, *inter alia,* adopted the IDVP Phase I Program Management Plan.[39] Under the Overall Management Plan, the joint Bechtel-PG&E team was referred to as the Diablo Canyon Project (DCP) and it was responsible for executing the Internal Technical Program (ITP). The purpose of the ITP was to (a) provide an additional design verification effort for the assurance of the overall adequacy of the design of the plant; (b) develop data and information in support of the IDVP; (c) respond to IDVP open items and findings; and (d) implement design modifications or other corrective actions arising from the verification program.[40]

Under the Phase I program, the seismic verification effort was initially based upon a sampling process.[41] The early findings of the sampling program led the applicant to review the entire scope of certain engineering activities. In order to save time and best assure final NRC approval of the verification effort, the applicant decided in the summer of 1982 to expand the seismic program to evaluate the total seismic design of safety-related structures, systems, and components.[42] This broad review

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[34] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

[35] *Id.* at 1.3-6.

[36] App. Exh. 89, IDVP Phase II Management Plan, at 1.

[37] *Id.* at 8.

[38] App. Exh. 157, SECY-82-414; App. Exh. 159, Memorandum from W. Dircks to S. Chilk indicating Commission approval.

[39] App. Exh. 90, IDVP Final Report, Vol. I, at 1.4-1.

[40] *Id.* at 1.4-1 to -2.

[41] *Id.* at 1.4-2.

[42] *Id.*; App. Exh. 91, ITP Design Verification Program Phase I Final Report (Oct. 19, 1983) (hereinafter ITP Phase I Final Report), at 1.5.2-1 to -2.

*(Continued)*

enveloped the findings of the previous IDVP and ITP seismic reviews and made it unnecessary to review older analyses and calculations that were to be redone by the ITP. In view of the enlarged ITP seismic review,[43] the IDVP program was changed from one of sampling original designs to one of verifying the ITP's seismic work. The IDVP examined the scope, criteria and methodology of the ITP work for consistency with the license application and then verified samples of that work.[44] In addition, the staff reviewed the seismic verification efforts of the ITP and the IDVP on a continuing basis.

The IDVP also selected samples of the original engineering design work for the Phase II nonseismic verification.[45] The samples were reviewed and analyzed by the IDVP against verification criteria from the program management plan. If the criteria were not satisfied, the initial samples were reanalyzed or additional samples were identified for verification. When the IDVP identified a potentially generic concern, the ITP was required to perform a review for that concern for all applicant-designed, safety-related systems.[46] The IDVP then evaluated these ITP reviews and documented their findings in Interim Technical Reports (ITRs) for the staff to review. In addition to the nonseismic reviews performed by the ITP at the direction of the IDVP, the ITP independently conducted a functional design review that covered a portion of each of

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This phase of the work by the Bechtel-PG&E team is referred to as the Corrective Action Program (CAP). Thus, there are several labels which may be applied to work carried out by that group (*i.e.*, DCP, ITP, CAP). Because our previous references to the work done by the Bechtel-PG&E team in the proceeding have been to the ITP, we shall continue to use ITP as a catchall phrase to denote work done both by the applicant subsequent to November 1981 as well as by the Bechtel-PG&E team.

[43] The complete ITP seismic review program is described in the ITP Phase I Final Report, App. Exh. 91.

[44] App. Exh. 90, IDVP Final Report, Vol. I, at 3.5-7.

The seismic design review resulted in thousands of minor modifications to steel frame structures and supports for piping, raceways, instrumentation, instrument tubing and equipment. App. Exh. 91, ITP Phase I Final Report, at 1.8.6-2 and Appendix 1E. A large number of modifications must be expected when seismic response spectra are changed, because many similar structural components are included in each individual seismic analysis and each component may be affected by a change in the seismic response spectra. For example, several pipe support modifications could result from a single change in one pipe analysis and that piping design may be repeated hundreds of times. *See id.* at 2.2.1-22 to -36 (Table 2.2.1-3), 2.2.1-37 to -51 (Table 2.2.1-4), 2.2.2-17 to -24 (Table 2.2.2-1). *See also* Moore Tr. D-412.

[45] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-8 to -9. The entire IDVP verification program (*i.e.*, seismic and nonseismic) is documented in sixty-three interim technical reports (App. Exhs. 93 to 155) and a four-volume final report that contains the IDVP's conclusions (IDVP Final Report) (App. Exh. 90).

[46] Only a few of the findings from the nonseismic design review resulted in modifications to plant systems and the alterations were minor. App. Exh. 92, ITP Phase II Final Report Design Verification Program (hereinafter ITP Phase II Final Report), at 3-2 to -3. For example, minor modifications were performed involving the following: (1) rerouting of certain electrical circuits to assure circuit independence; (2) electrical changes to the control room ventilation and pressurization system to allow the single failure criterion to be met for Unit 1 without the availability of Unit 2 power supplies; (3) auxiliary feedwater system alterations to prevent inadvertent overpressurization of certain components; (4) strengthening of doors; and (5) installing flow limiters and dampers. *Id.* at 3-3 to -31.

the applicant-designed, safety-related nonseismic systems.[47] Unlike the seismic review, the entire design of applicant-designed, safety-related systems was not reviewed.

## II. FINDINGS ON CONTESTED ISSUES

As previously noted, the real issue in this reopened proceeding is whether, in view of the conceded weakness of the Diablo Canyon design quality assurance program, the applicant's verification efforts demonstrate that the safety-related structures, systems and components of the plant are properly designed (*i.e.*, conform to the various licensing criteria for the facility). Although the applicant presented evidence to establish that it verified the design of both Diablo Canyon Units 1 and 2, we make no findings with respect to Unit 2. The two units are nearly identical from a design standpoint, but the applicant's verification efforts for Unit 2 differ from those for Unit 1. Significantly, the IDVP had no direct involvement in the Unit 2 verification program. Rather, the applicant has established an internal review organization for Unit 2 to evaluate deficiencies identified for Unit 1 and, if appropriate, to correct these deficiencies as they appear in Unit 2. The Unit 2 verification is still ongoing and has not been finally reviewed by the staff. Nor has the staff issued a safety evaluation report supplement on the Unit 2 verification. In the circumstances, we believe it is most appropriate to sever the question of the Unit 2 design verification from the proceeding and decide at this time only the issues related to Unit 1.

A. In issues 1 and 2, the Governor and joint intervenors challenge the scope of the applicant's verification program and, in effect, dispute the ability of the applicant's verification efforts to provide the same assurance of proper design as a satisfactory quality assurance program.

Specifically, in issues 1(a) and (b), the joint intervenors and the Governor assert that the scope of the IDVP review was too narrow because it did not verify samples from each design activity and from each design group performing a particular design activity. Issues 2(a) and (b) raise the same questions but with regard to the ITP verification efforts. The joint intervenors and the Governor also contend in issues 1(c) and 2(c) that the IDVP and ITP verification efforts were flawed because they did not have statistically valid samples from which to draw conclusions. Because there was a marked difference in the manner in

[47] The modifications required by the ITP's functional design review are described in the ITP Phase II Final Report. App. Exh. 92 at 2-5 and Appendix B.

which the seismic and nonseismic verifications were conducted, we first treat the seismic verification by the IDVP and the ITP, then in section 2, we deal with the nonseismic verification.[48]

1. The ITP essentially redid all of the seismic design for safety-related structures, systems and components, while the IDVP oversaw and verified selected samples of the work.[49] The ITP reanalyzed the design of portions of the containment, the auxiliary building, the fuel handling building, the turbine building and the intake structure. All large bore piping and pipe supports were reanalyzed, and small bore piping and pipe supports were reviewed either by sampling or on a generic basis. The ITP reviewed or reanalyzed the safety-related mechanical, electrical, and instrumentation and control equipment to assure that these components were seismically qualified. In addition, the ITP examined the design of all safety-related electrical raceways and heating, ventilation, and air conditioning (HVAC) ducts and supports. Finally, the ITP sampled the safety-related instrument tubing and supports to ensure their seismic qualification.[50] Thus, with respect to the seismic design, the work of the ITP became the design of record.[51]

The ITP's seismic design work was done under a quality assurance program that met the provisions of 10 C.F.R. Part 50, Appendix B.[52] In addition, this work was independently verified by the IDVP. In each of the areas of seismic design addressed by the ITP, the IDVP verified the work by reviewing selected samples. The exact approach taken by the IDVP varied depending upon the nature of ITP work.[53] For all reviews, however, the IDVP first compared the scope of the ITP work with the applicable license criteria, and then ascertained that the analytical methods used by the ITP were valid, verifying such items as modeling techniques, model constraints, assumptions and the levels of model sophistication. In each seismic design area, the IDVP selected a sample

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[48] *See* note 67, *infra*, for discussion of issue 1(d). The remaining parts of 1 and 2, issues 1(e) and 2(d), pertain to Unit 2. *See* p. 582, *supra*.

[49] Anderson *et al.* [This panel consisted of R. Anderson, G. Cranston, G. Moore, L. Shipley and W. White.] Tr. fol. D-224 at 5-6, 9-10; Cooper *et al.* [This panel consisted of W. Cooper, R. Cloud, J. Krechting and R. Reedy.] Tr. fol. D-1459 at 1/2-12 to -20; App. Exh. 100, ITR 8, at 1-2.

[50] Anderson *et al.* Tr. fol. D-224 at 6; Seed *et al.* [This panel consisted of R. Anderson, H. Seed, L. Shipley and W. White.] Tr. fol. D-652 at 7-8; App. Exh. 91, ITP Phase I Final Report, at 1.5.1-3 to -4.

[51] Cooper *et al.* Tr. fol. D-1459 at 1/2-13.

[52] The adequacy of the quality assurance program covering the ITP's work is discussed subsequently. *See* pp. 613-17, *infra*.

[53] App. Exh. 100, ITR 8, at 1-2.

The ITP seismic verification work was divided into three categories according to the methods used: complete reanalysis (*e.g.*, Fuel Handling Building); review of existing analyses followed by reanalysis of deficient items (*e.g.*, large bore piping); and reviews of samples to demonstrate conservative design (*e.g.*, small bore piping). *Id.*

of calculation packages for detailed review. The review was designed to investigate the specific concerns that the IDVP developed during earlier IDVP reviews, and to ensure the complete evaluation of the process utilized by the ITP. The calculation packages were verified by design review or by performing independent analyses, or a combination of these techniques. IDVP samples consisted of in-progress and completed ITP work. In certain instances, questions arose which caused additional samples to be evaluated by the IDVP. For each area of ITP work reviewed, the IDVP issued an ITR documenting the results of its review.[54] Thus, the final seismic design derived from the ITP's efforts and the IDVP review of those efforts subjected the design of Diablo Canyon to a measurably greater level of scrutiny than could have been provided by a quality assurance program complying with Appendix B.[55]

The Governor asserts, however, that the seismic verification was insufficient because the ITP's redesign efforts did not encompass all elements of the seismic design. Specifically, he claims that small bore (less than 2-inch diameter) piping was requalified not by 100 percent review, but through a program of generic reviews and sampling, and that instrument tubing supports were also requalified by sample calculations. He charges that the ITP reviewed equipment only if the response spectrum governing its seismic design had changed and, even then, the ITP only

[54] Cooper *et al.* Tr. fol. D-1459 at 1/2-13 to -20.
The following ITRs document the IDVP review of the ITP seismic verification work:

| Applicant Exhibit # | ITR Number | Subject |
|---|---|---|
| 142 | 50 | Containment Annulus |
| 143 | 51 | Containment Annulus |
| 144 | 54 | Containment Building |
| 145 | 55 | Auxiliary Building |
| 146 | 56 | Turbine Building |
| 147 | 57 | Fuel Handling Building |
| 148 | 58 | Intake Structure |
| 149 | 59 | Large Bore Piping |
| 150 | 60 | Pipe Supports |
| 151 | 61 | Small Bore Piping |
| 152 | 63 | HVAC Ducts, Electrical Raceways, Instrument Tubing and Associated Supports |
| 153 | 65 | Rupture Restraints |
| 154 | 67 | Equipment |
| 155 | 68 | Soils |

[55] Moreover, the nature and breadth of the seismic design review (*i.e.*, essentially 100 percent) eliminates any reasonable argument that the review was flawed because statistically valid sampling techniques were not used.

evaluated safety-related equipment designed by the applicant, not others.[56]

None of the Governor's challenges detracts from the adequacy of the applicant's seismic verification programs. Small bore piping at Diablo Canyon was designed by computer-based analysis or by the use of span criteria.[57] The ITP verification was carried out by "generic" reviews,[58] and by sampling.[59] The ITP reported the results of some 80 piping analyses, involving approximately 1,550 piping spans,[60] carried out under the generic and sampling programs.[61] Noting the ITP's use of computer-based dynamic analysis and its limited use of the less conservative span rules. the IDVP concluded that the ITP methods and coverage were acceptable and the ITP analysis ensured that small bore piping was properly designed.[62] We agree. There is no need to test every repetitive pipe configuration. The ITP's broad coverage in its generic and sampling reviews was sufficient to assure adequacy of the piping design.

The seismic design of instrument tubing supports, like that of small bore piping, need not be verified by 100 percent reanalysis. There are only a few basic seismic designs of instrument tubing supports, although there are many applications of each design. The ITP selected for review a sample of eighty-eight supports that represented worst case and enveloping situations. Of these supports, the analyses indicated that two were inadequate as a result of their specific cantilevered configuration. All tubing supports in the plant were then examined for this configura-

[56] Gov. PF at 29-31.

[57] Span criteria are analytically determined rules which govern the spacing between seismic supports in a run of piping (*i.e.*, the length of the span of pipe between supports). App. Exh. 122, ITR 30, at 6 and A-6; "Seismic Evaluation for Postulated 7.5M Hosgri Earthquake" (Hosgri Report), Amendment No. 50 to operating license application (June 3, 1977), Vol. II, at 8-3 to -4 and Figure 8-1. The span rules were revised by the ITP to include the effect of insulation and spectra revisions, and to provide more user guidance. App. Exh. 91, ITP Phase I Final Report, at 2.2.2-6.

[58] The "generic" program encompassed small bore piping and piping analyses issues identified by the IDVP and ITP reviews as having a potential for causing modifications. *Id.* at 2.2.2-1. Specifically, the program included the following piping configurations: those previously analyzed by dynamic analysis; those in which safety-related valves are supported by pipes; those subject to thermal or seismic movement of anchors; those at boundaries between code requirements; and those pipes subject to thermal stresses previously qualified by span rules. *Id.* at 2.2.2-4 to -6. The generic review program was carried out primarily by dynamic analyses. *Id.* at 2.2.2-8 to -9.

[59] *Id.* at 2.2.2-1.

Under the sampling program a number of piping configurations designed using span criteria were selected to undergo dynamic analysis as well. The selection of samples was made to address a number of specific design configurations and issues not included in the generic review, and to demonstrate the qualification of piping that was designed using span criteria. *Id.* at 2.2.2-2 to -3, -6 to -8, -10 to -11.

[60] The computer analysis of a piping configuration generally includes many (typically ten to fifty) supports. Thus, a single piping analysis checks the design adequacy of many pipe spans. *See id.* at 2:2.2-17 to -24 (Table 2.2.2-1).

[61] *Id.* at 2.2.2-8 to -11, 2.2.2-17 to -24 (Table 2.2.2-1).

[62] App. Exh. 151, ITR 61, at 54, 60.

tion and no other deficient cantilevered supports were found.[63] The IDVP review of the ITP effort confirmed that the analyzed tube support configurations included worst case situations, and concluded that the tube supports throughout the plant were adequate.[64] Once again, we agree with the IDVP's conclusion. Because of the repetitive nature of the instrument tubing support design, there is no need to test every support. The breadth of the ITP review, which included worst case analysis, was sufficient to ensure proper instrument tube support design.

Finally, with respect to the Governor's last challenge to the sufficiency of the seismic review, we find that the seismic qualification of safety-related equipment was not deficient. The ITP determined new seismic response spectra for all structures except the containment shell. In reviewing the equipment for qualification to the new spectra, the ITP reviewed all safety-related equipment, even that in the containment, so that no equipment was overlooked.[65] Nor was the seismic review flawed because the IDVP did not review the qualification of safety-related equipment designed by Westinghouse. We deal with the question of the sufficiency of Westinghouse-designed equipment subsequently.[66] Suffice it to say at this point that Westinghouse itself performed an adequate seismic design review of the equipment and systems it supplied.

We conclude, therefore, that the seismic redesign process carried out by the ITP and reviewed by the IDVP provides adequate confidence that the seismic design of the structures, systems and components at Diablo Canyon Unit 1 is proper and meets licensing criteria.[67]

2. Unlike the seismic verification under which essentially all of the Diablo Canyon seismic design was reviewed, the applicant's nonseismic design review efforts were less ambitious. Although both the IDVP and

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[63] App. Exh. 91, ITP Phase I Final Report, at 2.6-1 to -4.

[64] App. Exh. 152, ITR 63, at 47, 54; App. Exh. 90, IDVP Final Report, Vol. II, at 4.6.8.2-1 to -4.

[65] Seed *et al.* Tr. fol. D-652 at 7-8, 61-64; App. Exh. 91, ITP Phase I Final Report, at 2.3-1.

Mechanical equipment was checked by one of three methods: flexible items (having natural frequency less than 33 Hertz) were subjected to dynamic analysis; rigid items were qualified to equivalent static loads or by dynamic analysis; and some equipment was qualified by testing on a shake table. App. Exh. 91, ITP Phase I Final Report, at 2.3.1-5; App. Exh. 154, ITR 67, at 10, 17, 29. Similar methods were used to verify the electrical and HVAC equipment items. App. Exh. 91, ITP Phase I Final Report, at 2.3.2-2, 2.3.3-2 to -3.

[66] *See* pp. 608-10, *infra.*

[67] In issue 1(d), the Governor also challenges the sufficiency of the IDVP seismic review program claiming that, instead of independently verifying analyses for Diablo Canyon, it merely checked data inputs to the applicant's design models. The record is replete with instances in which the IDVP carried out its own calculations, both in the seismic and nonseismic areas of the verification. The evidence also demonstrates that design reviews carried out in lieu of independent analyses were far more extensive than a mere checking of input data. Cooper *et al.* Tr. fol. D-1459 at 1/2-16, -19 to -20, -28 to -29, -34 to -35; Cloud Tr. D-1939-41, D-1944-45. Moreover, we find that the IDVP's approach of verifying samples by a combination of reanalysis and design review is sufficient to provide adequate verification of design. The value of independent recalculations is not disputed, but there is no indication that this approach is essential to provide assurance of design efficacy. Cloud Tr. D-1937-38; Roesset Tr. D-2247-48.

the ITP verified portions of the nonseismic design of the facility, their combined efforts did not encompass the entire nonseismic safety-related design. For example, neither the IDVP nor the ITP verified samples from each design activity and each design group performing that activity, as alleged to be necessary by the Governor and the joint intervenors in issues 1(a) and (b), and 2(a) and (b). Nor did the IDVP and ITP select the portions of the nonseismic design work they reviewed on a statistically valid basis (*i.e.*, they did not randomly sample the universe of engineering design decisions), as urged by the Governor and the joint intervenors in issues 1(c) and 2(c). Because the nonseismic review was not all encompassing and not based on statistically valid sampling techniques, the Governor and the joint intervenors claim that the applicant's verification program is so seriously flawed that it cannot properly be used as a basis for reaching conclusions about the unreviewed portions of the nonseismic design. The applicant and the staff, on the other hand, assert that the scope and nature of the applicant's nonseismic design review are more than sufficient to support the conclusion that the Diablo Canyon design meets applicable licensing criteria.

Specifically, the Governor and the joint intervenors assert that because the design samples selected by the verification program were chosen deliberately on the basis of certain engineering judgments, and not randomly, the sample selection process was biased. Thus, the argument continues, no statistically valid conclusion regarding probabilities of errors or error rates can be drawn for the unreviewed portions of the nonseismic design; and, in order to verify satisfactorily the nonseismic design, the applicant must go back and either randomly sample the universe of nonseismic design decisions or review 100 percent of it.

This argument essentially overlooks the standard by which the applicant's program is to be judged. We must determine whether the nonseismic verification program provides "adequate confidence" that the nonseismic design of safety-related structures, systems and components is proper so that such structures, systems and components will perform satisfactorily in service.[68] This qualitative standard is not numerically

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[68] Pointing to the Commission's regulations, 10 C.F.R. 50.57(a)(1), the Governor and the joint intervenors repeatedly assert in their proposed findings that the applicant's verification program, in order to be sufficient, must demonstrate that the design of Diablo Canyon meets its license application requirements or licensing criteria. The application requirements and licensing criteria for Diablo Canyon, like any nuclear power plant, are spelled out in the various documents comprising the operating license application including, most prominently, the applicant's *Final Safety Analysis Report* (FSAR). The FSAR is a multivolume description of the entire facility containing literally thousands of so-called "licensing criteria" ranging from safety significant ones to insignificant and extremely minor specifications or descriptions of details that have no safety implications. *See* 10 C.F.R. 50.34(b). In their proposed findings of fact, the Governor and the joint intervenors do not distinguish between safety significant and nonsafe-

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quantifiable into expressions of probability of errors or error rates, as the Governor and the joint intervenors would have it. Even if a statistically valid error rate were available to forecast the errors in the unreviewed portions of the nonseismic design,[69] in all but certain obvious situations, such a rate would be of little utility in judging the adequacy of the verification of the nonseismic design of Diablo Canyon. In part, this is because no acceptable rate of design errors for nuclear power plants has ever been determined.[70] Thus, the ultimate determination regarding the adequacy of the plant's design remains a qualitative judgment and we must turn to the verification work that was performed to ascertain whether its scope and quality are sufficient to provide the requisite assurance of design adequacy.[71]

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ty significant licensing criteria. For example, the Governor and joint intervenors argue, relying on the staff's and the applicant's witnesses, that the nonseismic design does not meet licensing criteria because it is a virtual certainty that there remain undetected design errors in the unreviewed portions of the design. JI PF at 14-16; Gov. PF at 5-8, 9-11. But the witnesses relied upon by the Governor and joint intervenors all testified that not only was it likely there remained some design errors, but that it was extremely unlikely any of the errors were safety significant. Cloud Tr. D-1543, D-1545; Schierling Tr. D-2662-63, D-2665; Knight Tr. D-2706. In effect, the Governor and joint intervenors champion form over substance. We reject their position. The central issue with respect to the proper design of Diablo Canyon, or any other facility, is the conformance of the design to the significant and substantive safety requirements and licensing criteria. To conclude otherwise would ignore reality and substitute "perfection" for the regulatory standards of "adequate confidence" and "reasonable assurance." *See* p. 578, *supra*.

[69] Dr. Stanley Kaplan, an engineer and applied mathematician, appeared as an expert witness for the applicant. Dr. Kaplan used the results of the nonseismic design verification work of the IDVP and applied Bayesian techniques to predict an error rate for the original design of the plant (*i.e.*, errors per design element). Also, using the judgment of the engineers associated with the verification effort that the errors identified by the verification were minor and of little safety significance, Dr. Kaplan applied his methodology to determine the likelihood of safety significant design errors remaining in the unsampled portions of the nonseismic design. Kaplan and Anderson Tr. fol. D-1161 at 56-63. Dr. Kaplan, however, cautioned (*id.* at 45) that his "numerical results are to be interpreted with a large grain of salt. . . ." *See id.* at 17-22.

The Governor's and the joint intervenors' expert witnesses, Drs. Apostolakis and Samaniego — both statisticians — reject out of hand Dr. Kaplan's projected error rate because it was calculated using nonrandomly selected samples. Samaniego Tr. D-2394-95; Apostolakis Tr. D-2343. Because we find little utility in the determination of error rates (or their accuracy) for the qualitative judgment we must make on the adequacy of the verification program for the nonseismic design, we need not decide the validity of Dr. Kaplan's calculations.

[70] Kaplan and Anderson Tr. fol. D-1161 at 67-70; Apostolakis Tr. D-2354, D-2369; Knight *et al.* [This panel consisted of J. Knight, H. Schierling and J. Wermiel.] Tr. fol. D-2649 (Contention 2) at 7-8.

According to the Governor and joint intervenors, the evidence indicates that, in spite of the verification program, there remain errors in the unreviewed portions of the nonseismic design that represent failures to meet licensing criteria. This fact, they claim, renders the verification program inadequate. Gov. PF at 9-11, 38-39; JI PF at 14. Thus, the Governor and the joint intervenors apparently would accept only a zero error rate. *See* note 68, *supra*.

[71] While it is unnecessary to consider the statistical question in more depth, we note our skepticism that a statistically valid design verification program, as thorough as the applicant's verification efforts, could have been developed and implemented. No such program has ever been developed for a nuclear power plant. Apostolakis Tr. fol. D-2313 at 12; Samaniego Tr. D-2408-10, D-2451. Although theoretically possible, implementation presents formidable obstacles such as identifying and stratifying the many thousands of design decisions that went into the facility so they may be randomly sampled. Kaplan and Anderson Tr. fol. D-1161 at 5-6; Apostolakis Tr. D-2335-44. It must be borne in mind that the subject

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The IDVP and the ITP each took a different approach to verify the nonseismic design work. The IDVP chose for review three specific safety-related systems that included work from all the applicant's internal design groups and the service-related contractor who performed the most significant nonseismic design work.[72] It also selected two areas of safety-related analysis applicable to many other systems.[73] The majority of the IDVP's nonseismic verification involved the performance of independent calculations or analyses using models generally different than

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under investigation is the design adequacy of a complex facility consisting of a multitude of engineered systems, each with its own function and each with some potential for interacting in various ways with the other plant systems. Each "design element" or design decision for a particular system involves input from previous determinations for that system and for interacting systems. We are not persuaded that random sampling of such elements is necessarily the most effective means for addressing design adequacy. Rather, a coherent sampling scheme devised in view of a system's characteristics, its function, and its interaction with other systems appears to us to be a more acceptable method for ascertaining the adequacy of the design of a nuclear power plant. Cooper *et al.* Tr. fol. D-1459 at 1/2-14, -24 to -25; Anderson *et al.* Tr. fol. D-224 at 25-27.

[72] Cooper *et al.* Tr. fol. D-1459 at 1/2-24.

[73] *Specifically, the IDVP selected the auxiliary feedwater (AFW) system, the control room ventilation* and pressurization (CRVP) system, and the safety-related portion of the 4160 volt (V) electric distribution system for review. As stated by the IDVP:

> The AFW system was selected because its design represents an interrelationship of several design criteria and interfaces. Specifically, it involves interface with NSSS [Nuclear Steam Supply System] vendor criteria, with containment design criteria, interface of PGandE internal design organizations, and the methodology of determining a water system's mechanical, electrical, and control component design criteria. In addition, AFW systems often appear in the dominant accident sequences in various probabilistic risk assessment programs.
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> The CRVP system was selected because it too represents an interrelationship of several design criteria and interfaces. Specifically, it involves interface with a service-related contractor, interface of PGandE internal design organizations, and interface with the control room habitability criteria. It also represents a contrast of design methods since it is an air system rather than a water system.
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> The safety-related portion of the 4160 V electrical distribution system was selected because it is the basic power supply for safety-related electrical equipment. It also represents an interrelationship of several design criteria and involves the interfaces among several PGandE internal design organizations.
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> The three sample systems were designed by different engineering groups within PGandE, thus providing for evaluation of a broad spectrum of the PGandE engineering organization.
>
> In addition, the IDVP selected two areas of safety-related analyses for review: the integrated dose analyses; and the temperature, pressure and humidity analyses as they affect environmental qualification of equipment. These analyses were selected since this work was done almost exclusively by three service-related contractors and utilized by PGandE. The service-related contractors were different and their work involved a flow of design information through PGandE engineering groups.
>
> For the three selected sample systems, a complete vertical verification of the system design was performed. The applicable licensing criteria were identified, and a system design chain was developed. The system's design was then reviewed to determine if the licensing criteria were satisfied. The review included the aspects of mechanical, electrical and instrumentation and control design.
>
> In addition, the IDVP performed the following verifications of the sample systems. The IDVP verified the fire protection provided for the sample systems, including the separation, fire barriers, suppression and detection systems provided in areas containing sample system components. The IDVP verified that the AFW and CRVP systems were adequately protected from the effects of a high energy line break (HELB), high energy line crack (HELC), and moderate energy line break (MELB).

Cooper *et al.* Tr. fol. D-1459 at 1/2-21 to -23.

those employed in the original design.[74] When the IDVP identified a concern (*e.g.*, a design error) that it believed was generic (*i.e.*, having the potential for being repeated in other systems), this concern was then addressed by the ITP for *all* applicant-designed systems.[75] In turn, the ITP's verification work was sampled by the IDVP and the results reported in an ITR.[76]

The IDVP verification samples for nonseismic design encompassed the work of the primary applicant engineering design groups (civil, mechanical, electrical, instrumentation and control, and heating and ventilation). It also covered the work of the three major service contractors in the nonseismic area: Quadrex (formerly Nuclear Services Corporation) — jet impingement and pipe whip analysis; EDS Nuclear Inc. — heating and ventilation system design and other activities; and Radiation Research Associates — radiation dose calculations.[77]

In addition to reviews resulting from the identification of concerns by the IDVP in the nonseismic design area, the ITP independently per-

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[74] *Id.* at 1/2-35.

[75] The ITP addressed the following concerns identified by the IDVP:

> all areas of analyses of pressure, temperature and humidity due to HELB; selection of system design pressure and temperature; selection of differential pressure across power operated valves; redundancy of power supplies for shared systems; separation and single failure criteria for mutually redundant circuits; and jet impingement effects of HELB inside containment.

*Id.* at 1/2-24.

[76] App. Exhs. 137 to 141.

[77] App. Exh. 90, IDVP Final Report, Vols. I and II, at 4.2.2-6 to -8, 4.7.1-1 to 4.7.7-5.

The IDVP did not review the work of all service contractors. For example, it did not review the work of Westinghouse, Western Canada Hydraulic Laboratories (Western Canada), Stafco Associates (Stafco), and the IDVP contractors, Cloud Associates and Teledyne. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.4-3; Reedy Tr. D-1486. We have reviewed each of these excluded contractors and conclude that because of the circumstances in each case, the exclusions were reasonable and do not render the verification efforts inadequate as claimed by the Governor and joint intervenors. As we discuss *infra*, pp. 608-10, Westinghouse had its own properly functioning quality assurance program that assured the adequacy of both the services it performed and the equipment it designed for the applicant. Although Western Canada did not have a proper quality assurance program (App. Exh. 157, SECY-82-414, Encl. 5 at 5), Western Canada's work in vortex analysis — the same work it performed for the applicant — had been audited in a generic review and found sufficient by the NRC staff. App. Exh. 101, ITR 9, at A52; Cooper Tr. D-1478-79, D-1481-82, D-1750-51. Stafco assisted in the preparation of a list of safety-related structures, systems and components and in updating the FSAR. Because Stafco did not perform design work, it was properly excluded from the design verification program. Reedy Tr. D-1486, D-1488. Finally, with respect to the IDVP participants, Teledyne had a satisfactory quality assurance program that attests to the sufficiency of its design work. App. Exh. 157, SECY-82-414, Encl. 5 at 4. In any event, the ITP reviewed the seismic work previously performed for the applicant by Teledyne. App. Exh. 91, ITP Phase I Final Report, at 2.2.3-5. Cloud Associates, on the other hand, did not have a quality assurance program. App. Exh. 157, SECY-82-414, Encl. 5 at 5. Of the three projects Cloud Associates performed for Diablo Canyon (a review of pipe whip restraints, a systems interaction program, and a research program on seismic capability of nonseismic design components), only pipe whip restraint comprised design work that would normally have been subject to review by the IDVP, but was excluded because Cloud Associates was a member of the IDVP. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.4-3; App. Exh. 156, SECY-82-89, Encl. 1 at II, p. 4. That Cloud Associates was not reviewed is not now important. As part of the complete seismic review, the ITP re-evaluated all rupture restraints inside and outside containment to assure they were properly designed and installed. Staff Exh. 37, SSER 19, at C.4-2 to -3. Thus, the exclusion of these five service-related contractors does not render the applicant's verification efforts insufficient.

formed a functional design review of the applicant-designed, safety-related mechanical, electrical and ventilation systems. The instrumentation and controls for mechanical systems, and all of the safety-related mechanical, electrical and ventilation systems were reviewed to assure adequate protection against a series of postulated hazards.[78] This nonseismic evaluation was performed in accordance with an NRC-approved quality assurance program meeting the criteria of Appendix B.[79]

While the scope of the nonseismic review of the Diablo Canyon safety-related systems was not as complete as the seismic review, an appreciable portion of the nonseismic design was verified.[80] There were errors identified that required reanalysis and, in some instances, physical modifica-

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[78] Anderson *et al.* Tr. fol. D-224 at 17-19; Kaplan and Anderson Tr. fol. D-1161 at 64-66.

[79] Anderson *et al.* Tr. fol. D-224 at 7; Staff Exh. 36, SSER 18, at C.2-3 to -4; Dick *et al.* [This panel consisted of C. Dick, M. Jacobson, S. Skidmore and T. de Uriarte.] Tr. fol. D-847 at 9.

The Governor seeks to have us discount (as an applicant trial ploy not worthy of belief) that portion of the ITP review work that was not performed for, and reviewed by, the IDVP. Gov. PF at 31-34. The Governor argues that prior to the hearing none of this ITP work was represented by the applicant as an additional verification effort and, in any event, the ITP review was neither documented to the same extent as the IDVP reviews, nor done to the same depth as the IDVP work. But the existence of the separate ITP review is evident from the applicant's semi-monthly reports as early as February 1982, and contrary to the Governor's assertions, the applicant's June 1983 Phase II Final Report clearly identifies this ITP review effort. App. Exh. 92, ITP Phase II Final Report, at iv. Moreover, the fact that the verification work is not documented in the same fashion as the work carried out in conjunction with the IDVP is a reflection of the fact that the latter program had reporting requirements imposed upon it by the Commission and the NRC staff. The ITP review work is recorded in the applicant's files and open items (*i.e.*, errors) found during the course of this review are discussed in the Phase II final report. Anderson Tr. D-1426. App. Exh. 92, ITP Phase II Final Report, at 3-22 to -31. Thus, absent a valid showing that the work is flawed, or an objection to its admission as evidence, neither of which was made, the ITP's functional design review stands as significant evidence of the adequacy of the Diablo Canyon nonseismic design.

[80] The applicant's witness Anderson estimated that the total nonseismic design review (IDVP plus ITP) encompassed about seventy-five percent of the engineering work at Diablo Canyon. Tr. D-1419-20, D-1425. He readily admitted, however, that his figure could be characterized as rather "soft." Tr. D-1441, D-1426-27, D-1429-33. The Governor and the joint intervenors take issue with Mr. Anderson's estimate. Gov. PF at 31-35; JI PF at 2-3. They object because the figure is an estimate, not a precise number, and because the ITP functional design review component of Mr. Anderson's estimate was neither mandated nor reviewed by the IDVP. They assert that if the latter component of the estimate is discarded, the IDVP only reviewed twenty-three percent of the design elements of the nonseismic work. This argument overlooks the review by the ITP performed at the direction of the IDVP (*see* p. 590, *supra*), and as previously indicated, there is no reasoned basis for discarding the ITP functional design review. *See* note 79, *supra*, and accompanying text. Further, Mr. Anderson's seventy-five percent estimate dealt with total *engineering work* covered by both the IDVP and ITP reviews, not *design elements*. Anderson Tr. D-1419-20, D-1427, D-1436, D-1438-39. The two are vastly different. There are numerous design elements of varying significance in the engineering work involved in a project of this magnitude.

Moreover, it is not the exact quantification of work reviewed that is critical. The important consideration is that the scope and implementation of the nonseismic verification program was sufficient to test thoroughly the design process in order to discover any defects in that process. Here, the applicant's verification program encompassed three systems in their entirety (covering the spectrum of applicant's in-house design groups and the interrelationships of all significant design criteria and interfaces) and parts of all the remaining nonseismic systems. This slice of the nonseismic design process was sufficient to uncover any significant inadequacies in the design process.

tions were necessary in order to comply strictly with licensing criteria,[81] but all the errors were judged to be of minor safety significance.[82]

The IDVP's sampling method involved a complete (vertical) review of three dissimilar systems, followed by an analysis by the ITP (across all systems horizontally) to search for generic problems suggested by errors found in the vertical review. By using this approach, the IDVP was able to obtain a broad and comprehensive understanding of the nonseismic design of Diablo Canyon Unit 1.[83] As a result of this review, the IDVP concluded that while there may exist errors in the unsampled portion of the design, the likelihood of the existence of a safety significant error was small.[84]

To a certain degree, this judgment was tested by the ITP's own review and found valid. The ITP review sampled an additional portion of the safety-related, nonseismic design of the plant and, once again, it found only a small number of errors, none of which was safety significant. This led the ITP to conclude, with a high degree of assurance, that the nonseismic design of the plant was adequate.[85]

We are not without some puzzlement as to why, having reviewed so much of the plant, the applicant did not carry through to a total review of the nonseismic design. In light of the history of the Diablo Canyon facility and the considerable time and resources already expended by the applicant on the verification programs, such an additional undertaking might well have proven a provident step in order to dispel the inevitable speculation as to the adequacy of the unreviewed portions of the nonseismic design. Nevertheless, on the basis of all the evidence, we find that the verification efforts of the IDVP and ITP were sufficient to provide adequate confidence that the nonseismic design criteria have been

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[81] *See* notes 46 and 47, *supra.*

[82] The Governor and joint intervenors object to this characterization of the nonseismic design errors that were discovered because no formal analysis was performed to assess their seriousness or their potential for reducing the plant's margin of safety. Gov. PF at 16-17; JI PF at 15. They contend that the latter determination requires the performance of a probabilistic risk assessment. *See* Apostolakis Tr. fol. D-2313 at 10-11. But neither the Governor nor the joint intervenors presented any direct evidence to dispute the expert opinions of the staff and applicant witnesses that none of the errors found by the verification program was safety significant. Anderson *et al.* Tr. fol. D-224 at 12-14; Anderson Tr. D-345-46, D-1420; Knight Tr. D-2696-97, D-2819; Cooper *et al.* Tr. fol. D-1459 at 1/2-32. We find that the expertise of the applicant's and the staff's witnesses in the design, construction and operation of nuclear power plants qualifies them to evaluate the safety significance of such nonseismic errors, at least to the point of determining whether the errors warrant a quantitative evaluation. While we agree that as a general proposition only a formal analysis can provide a quantitative assessment of an error's significance, our review of the nonseismic errors identified by the IDVP and ITP leads us to concur in the judgment of the applicant's and the staff's experts that the errors are of minor safety significance.

[83] Cooper *et al.* Tr. fol. D-1459 at 1/2-25.

[84] *Id.* at 1/2-32.

[85] Anderson *et al.* Tr. fol. D-224 at 19-21; App. Exh. 92, ITP Phase II Final Report, at 5-1.

met for Unit 1. Through sampling which covered all the engineering disciplines and types of analyses, and which encompassed a major portion of the plant's design, the IDVP and ITP concluded that the original design process was efficacious. The NRC staff concurred with this conclusion. The errors found were few, of minor significance, and did not indicate a pervasive weakness in any design area. We concur in the judgments of the ITP, IDVP and staff that the level of assurance provided by the applicant's verification efforts is comparable to that which would be afforded by a properly functioning quality assurance program.[86]

B. Issue 3(f)(iii) concerns a possible phenomenon known as "uplifting." In theory, uplifting may occur when a seismic event produces a high horizontal acceleration. In some circumstances this acceleration tends to produce a shift in the center of mass of a rigid structure, perhaps to the extent of causing a building tilt. In a rigid building such as the Diablo Canyon containment, if uplifting occurs, one side of the base mat would lift away from the underlying rock or soil thereby causing increased vertical acceleration in the structure. No specific analysis was done by the ITP or IDVP concerning the effect on equipment of increased vertical acceleration caused by uplift. Only the effect of uplift on the containment mat was studied by the ITP.[87] Joint intervenors and the Governor assert that the applicant should have analyzed the effects of uplift on equipment qualification in the Diablo Canyon containment.[88]

The uplift phenomenon is a relatively recent concept that evolved from discussions among seismologists, rather than from observation of seismic events.[89] It has never been identified as a source of actual damage to a structure and there is no NRC regulation or staff guidance requiring that the seismic analyses for a nuclear power plant include this phenomenon.[90]

The Governor's expert witness, Dr. Jose M. Roesset, opined that some uplifting of the Diablo Canyon containment would occur at the

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[86] We note that, in theory, a design quality assurance program will provide 100 percent review of the design work. The record is clear, however, that such a program can never assure that there will be no design errors. Anderson and Kaplan Tr. D-1176-81; Hubbard Tr. D-2130-31, D-2134-35; Apostolakis Tr. D-2376-77. Indeed, Appendix B only provides that the purpose of a quality assurance program is "to provide adequate confidence that a [safety-related] structure, system or component will perform satisfactorily in service." *See* 10 C.F.R. Part 50, Appendix B, Introduction. Here, the applicant's verification program has provided that level of confidence.

[87] White Tr. D-828.

The applicant's analysis of the possible uplift of the containment mat found that the maximum stress on the reinforcing steel was within allowable limits stated in the FSAR and Hosgri Report. Seed *et al.* Tr. fol. D-652 at 71-72.

[88] Gov. PF at 43-45; JI PF at 20-21.

[89] Seed Tr. D-687-88; Roesset Tr. fol. D-2206 at 6.

[90] White Tr. D-669-71, D-680-82; Seed Tr. D-684; Cloud Tr. D-1890; Polk Tr. D-2506.

peak ground acceleration established for the site.[91] According to Dr. Roesset, such uplift would amount to only a fraction of an inch and would cause only a small (approximately ten percent) increase in vertical acceleration. Such uplift also would result in a negligible increase in seismic displacement and velocity.[92] He noted further that most of the effects of uplift are beneficial.[93]

The applicant's expert witnesses were unwilling to concur in Dr. Roesset's opinion that uplifting of the Diablo Canyon containment would occur at design basis ground accelerations.[94] Rather, these experts were only willing to concede that uplift was possible at such accelerations.[95] But, like Dr. Roesset, the applicant's experts (as well as those of the staff) agreed that should uplift occur at Diablo Canyon it would amount to only a fraction of an inch and would cause only very small increases in the vertical acceleration of the reactor building.[96] Thus, there is agreement among all the expert witnesses that the effects of uplift on the vertical accelerations of the Diablo Canyon reactor building would be extremely small.

The equipment inside the containment is seismically qualified for a total acceleration obtained by taking the square root of the sum of the squares of each of three accelerations (two horizontal and one vertical).[97] A small increase in the vertical acceleration on the order of that resulting from any uplift of the Diablo Canyon containment (*i.e.*, ten to fifteen percent) would cause an insignificant increase in the total

[91] Roesset Tr. fol. D-2206 at 7-8.

In part, Dr. Roesset based his opinion on the work of R.P. Kennedy that indicated an increase in the response spectra of a high temperature gas-cooled reactor (HTGR) in the high frequency range, and on the work of J.P. Wolf that indicated the possibility of uplift of a "typical reactor" building at a peak ground acceleration of 0.167*g* if the containment were on a rock base. Roesset Tr. fol. D-2206 at 7. *See* Kennedy, Short, Wesley and Lee, *Effect of Non-Linear Soil Structure Interaction Due to Base Slab Uplift on the Seismic Response of a High-Temperature Gas-Cooled Reactor (HTGR)*, 38 Nuclear Engineering and Design, No. 3 (1976); Wolf and Skrikeru, Seismic Excitation with Large Overturning Moments: Projecting Base Mat or Lifting-off?, paper presented at the Conference on Structural Analysis, Design and Construction in Nuclear Power Plants, Porto Alegre, Brazil (Apr. 1978).

[92] Roesset Tr. D-2273-74, D-2276-77.

[93] Roesset Tr. fol. D-2206 at 8; Tr. D-2271.

[94] Seed Tr. D-687; White Tr. D-668-69; Holley Tr. D-1874-76; Biggs Tr. D-1881.

[95] *See e.g.*, White Tr. D-671, D-675; Seed Tr. D-687; Holley Tr. D-1874-75.

In general, the applicant's expert witnesses did not endorse Dr. Roesset's opinion that uplift would occur because, if the mathematical model was expanded to include all of the relevant factors, uplift would most likely not be shown to occur. In other words, the less detailed model relied upon by Dr. Roesset necessarily predicts the phenomenon. White Tr. D-668, D-671; Holley Tr. D-1874-76. *See also* Polk Tr. D-2503. In reality, uplift has never been found to occur in a structure. White Tr. D-669-70; Seed Tr. D-684. To model the uplift phenomenon properly would be an exceedingly complex and time-consuming task. Seed Tr. D-687-89, D-694; White Tr. D-682, D-691. Moreover, the Diablo Canyon reactor building base mat is constructed with a deep concrete key poured into a rock foundation that would have to be torn before uplift could occur. White Tr. D-691; Seed and White Tr. D-695-96.

[96] White Tr. D-672, D-682; Holley Tr. D-1876; Biggs Tr. D-1882-83; Miller Tr. D-2507-08; Kuo Tr. D-2504-05.

[97] Biggs Tr. D-1882-83; Kuo Tr. D-2514; Cloud Tr. D-1886-87.

acceleration obtained from the combination of the horizontal and vertical accelerations.[98] This result is reduced even more because the uplift acceleration is not in phase with the seismic vertical acceleration and cannot be considered additive to the peak vertical acceleration.[99] Moreover, the equipment inside the containment is already seismically qualified within margins more than sufficient to accommodate any increase in vertical acceleration as a result of uplift.[100] We find, therefore, that, even if uplift should occur, its detrimental effects would be insignificant. The applicant need not include as part of its seismic verification any seismic modeling and analysis of the effects of uplift on equipment inside the reactor building.

C. Issues 3(f)(iv) and (v) concern the modeling of soil springs for the Diablo Canyon auxiliary building. The term "soil springs" is applied to the methodology used in seismic analysis to represent motion resistance and damping characteristics of the foundation media around a structure. In other words, the soil media is assumed to act like a spring in a seismic event. Here, the Governor and joint intervenors complain that the soil properties used to establish soil spring constants were not properly specified and that the use of soil springs did not adequately account for all soil structure interactions.

At Diablo Canyon, the auxiliary building has foundations at elevations 85 feet and 100 feet above sea level. In modeling the building for the seismic reanalysis, the ITP assumed the base to be rigid at the lower (85-foot) elevation and soil springs were used to represent the intervening rock between that level and the 100-foot level. The Hosgri Report and the FSAR allow a rigid or fixed-base analysis for stiff rock, indicated by a shear wave velocity at or above 3500 feet per second (fps).[101] Soil springs are used to model less stiff soil.

The ITP, relying on data supplied by Harding Lawson Associates,[102] determined the shear wave velocity at 100 feet to be about 2700 fps, supporting the use of soil springs at this elevation.[103] The ITP had examined

[98] Biggs Tr. D-1882-83; Kuo Tr. D-2514.

[99] Biggs Tr. D-1885; Roesset Tr. D-2282.

[100] Cloud Tr. D-1886-87; Knight Tr. D-2512-14.

Dr. Roesset called for an analysis of the effects of uplift on the equipment inside containment for the "sake of completeness." Tr. D-2273-74. But he conceded that he did not know what equipment was in containment and how it was qualified for vertical acceleration. Tr. D-2214-15.

[101] *See* FSAR Section 3.7A at 6.

Soil stiffness may be determined from measured values of shear wave velocity of the soil.

[102] The engineering firm of Harding Lawson Associates did the major soils analyses (*i.e.*, geotechnical studies) for the Diablo Canyon site. A review conducted as part of the seismic verification program found that Harding Lawson Associates had not implemented a quality assurance program for the soil work for Diablo Canyon. App. Exh. 155, ITR 68, at 2.

[103] Seed *et al.* Tr. fol. D-652 at 20-21; White Tr. D-700-01.

the Harding Lawson data and found it to be reasonable and comparable to data obtained by other companies doing work involving the soil under the auxiliary building.[104] As part of its review, the ITP performed parametric studies in which the soil geometry and the soil springs under the 100-foot foundation were varied for a number of stiffness values. The shear wave velocities ranged from 6,000 fps (very rigid) to 2,000 fps (less rigid), with the value 2,775 fps serving as the base case.[105] The results of these analyses showed that there was little variation in the shear stresses for the auxiliary building walls and that generally the base case yielded the highest values.[106] In other words, the auxiliary building was found to be qualified for shear forces associated with all credible soil stiffness properties.

Because Harding Lawson Associates had not implemented a quality assurance program for their work at Diablo Canyon, the IDVP developed an extensive program to verify the soils work.[107] That program again verified the reasonableness and reliability of the soils data.[108] The IDVP also reviewed the ITP's auxiliary building analysis and found it acceptable.[109] We find that the ITP properly addressed the soil properties in its modeling of the auxiliary building and that the model used was appropriate.

The Governor and the joint intervenors assert, however, that the ITP erred by using soil springs in modeling the auxiliary building at the 100-foot elevation. They claim a fixed-base analysis at that elevation should have been used and that such an analysis would show increased shear wall forces that have not been analyzed.[110] The Governor's and joint intervenors' position is not supported by the evidence.[111] The foun-

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[104] White Tr. D-774; Seed and White Tr. D-811-13.

[105] App. Exh. 145, ITR 55, at 20, 21 (Table 3), 23 (Table 5); Seed and White Tr. D-700-06.

[106] White Tr. D-706; App. Exh. 145, ITR 55, at 23 (Table 5).

[107] App. Exh. 155, ITR 68, at 2, 4-5; Cloud Tr. D-1942-43, D-1997-99, D-2013-14.

[108] Cloud Tr. D-2002-03, D-3124; Cloud *et al.* [This panel consisted of R. Cloud, J. Biggs, M. Holley and R. Wray.] Tr. fol. D-1843 at 3-8.

[109] App. Exh. 90, IDVP Final Report, Vol. I, at 4.4.2-9 to -10; Cloud Tr. D-1848-49.

The IDVP performed parametric calculations similar to the ITP's to determine the spring constants and their results were in reasonable agreement with those of the ITP. App. Exh. 145, ITR 55, at 25; Cloud Tr. D-1905. The staff also concurred in the ITP's conclusions. Kuo *et al.* [This panel consisted of J. Knight, P. Kuo, H. Polk, C. Miller, A. Philippacopoulos, C. Costantino and P. Wang.] Tr. fol. D-2463 at 16-18.

[110] Gov. PF at 40-41; JI PF at 21-22.

[111] Although the Governor and the joint intervenors do not identify the source for their claim, they apparently reach their conclusion that the ITP should have used a fixed-base analysis for the 100-foot elevation by taking out of context from App. Exh. 145, ITR 55, at 24 (Table 7), "IDVP Soil Parameters," the IDVP's best estimate of 3500 feet per second shear wave velocity for the auxiliary building foundation material. They ignore, however, the IDVP's explanation of those soil parameters and the IDVP's conclusion that the values of the soil springs used by the ITP were acceptable. App. Exh. 145, ITR 55, at 25.

dation material at elevation 100 feet has a shear wave velocity of 2500 to 2700 fps (not 3500 fps as the Governor and the joint intervenors assert); accordingly, the ITP's use of soil springs in its modeling for that elevation is justifiable.[112] Moreover, the parametric studies performed as part of the seismic reanalysis demonstrate that there would be a negligible change in shear wall forces even if the foundation at the 100-foot elevation were to be considered fixed. All such forces are well within the margins for which the shear walls are qualified.[113]

D. Issue 3(o) questions the modeling of the fuel handling building. Specifically, it challenges the use of the translational and torsional response of the auxiliary building as input to the fuel handling building and the number of dynamic degrees of freedom used in the model.

The fuel handling building is, in essence, a small superstructure to the auxiliary building and will experience the motion of the auxiliary building during a seismic event.[114] The ITP therefore modeled the auxiliary building, including the fuel handling building, and then used the appropriate response of that building as input to a separate model of the fuel handling building in order to determine local responses.[115] The IDVP concluded that the modeling was consistent with good engineering

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[112] *See* p. 595, *supra*.

The Governor's expert agreed that if the soil properties printed in App. Exh. 145, ITR 55, at 23 (Table 5) were correct, then the ITP's analysis was proper. Roesset Tr. D-2217-19, D-2249-50. Subsequently, one of the applicant's expert witnesses rechecked those numbers and found them correct. Cloud Tr. D-3111.

[113] App. Exh. 145, ITR 55, at 21 (Table 3), 23 (Table 5); Cloud *et al.* Tr. fol. D-1843 at 3-9; Biggs Tr. D-1907-08; White Tr. D-713-19.

Alternatively, the Governor and the joint intervenors argue that the ITP should have considered a softer soil spring in modeling the auxiliary building because the material underlying the foundation has a shear wave velocity of only 1500 fps. They claim that, in the event of an earthquake, the auxiliary building would be subject to rotational effects (*i.e.*, a rocking motion) because the structure is embedded at different elevations in soils of widely varying stiffness. Gov. PF at 41-43; JI PF at 21-22. This argument, like the other, is unsupported by the evidence and footed on an inappropriate reference. The Governor and joint intervenors rely upon App. Exh. 155, ITR 68, Figure 15 at 81 to conclude that the material underlying the auxiliary building is soft. But this figure applies to the soil underneath the diesel fuel oil tanks, not the auxiliary building. App. Exh. 155, ITR 68, at 28-30; Cloud Tr. D-3110-11; White Tr. D-3136. Further, the seismic refraction tests used to produce the data in Figure 15 are not as reliable as cross-hole and up-hole testing techniques used to generate the data relied upon by the ITP. Cloud Tr. D-1996-2002, D-3112, D-3122, D-3125; White Tr. D-3136; Roesset Tr. D-2269. Accordingly, the ITP need not have considered such a soft soil spring in its modeling.

[114] Seed *et al.* Tr. fol. D-652 at 81-82; Cloud *et al.* Tr. fol. D-1843 at 3-18.

[115] Seed *et al.* Tr. fol. D-652 at 81-82; Cloud *et al.* Tr. fol. D-1843 at 3-18.

The seismic analysis of the auxiliary building, including the fuel handling building superstructure, was performed by using a lumped mass-spring model. Seed *et al.* Tr. fol. D-652 at 81. The model used a five percent eccentricity of mass to account for the effects of accidental torsion and the appropriate translational time-history was applied at its base. *Id.*

The fuel handling building was decoupled from the auxiliary building and analyzed separately using three-dimensional finite element models. The seismic input motions at the base of these models consisted of acceleration time-histories (translational and torsional) from the auxiliary building dynamic analysis developed at the center of mass at elevation 140 feet. The geometric eccentricity of the fuel handling building relative to the 140-foot elevation center of mass was accounted for by applying the translational time-history together with the eccentric distance times the torsional time-history. *Id.*

practice and acceptable.[116] The staff concurred in the IDVP's conclusion.[117]

In order to analyze the fuel handling building, the ITP first developed a large static model, which was then divided into two smaller dynamic models. The total number of degrees of freedom was reduced to make the analysis more manageable.[118] This reduction was accomplished by using standard industry procedures.[119] A sufficient number of dynamic degrees of freedom were included to determine adequately peak accelerations.[120] We find, therefore, that the modeling of the fuel handling building was appropriate.[121]

E. Issue 3(q) concerns the soils analyses for the buried diesel fuel oil tanks at the facility site. The soils analyses for the diesel fuel tanks, like those for the auxiliary building dealt with in issue 3(f)(iv), were done by Harding Lawson Associates.[122] They performed the original seismic qualification analyses for the diesel tanks. In 1983 as part of the ITP seismic verification, Harding Lawson reanalyzed them.[123] The IDVP then reviewed the Harding Lawson requalification analyses and conducted several alternative analyses including parametric studies covering a range of soil properties for the backfill around the diesel fuel tanks.[124]

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[116] Cloud *et al.* Tr. fol. D-1843 at 3-18 to -19.

[117] Kuo *et al.* Tr. fol. D-2463 at 21.

[118] Staff Exh. 36, SSER 18, at C.3-26; Seed *et al.* Tr. fol. D-652 at 81-83; Cloud *et al.* Tr. fol. D-1843 at 3-18 to -19.

[119] Cloud *et al.* Tr. fol. D-1843 at 3-18 to -19. *See also* NUREG-0675, Supplement No. 20, Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Dec. 1983) at C.3-6 to -7.

[120] Cloud *et al.* Tr. fol. D-1843 at 3-19.

[121] At the time of the hearing, no written confirmation of a certain aspect of the ITP's input into the model of the fuel handling building had yet been provided to the staff, as the staff had requested. Miller Tr. D-2528-29. On the basis of the applicant's oral representation, the staff concluded that the ITP procedures were acceptable. Kuo *et al.* Tr. fol. D-2463 at 20-21. Because no written confirmation had been received from the applicant, however, the Governor states that "no findings are yet possible." Gov. PF at 45. That is not the case. The applicant has fully met its burden of proof on this issue. The fact that the staff sought written confirmation from the applicant that separate time histories were applied at the base of each column in the fuel handling building model is irrelevant. Though it is without effect on our findings, we note that the written confirmation requested by the staff has since been provided by the applicant. *See* letter from J. Schuyler, PG&E, to G. Knighton, NRC, dated November 17, 1983, at 2-3. Moreover, by not filing adequate proposed findings on the issue, the Governor, in effect, has waived it. The joint intervenors filed no proposed findings on this issue. *See* note 18 and accompanying text, *supra.*

[122] Because Harding Lawson had not implemented a quality assurance program for their original work at Diablo Canyon, the IDVP performed an extensive review of the soils work for the fuel tanks and found it reasonable and acceptable. Cloud *et al.* Tr. fol. D-1843 at 3-23 to -24; App. Exh. 90, IDVP Final Report, Vol. II, at 4.9.2-1 to -2, -6 to -9; App. Exh. 155, ITR 68, at 28-30, 36-37, 41. That soils work was also reviewed by the ITP. White Tr. D-767, D-774; Seed Tr. D-770, D-772-73.

[123] App. Exh. 155, ITR 68, at 33-34.

[124] *Id.* at 34-40; Cloud *et al.* Tr. fol. D-1843 at 3-23.

The IDVP found the Harding Lawson work acceptable and concluded that the diesel tanks meet licensing criteria.[125]

The Governor and the joint intervenors assert, however, that the variation in the Harding Lawson soils data for the backfill around the diesel tanks, and the variation between the data for the rock underlying the fuel tanks and that under the auxiliary building, demonstrate that the original data are unreliable and should not be used for qualifying the tanks.[126] We disagree. The Harding Lawson soils data were checked and rechecked, and the IDVP's parametric studies demonstrate that the qualification of the fuel tanks is not sensitive to the variation in the backfill soil properties about which the Governor and the joint intervenors complain.[127]

The properties of the rock under the diesel tanks and auxiliary building vary because the structures are in different locations.[128] The seismic analyses for the fuel tanks included properties for the underlying rock obtained using seismic refraction tests. These tests are considered to be relatively unreliable for the Diablo Canyon site, and would give results indicating the rock is less rigid (softer) than is actually the case.[129] The use of soft rock properties in the seismic analysis results in greater calculated strains in the tanks than would the use of more rigid rock. Thus, the analyses performed were conservative.[130] Additionally, the rock below the tanks has a small effect on the tanks' response.[131] Accordingly, we find that the data used for the diesel fuel tank analyses were adequate to demonstrate that the tanks are properly qualified.[132]

F. Like issue 3(q), issue 3(r) also questions the soils analysis of backfill material.[133] In particular, the Governor and the joint intervenors challenge the soils analysis of the backfill covering the circulating water

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125 App. Exh. 155, ITR 68, at 41; App. Exh. 90, IDVP Final Report, Vol. II, at 4.9.2-6 to -9; Cloud *et al.* Tr. fol. D-1843 at 3-24.

126 Gov. PF at 46-48; JI PF at 22-23.

127 App. Exh. 155, ITR 68, at 29, 37, 41; Cloud *et al.* Tr. fol. D-1843 at 3-23 to -24; Cloud Tr. D-1982-83, D-1988.

128 Cloud Tr. D-1998, D-3112, D-3122-23; White Tr. D-3136.

129 Cloud Tr. D-1998-99; *see* note 113, *supra.*

130 Cloud Tr. D-2001, D-3124-25.

131 App. Exh. 155, ITR 68, at 41.

132 The joint intervenors also question the soil properties underlying the diesel fuel oil tanks. The joint intervenors have misapprehended the data. They compare an equation for compressional wave velocity for the rock under the diesel fuel tanks (App. Exh. 155, ITR 68, at 38) with one that expresses shear wave velocity under the circulating water intake conduits and auxiliary saltwater piping (App. Exh. 155, ITR 68, Figure 25 at 91). These equations must be converted to the same component of wave velocity before any comparision is made.

133 Gov. PF at 48-52; JI PF at 23-24.

intake (CWI) conduits and the auxiliary saltwater (ASW) piping used in the seismic qualification of these components.[134]

The soils analysis for the ITP's seismic qualification of the CWI conduits and the ASW piping was performed by Harding Lawson Associates. As we have previously found, the Harding Lawson soils work was reviewed by the IDVP and found acceptable.[135] In this instance, Harding Lawson took test borings of the backfill and then performed laboratory tests on the specimens to arrive at soil property values.[136] The IDVP reviewed the seismic analyses and found that the Harding Lawson soil and rock properties were acceptable and that the CWI conduits and ASW piping meet licensing criteria.[137]

The Governor and the joint intervenors charge that the soils data do not represent the actual backfill at the site over the conduits and piping, because no correction factor was applied to the data to adjust for the sample disturbance that occurs when laboratory values for soil properties are used. If a correction factor were applied, they claim, the soil properties would be different than the ones used in the qualification analysis.[138] The properties of the backfill over the CWI conduits and ASW piping about which the Governor and the joint intervenors complain are negligible factors, however, in the seismic qualification of these components.[139] The conduit and pipes are surrounded by rock on the sides and bottom (*see* note 134, *supra*) and the rock determines the seismic response of these components, not the backfill on top of them.[140] Because the effect of the backfill on the seismic response of these compo-

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[134] Each of the two circulating water intake conduits (one for each unit) is a 16 ft. by 30 ft. reinforced concrete structure containing two parallel, essentially square, tunnels (approximately 12 ft. by 12 ft.). The conduits parallel one another and extend approximately 1600 feet from the turbine building to the intake structure. The conduits are located in trenches excavated into rock and covered on top with some twenty feet of backfill. Two 24-inch diameter steel auxiliary saltwater pipes, placed one over the other, run parallel to and on one side of each of the CWI conduits in a narrow, shallow trench. One side of the trench is formed by the concrete sidewall of the CWI and the other by rock. The bottom of the trench consists of a concrete lip projecting from the CWI concrete sidewall. The ASW pipes are attached to this concrete lip at 40-foot intervals and are surrounded in the trench by compacted sand. The ASW pipes are then covered with the same backfill material as the CWI conduits. App. Exh. 155, ITR 68, at 42 and Figure 21, at 87; Seed Tr. D-837-40.

[135] *See* p. 596, *supra*.

[136] App. Exh. 155, ITR 68, at 42-45.

As part of the seismic analyses, the data were used to plot soil shear modulus against strain and soil damping ratio as a function of strain. App. Exh. 155, ITR 68, at 89-90 (Figures 23 and 24). Those test data were then matched to a standard soils curve (*i.e.*, Seed & Idriss 1970 sand curve) and that curve was employed in the qualification analyses. Seed *et al.* Tr. fol. D-652 at 85-86.

[137] App. Exh. 90, IDVP Final Report, Vol. II, at 4.9.2-10; App. Exh. 155, ITR 68, at 51.

[138] Gov. PF at 50-52; JI PF at 23-24.

[139] Seed *et al.* Tr. fol. D-652 at 86; White Tr. D-805; Seed Tr. D-836-40, D-3142-43.

[140] Seed *et al.* Tr. fol. D-652 at 86; Seed Tr. D-836-39, D-3142-43.

Indeed, the Governor's expert conceded that the effect on the seismic response of backfill over the conduits and piping was small. Roesset Tr. D-2254, D-2256.

nents is insignificant, it was acceptable for the ITP and IDVP to use the Harding Lawson data without correction factors in the seismic qualification of the CWI conduits and the ASW piping.

G. The joint intervenors allege in issue 4(i)(1) that the IDVP unjustifiably accepted a deviation from the FSAR fire protection licensing criteria for the room containing the motor driven auxiliary feedwater (AFW) pumps, because there is a large pipe chase covered with a grate in the ceiling of that room.[141] This purported deviation was discovered by the IDVP during its review of the auxiliary feedwater system and resulted in the issuance of EOI 8038. The EOI was issued because the description of the AFW pumproom in the FSAR, if read literally, was subject to misinterpretation (*i.e.*, the ceiling was described only as a 2-foot thick concrete slab), and the existence of the ceiling pipe chase appeared to violate the FSAR fire zone separation licensing criteria.[142]

The ITP responded to the IDVP's concern by submitting a fire hazards analysis demonstrating that a fire is unlikely to propagate through the pipe chase because of the actual combustible loading in the pumproom, the movement of air through the ceiling opening and the curbing surrounding the pipe chase in the area above. The ITP analysis confirmed that the plant was originally designed to meet the fire zone separation criteria with the open pipe chase in the AFW pumproom ceiling.[143] The IDVP concurred in the ITP's analysis. It agreed that the FSAR fire zone commitment had been met and that a fire in this area of the facility would not adversely affect the safe shutdown functions of the AFW system.[144] On the basis of the uncontroverted evidence of the origins of this purported deviation and the ITP's fire propagation analysis, we find — like the IDVP — that there is no deviation from licensing criteria because of the open pipe chase in the AFW pumproom.[145]

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[141] The Governor failed to file proposed findings of fact on issue 4(i)(1). *See* note 18, *supra* and accompanying text.

[142] Krechting and Cooper Tr. fol. D-2040 at 4-16; App. Exh. 110, ITR 18, at 4-1, 5-3. *See* "Fire Protection Review, Units 1 and 2 Diablo Canyon Site" (Amendment No. 51 to operating license application) at 4-18 to -19.

[143] Connell *et al.* [This panel consisted of R. Anderson, E. Connell and G. Moore. W. Vahlstrom was subsequently added to the panel. Tr. D-531.] Tr. fol. D-487 at 22; App. Exh. 110, ITR 18, at 4-1, 5-3.

[144] App. Exh. 110, ITR 18, at 4-1, 5-3.

[145] The joint intervenors presented no witnesses or documentary evidence on this issue and did not cross-examine any of the applicant's witnesses on it. They claim, nevertheless, that the FSAR fire zone separation license criteria are not met based solely on a selective use of staff witness Kubicki's answers to their questions on cross-examination. *See* JI PF at 24-25. They assert that the existence of the open pipe chase precludes a complete fire barrier. Because the staff had no knowledge of the air flow patterns for the AFW pumproom, and it is possible for the products of combustion to travel through the pipe chase thereby propagating fire to another part of the plant, the joint intervenors contend there is no assurance the licensing criteria are met. *Id.* In addition to its other flaws, the joint intervenors' argument ignores the uncontradicted testimony of the applicant's witnesses that there are insufficient combustibles

*(Continued)*

H. Issue 4(l) addresses the adequacy of the applicant's analysis of possible jet impingement effects on the design and qualification of safety-related equipment and piping inside the containment. Because a break or crack in a line carrying high energy steam or water might result in damaging jets from the failed pipe, the NRC has long required that safety-related structures, systems and components in the vicinity of potential break locations be analyzed for (and, if necessary, protected from) the effects of jet impingement.[146] In like manner, the Diablo Canyon FSAR requires that the applicant protect all safety-related structures, systems and components from the damaging effects of jet impingement.[147]

When the IDVP reviewed Diablo Canyon records, however, no documentary evidence of jet impingement analyses for safety-related systems, structures and components inside containment was found. Consequently, EOI 7002 was issued.[148] In response to this EOI, the ITP performed an extensive analysis of jet impingement effects of high energy line breaks.[149] That ITP analysis was verified by the IDVP on a sampling basis. In addition, the verification included a review of the ITP reanalysis procedure, a review of the ITP field review (including an independent walkdown to verify identification of safety-related targets) and a review of the ITP safety evaluation of impinged targets.[150] On the basis of its verification (reported in ITR 48), the IDVP concluded that the ITP analysis met the FSAR licensing criteria.[151]

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in the pumproom for a fire to propagate. Connell *et al.* Tr. fol. D-487 at 22. More important, however, is the fact that we do not rely upon the staff's testimony to reach our conclusion that there is no deviation from the FSAR fire zone separation criteria.

We note that the staff did not review the IDVP's analysis in reaching its conclusion that the open pipe chase in the ceiling of the AFW pumproom did not present a deviation from licensing criteria. Kubicki Tr. D-2873. Rather, the staff conducted an independent review of the Diablo Canyon fire protection program using 10 C.F.R. Part 50, Appendix R (Fire Protection). As a result, the staff concluded that the pipe chase opening did not represent a significant degradation of the level of fire safety in the room and that any fire propagation would not represent a significant threat to adjoining areas. This conclusion was based on the defense in depth concept, administrative controls limiting combustibles, the existence of firewalls and the availability of automatic and manual fire protection systems. Kubicki Tr. D-2874-75.

146 *See* 10 C.F.R. Part 50, Appendix A, General Design Criterion (GDC) 4.

147 FSAR Section 3.6.

148 App. Exh. 140, ITR 48, at 3-1.

It should be noted that the degree of analysis and documentation of jet impingement required today is greater than for earlier-designed plants such as Diablo Canyon. Connell *et al.* Tr. fol. D-487 at 26.

149 Connell *et al.* Tr. fol. D-487 at 25-26; Krechting and Cooper Tr. fol. D-2040 at 4-21 to -22.

150 Krechting and Cooper Tr. fol. D-2040 at 4-21 to -22.

151 App. Exh. 140, ITR 48, at 7-1.

After identifying several potential problems and seeking additional information from the applicant, the staff agreed that the applicant met the FSAR jet impingement licensing criteria. Staff Exh. 36, SSER 18, at C.4-29; Staff Exh. 37, SSER 19, at C.4-2. But the staff has yet to close its review of the applicant's jet impingement analysis. The matter still under investigation, however, is not pursued in issue 4(l). Staff Exh. 37, SSER 19, at C.4-2.

The Governor and the joint intervenors challenge the ITP analyses and the IDVP conclusion that the applicant has met the FSAR jet impingement licensing criteria, claiming that analyses were not performed for postulated breaks in each line inside the containment required by the FSAR.[152] In accordance with its interpretation of the FSAR, the ITP performed jet impingement analyses for high energy lines with a temperature above 200°Fahrenheit(F) *and* pressure above 275 pounds per square inch gage (psig).[153] The Governor and the joint intervenors assert that the FSAR actually requires jet impingement analysis for postulated breaks in all lines exceeding 200°F *or* 275 psig. The difference between the two interpretations results in the exclusion of three lines inside containment from the ITP's analysis that the Governor and the joint intervenors would include.[154] Thus, those three lines were not analyzed as part of the ITP jet impingement analyses. These three lines were, however, "looked at" informally by the ITP.[155]

We believe the most prudent interpretation of the FSAR is that one which requires jet impingement analysis on lines having a temperature above 200°F *or* a pressure above 275 psig.[156] Therefore, in order to comply with the FSAR licensing criteria, the applicant must formally analyze (*i.e.*, in the same fashion it analyzed the other lines inside containment) the three lines identified by its witness.[157] The applicant must report the results of its analyses to the staff and, if necessary, make any appropriate modifications prior to commercial operation.[158]

I. In issue 4(t), the joint intervenors assert that the IDVP accepted without proper justification a deviation from licensing criteria because the short circuit current for the circuit breakers on three 4160 volt (V)

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[152] Gov. PF at 54-56; JI PF at 25-27.

[153] Connell *et al.* Tr. fol. D-487 at 25-26; Connell Tr. D-584.

[154] Connell Tr. D-613-14.

[155] Connell Tr. D-616-17.

[156] The most rational, technically based criteria for postulating line breaks for the purpose of jet impingement analysis appear in the FSAR section that addresses lines outside of containment. Recognizing that jets may result from line cracks or breaks, this section requires such failures to be postulated for lines with service temperature above 200°F, or design pressure exceeding 275 psig. *See* FSAR at 3.6-16 to -17. This same rationale should control the analyses inside containment.

[157] Because the applicant's witness Connell did not elaborate upon his remark that these three lines "have been looked at, but they are not part of the formal jet program re-analysis . . ." (Tr. D-616-17) and there is no other evidence concerning these lines, we are unable to conclude on the basis of the record evidence that the applicant has complied with the FSAR licensing criteria.

[158] Although the applicant's witness did not identify the three lines inside the containment by name, it appears the lines were identified by the applicant in its response of October 12, 1983 to the NRC staff concerning the ITP jet impingement analyses. *See* letter from J. Schuyler, PG&E, to D. Eisenhut, NRC, dated Oct. 12, 1983 at 1-2; *see also* Staff Exh. 37, SSER 19, at C.4-2. If our assumption is correct that the lines are the excess letdown line, reactor coolant pump seal vent and leakoff lines, and reactor coolant pump seal water injection line, then it appears that the lines are small diameter ones with relatively low energy content that would not be expected to fail or to produce high energy jets.

safety-related switchgear buses was calculated to be greater than the nameplate ratings on the breakers.[159] This situation was identified by the IDVP during the nonseismic design review of the 4160 V electrical distribution system and became the subject of EOI 8022.[160] The nameplate rating of the 4160 V circuit breakers is listed as 33 kiloamperes (kA), but the calculated short circuit current that the breakers might be required to interrupt is 39kA.[161] The IDVP declared the matter resolved when the ITP provided it with 1976 test data and a letter from the breaker manufacturer indicating the breakers can be relied upon to interrupt current up to 45kA.[162]

The joint intervenors object to the IDVP's resolution of this matter.[163] They assert that breakers are normally warranted only for the nameplate rating and, because it is not known whether the manufacturer is willing to warrant greater capacity for the breakers, we should find that the applicant's failure to install breakers with an adequate nameplate rating constitutes a violation of licensing criteria.

The FSAR, however, requires only that the applicant protect the emergency power supply with circuit breakers adequate to interrupt the calculated short circuit current.[164] Here, the manufacturer's 1976 test data demonstrate that the breakers in question will perform above the nameplate rating and interrupt the required short circuit current.[165] Moreover, in February 1983 the manufacturer explicitly responded to the applicant's inquiry concerning the interrupting capacity of the breakers stating that the breakers in question will interrupt 45kA.[166] It is clear, therefore, that the nameplate rating of the breakers in question is only a nominal rating and that the breakers will perform their intended function.[167] Accordingly, we find that the IDVP did not accept any deviation from licensing criteria.

J. In issue 5, the Governor and the joint intervenors charge that the applicant's verification program has failed to substantiate that the Diablo

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[159] The Governor failed to file proposed findings of fact on issue 4(t). *See* note 18 and accompanying text, *supra*.

[160] App. Exh. 116, ITR 24, at 5-1.

[161] Moore Tr. D-524.

[162] Krechting and Cooper Tr. fol. D-2040 at 4-30; Krechting Tr. D-2052-55.

[163] JI PF at 27-28.

[164] *See* FSAR Section 3.1.

[165] Connell *et al.* Tr. fol. D-487 at 35; Moore Tr. D-524-25; Vahlstrom Tr. D-532.

[166] Krechting Tr. D-2054-55.

The question whether the manufacturer will warrant the breakers is, in fact, irrelevant, because nothing in the appropriate FSAR licensing criteria concerns manufacturer warranties. Hence, it is unnecessary for us to reach the question whether the manufacturer's 1983 written response to the applicant's inquiry is an express warranty.

[167] Moore Tr. D-524-26.

Canyon facility, as built, conforms to the actual design drawings and analyses.[168] In particular, they assert that a number of past deficiencies in the PG&E program, combined with more recent purported lapses in configuration control uncovered by the IDVP, establish the applicant's continuing failure to reconcile design documents with the plant as built. Although the evidence indicates a past weakness in this area, the applicant's revised configuration control procedures under which all modifications have been done, coupled with the extent of the verification efforts of the ITP and the IDVP, demonstrate that the applicant's reconciliation of design documents is in conformity with Appendix B.[169]

As part of its seismic design verification program, the ITP performed field walkdowns of the Diablo Canyon structures, equipment, piping, and electrical, pipe and HVAC supports to ensure that the design documents of record reflected the actual physical conditions of the plant. Any deviations identified by the ITP were incorporated into the design documents of record.[170] Similarly, as part of its nonseismic review of design pressures and temperatures, and redundant electrical circuits, the ITP conducted field verifications of the design documentation of PG&E designed safety-related systems.[171]

Further, the applicant modified its configuration control procedures in 1981 and again in 1983 to improve past weaknesses in reconciling design documents with the plant as built.[172] The present procedure (Engineering Department Procedure 3.6 ON) deals with the initiation, processing, approval and documentation of design changes during the operating life of the plant. Specifically, it requires Priority I drawings of design changes (*i.e.*, documents pertaining to safety-related structures, systems and components that are required for the safe operation and maintenance of the plant) be revised to reflect as-built conditions within thirty days of the engineering department's acceptance of the design change completion package. All other drawings must be revised within ninety days.[173] The modification work performed at the site has conformed to this new procedure.[174] Experience under the procedure has shown that the construction department generally provides to the engineering department the as-built information within two weeks of the

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[168] Gov. PF at 56-59; JI PF at 28-33.

[169] 10 C.F.R. Part 50, Appendix B, III and VI.

[170] Anderson *et al.* Tr. fol. D-224 at 31; Moore Tr. D-363-64.

[171] Anderson *et al.* Tr. fol. D-224 at 31; Moore Tr. D-363-64. *See* App. Exh. 90, IDVP Final Report, Vol. II, at 4.8.3-1, 4.8.6-1.

[172] Moore Tr. D-362; App. Exh. 161, Engineering Department 3.6 ON and 3.7.

[173] App. Exh. 161, Procedure 3.6 ON (May 14, 1983) at 1, 10 and Procedure 3.7 at Attachment A; Moore Tr. D-348.

[174] Anderson *et al.* Tr. fol. D-224 at 32.

completion of the modification. The engineering department's acceptance requires one to three additional weeks depending on the nature of the modification.[175]

The IDVP also performed extensive field inspections to verify that the plant, as analyzed, is in conformity with the plant, as built, respecting both its seismic and nonseismic design. In its initial reviews of the seismic design of the facility, the IDVP conducted field verifications to validate the physical configurations. This verification was repeated on a sampling basis after the ITP's seismic reanalysis and the completion of necessary modifications.[176] The IDVP authenticated the as-built condition of the auxiliary feedwater system, control room ventilation and pressurization system, and the 4160 V electrical distribution system (*i.e.*, the nonseismic systems it reviewed). It verified samples of the nonseismic review work performed by the ITP at the IDVP's direction and it substantiated the as-built condition of all modifications resulting from the IDVP's nonseismic verification program.[177] On the bases of its verification efforts, the IDVP concluded that, with respect to the portions of the facility it field verified, the as-built plant properly implemented the essential design elements.[178] Finally, the IDVP audited the applicant's process for controlling the update of engineering documents which included both the method for controlling design changes and the update of documents to reflect the as-built condition. It concluded that the program was being effectively implemented.[179]

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[175] Moore Tr. D-354-56, D-360-61.

[176] Cooper *et al.* Tr. fol. D-1459 at 5-2 to -4; Anderson *et al.* Tr. fol. D-224 at 31-32; App. Exh. 142, ITR 50, at 17-19, 24; App. Exh. 143, ITR 51, at 7, 19; App. Exh. 144, ITR 54, at 5; App. Exh. 145, ITR 55, at 46; App. Exh. 146, ITR 56, at 33; App. Exh. 147, ITR 57, at 22; App. Exh. 148, ITR 58, at 18; App. Exh. 149, ITR 59, at 3-3; App. Exh. 150, ITR 60, at 8; App. Exh. 151, ITR 61, at 8-9, 13-15; App. Exh. 152, ITR 63, at 14-19; App. Exh. 153, ITR 65, at 8. *See also* App. Exh. 128, ITR 36, at 4-9; App. Exh. 130, ITR 38, at 2-1, 3-2 to -6, 4-3, 6-1.

[177] Cooper *et al.* Tr. fol. D-1459 at 5-1 to -2; Anderson *et al.* Tr. fol. D-224 at 32; App. Exh. 106, ITR 14, at 3-9 to -11; App. Exh. 110, ITR 18, at 3-1 to -2; App. Exh. 111, ITR 19, at 7; App. Exh. 112, ITR 20, at 2-3, 6-3; App. Exh. 113, ITR 21, at 2-2, 3-1; App. Exh. 114, ITR 22, at 2-2, 6-2; App. Exh. 115, ITR 23, at 3-1 to -8; App. Exh. 116, ITR 24, at 1-2, 3-4 to -5; App. Exh. 117, ITR 25, at 3-1 to -4; App. Exh. 118, ITR 26, at 3-1 to -4; App. Exh. 119, ITR 27, at 3-1 to -3; App. Exh. 120, ITR 28, at 3-1 to -4; App. Exh. 140, ITR 48, at 6-9 to -14; App. Exh. 141, ITR 49, at 4-1, 5-1.

[178] Cooper *et al.* Tr. fol. D-1459 at 5-4.

[179] Cooper *et al.* Tr. fol. D-1459 at 5-3 to -4; Reedy Tr. D-1640.

The Governor questions the IDVP's conclusion that the applicant's procedure is being effectively implemented. He asserts that the initial audit was unable to substantiate the implementation of the applicant's design control process because of a lack of information and that the follow-up audit did not attempt to verify the procedure because it was limited solely to verifying the correction of a number of other specific deficiencies found in the initial audit. Gov. PF at 58-59. The initial audit was unable to verify the process. Reedy Tr. D-1636; Gov. Exh. 48 at 33; Gov. Exh. 49 at 33. The follow-up audit was not limited to the matters claimed by the Governor. The applicant's quality assurance expert was emphatic that the follow-up audit specifically verified the efficacy of the applicant's process. Reedy Tr. D-1636-37. We credit that testimony. Further, the applicant's primary difficulty in the area of configura-

*(Continued)*

The Governor and the joint intervenors, however, point to a number of purported as-built discrepancies reported by the IDVP in various ITRs and assert that, as in the case of applicant's past weaknesses in this area,[180] these errors demonstrate that the applicant's configuration control process is still inadequate.[181] None of these asserted deficiencies had any quality assurance implications or demonstrated a pattern of inadequate configuration control procedures.[182] Indeed, the Governor's expert upon whose claims the Governor's assertions are based, conceded that, in general, the applicant's as-built drawings reflect the actual physical configuration of the plant.[183]

The Governor and the joint intervenors also cite these as-built discrepancies as evidence that certain analyses did not conform to the as-built configuration of the plant.[184] In some instances, however, the discrepancy was the result of a worst-case assumption being used in the analysis which would not necessarily reflect as-built conditions.[185] In a large majority of the cases cited as examples of configuration control discrepancies, the IDVP determined that the plant's licensing criteria were met when the as-built condition was analyzed. In a few instances, modifications were required. Our review of these discrepancies reveals that many of them were attributable to modeling differences. Further, our review leads us to conclude, as did the IDVP, that this type and number of discrepancies are not unusual for the scope of activities undertaken.[186] Nor do we believe these discrepancies represent a pattern of inadequate configuration control. Accordingly, we are satisfied that

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tion control was its inability to revise affected documentation in a timely manner. Gov. Exh. 11 at 10. Thus, we do not view the applicant's May 1983 amendment of engineering procedure 3.6 ON to prescribe thirty- and ninety-day limits on conforming documents (*see* pp. 605-06, *supra*) as inconsistent with the IDVP's March 1983 audit conclusion that the configuration control process was being effectively implemented.

180 As we previously stated, the evidence indicates the applicant had difficulties with configuration control. Gov. Exh. 11, Institute of Nuclear Power Operations Startup Assistance Visit to Diablo Canyon Nuclear Power Plant (Feb. 12, 1982), at 10; Moore Tr. D-361-62; Morrill Tr. D-2948-49. We note, however, that a number of the documents relied on by the Governor and joint intervenors to establish these past deficiencies fail in that regard. Their reliance on a Brookhaven National Laboratory analysis of vertical response of the containment annulus structure (JI Exh. 130) is misplaced. Rather, that report uncovered a modeling discrepancy, not an as-built discrepancy. JI Exh. 130 at 11; Knight Tr. D-2948. Similarly, Gov. Exh. 36 (EDS Nuclear, Inc. Project Summary Report (June 7, 1982)) does little to enhance their position. That report describes an EDS review of the quality control manuals of each of the applicant's departments to determine the self-sufficiency of each manual. *See* p. 616, *infra*. The EDS review was not an audit of the applicant's configuration control process. de Uriarte Tr. D-3148-49; Stokes Tr. D-3189.

181 Gov. PF at 57; JI PF at 29-31. *See* Hubbard Tr. fol. D-2084 at Tables 5-1 and 8-1.

182 Reedy Tr. D-1640-41; Morrill Tr. D-2948-49.

183 Hubbard Tr. D-2157.

184 Gov. PF at 57; JI PF at 29-31; *See* Hubbard Tr. D-2156.

185 Hubbard Tr. D-2157.

186 App. Exh. 90, IDVP *Final Report*, Vol. III, at 5.6-4.

applicant's reconciliation of design documents with the facility and with the design analyses is in compliance with the Commission's regulations.

K. In issue 6, the joint intervenors and the Governor charge that the applicant failed to verify that the design of Westinghouse-supplied, safety-related equipment met licensing criteria. Westinghouse was the vendor of the nuclear steam supply system (NSSS) at Diablo Canyon. As part of the verification program, the IDVP reviewed the Westinghouse-PG&E interface for the use of Hosgri spectra, but the applicant's verification program did not specifically validate the qualification of Westinghouse-supplied equipment. Accordingly, the joint intervenors and the Governor claim there is no meaningful assurance that the Westinghouse design of safety-related equipment at Diablo Canyon meets applicable licensing criteria.[187]

Contrary to this claim, however, the assurance that the Westinghouse-supplied equipment meets licensing criteria is provided by the Westinghouse quality assurance program. That program was sufficient during the time the NSSS was designed and at subsequent times when the Hosgri spectra reevaluations were performed. Consequently, the applicant's verification efforts were not deficient even though the scope of its program did not include review of Westinghouse-supplied equipment.

Inasmuch as the construction permit for the first Diablo Canyon unit was issued in 1968, much of the Westinghouse design work on the NSSS took place prior to the promulgation of the agency's quality assurance regulations, 10 C.F.R. Part 50, Appendix B.[188] During that period Westinghouse nuclear design work was carried out in compliance with the requirements of MIL Q 9858, which was the quality assurance specification used by the navy nuclear program and includes most of the requirements later incorporated into Appendix B.[189] Subsequent to the issuance of Appendix B, the Westinghouse program was conformed to the regulation but this modification did not change the basic characteristics of the Westinghouse program.[190]

The Westinghouse quality assurance program has been audited many times by utilities, architect-engineers and professional organizations, as well as the NRC.[191] Indeed, a number of NRC audits of the Westinghouse program occurred while the vendor was performing the reanalysis of the Diablo Canyon NSSS for the Hosgri spectra in the late 1970's and

[187] JI PF at 33-36; Gov. PF at 59-62.
[188] Haass Tr. fol. D-2906 at 2.
[189] Kreh Tr. D-1151.
[190] *Id.* at D-1131.
[191] *Id.* at D-1129-31.

then again in the early 1980's. There is no record of unsatisfactory performance.[192] In addition, Westinghouse has designed the NSSS for some fifteen, four-loop nuclear power plants similar to Diablo Canyon which have been licensed by the NRC.[193]

The Governor and the joint intervenors point, however, to several asserted design errors at Diablo Canyon which they claim proves the inadequacy of Westinghouse's quality assurance program.[194] First, they point out that Westinghouse inappropriately used tau-filtered spectra[195] instead of unfiltered spectra in its design reanalysis for the Hosgri spectra. As a result of the IDVP interface review, two areas where Westinghouse inappropriately applied tau-filtered spectra in the Hosgri reanalysis were discovered. But this was a communication (*i.e.*, interface) problem between the applicant and Westinghouse, not a problem with the vendor's quality assurance program. Once the information was interpreted by Westinghouse, it was applied in analyses in these two areas consistent with the vendor's properly functioning quality assurance program.[196]

Second, the Governor and the joint intervenors assert that a review by Brookhaven National Laboratory[197] of the IDVP interface verification found that thirty percent of the samples reviewed by the IDVP contained errors. This claim misconstrues the Brookhaven report. That report simply states that in thirty percent of the samples, the Westinghouse spectra did not completely envelop the Hosgri spectra.[198] Because Westinghouse has a generic seismic design, site specific spectra may exceed

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[192] *Id.* at D-1089, D-1114, D-1116, D-1129; Wiesemann Tr. D-1115.

[193] Hoch *et al.* [This panel consisted of J. Hoch, R. Wiesemann and E. Kreh] Tr. fol. D-1088 at 3-4.

Additional confirmation of the adequacy of the Westinghouse quality assurance program is provided by the IDVP's recent verification of the PG&E-Westinghouse interface for the transmittal and use of the Hosgri spectra. The IDVP found that the appropriate information had been transmitted to Westinghouse and that, with one minor exception, the vendor correctly implemented the Hosgri data in their qualification and evaluation process. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.3-1; App. Exh. 103, ITR 11, at 18. Similarly, as part of its nonseismic review of the AFW system, the IDVP again examined this interface and concluded that Westinghouse correctly used the applicant-calculated design parameters provided to the vendor for accident analyses. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.3-1 to -2; App. Exh. 114, ITR 22, at 3-1, 3-4.

[194] Gov. PF at 60; JI PF at 33-35.

[195] For an explanation of the tau effect, *see Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-644, 13 NRC 903, 962-65 (1981), *petitions for review denied*, CLI-82-12A, 16 NRC 7 (1982).

[196] App. Exh. 103, ITR 11, at 18-19; Cooper *et al.* Tr. fol. D-1459 at 6-1 to -2; Wiesemann and Kreh Tr. D-1136-41.

The Governor also charges that the Westinghouse quality assurance program was deficient because there was inadequate identification of the specific Diablo Canyon unit for design information transmitted between the applicant and Westinghouse. The NSSS vendor, however, had its own number and letter designation system for documents that distinguished between the two units. Reedy Tr. D-1650.

[197] Gov. Exh. 12, Summary and Evaluation Report, PG&E-Westinghouse Seismic Interface Review.

[198] *Id.* at 4-5.

the generic ones — the situation noted in the Brookhaven review. Where that occurs, the affected equipment is specifically evaluated by Westinghouse to ensure conformance with the site specific spectra.[199]

Finally, the Governor and the joint intervenors charge that recent modifications to the main control board were necessary because of errors by Westinghouse in its original seismic qualification analyses. The modifications, however, were solely the result of changes in the seismic floor response spectra for the auxiliary building.[200]

Thus, none of the matters asserted by the Governor and the joint intervenors demonstrates that the Westinghouse quality assurance program was not adequate at the time of the original NSSS design or the subsequent reanalyses for the Hosgri spectra. The applicant's verification effort was not flawed by its exclusion of Westinghouse-supplied equipment and the verification program could justifiably rely upon the existence of the Westinghouse quality assurance program to ensure the adequacy of the nuclear steam supply system.

L. In issue 7, the Governor and joint intervenors claim that the verification program did not identify the root causes of the failures of the applicant's design quality assurance program and did not determine if such failures raise generic concerns. They correctly assert that without the identification of the causes of the various design errors, and a determination whether the errors have generic implications, there can be no confidence that further design errors do not exist.[201]

The root causes of the failures in the Diablo Canyon's design quality assurance program have been, however, adequately identified and analyzed as part of the applicant's verification efforts. In November 1981, the applicant began "lookback" reviews of its own quality assurance program and that of its service-related contractors.[202] From these reviews, the applicant determined that the basic causes for its own quality assurance failures were its inadequate control of changes in FSAR descriptions, inadequate control of documents, and inadequate documentation of design inputs.[203] For service-related contractors, the applicant found the basic causes were PG&E's failure to require quality assurance controls prior to mid-1978, its failure to control transmitted information, its inadequate record disposition and its inadequate inter-

[199] Wiesemann Tr. D-1121, D-1135-36.
[200] Hoch Tr. D-1122-23.
[201] Gov. PF at 62-68; JI PF at 36-37.
[202] Dick *et al.* Tr. fol. D-847 at 1-2.
[203] *Id.* at 3.

face control.[204] In response to the basic causes identified by its review, the applicant then took appropriate corrective action.[205]

The IDVP and the ITP also evaluated the causes of the errors and deficiencies that were discovered in the design of Diablo Canyon. In addition to a group of isolated random causes, the IDVP identified two basic reasons for design errors: failure to control design interfaces and inadequate documentation of the original and revised design.[206] Further, the IDVP concluded that several factors related to the fact that the plant was designed over a fifteen year period during which requirements, criteria and engineering techniques were changing also contributed to design problems.[207] The IDVP then evaluated the cause of each EOI and generally documented this evaluation in the EOI files.[208] Each EOI was also reviewed for quality assurance implications.[209] Because the IDVP assumed the applicant's quality assurance program had been deficient, whenever they opened an EOI file the IDVP looked for generic (plant-wide) implications, which necessarily included consideration of the cause of an error.[210] Similarly, the ITP identified, as causes for errors in its seismic design, the evolution of technology, criteria and requirements, difficulties with interfaces and communications, and several random factors.[211] For seismic design errors, the ITP then correlated each Class A and B error identified by the IDVP and each Open

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[204] *Id.*

[205] *Id.* at 3-5.

[206] Cooper *et al.* Tr. fol. D-1459 at 7-1 to -2; App. Exh. 90, IDVP Final Report, Vol. III, at 6.3-1, 6.3.2-1, 6.3.3-1, and 6.3.4-1.

[207] *Specifically, the IDVP found that:*

> (1) Safety-related systems for DCNPP-1 were seismically designed twice to meet two sets of design criteria, and with a substantial interval of time between the two designs. (2) In addition to two complete designs, the plant had substantial additional design work performed as a result of recent NRC IE [Inspection and Enforcement] bulletins and TMI [Three Mile Island] requirements. (3) This multiple design work has occupied 15 years of calendar time. (4) Seismic design technology had advanced from a rudimentary effort in 1967 to a reasonably mature, systematic and sophisticated process today. In the natural course of this evolution, methodology and criteria have changed significantly. (5) Nuclear plant design naturally requires the transfer of large amounts of design information from one design group to another. In the case of DCNPP-1, these design interfaces existed in especially large numbers both within PGandE and between PGandE and independent firms. (6) Design document control practices in use at the time of the original design were not consistent with the eventual duration and complexity of the design process.

App. Exh. 90, *IDVP Final Report, Vol. III, at* 6.3-2.

[208] Cooper Tr. D-1722.

[209] Reedy Tr. D-1642-43.

[210] Cooper *et al.* Tr. fol. D-1459 at 7-3 to -5; Hubbard Tr. D-2160; Jacobson Tr. D-987.

[211] App. Exh. 91, ITP Phase I Final Report, at 1.8.2-2 to -3, 1.8.3-1 to 1.8.5-1; Dick *et al.* Tr. fol. D-847 at 5.

Item identified by the ITP with its cause.[212] In the nonseismic design area, the ITP found only insignificant errors with random causes.[213]

The Governor also claims that, in addition to identifying the causes of the various design errors and their generic implications, the applicant's verification programs should have isolated the quality assurance failures that permitted each of the original design errors to occur. Additionally, he asserts that purported errors made by the ITP during its reviews that subsequently were found by the IDVP also were caused by quality assurance lapses that should have been specifically identified.[214] But the root causes of the original quality assurance program failures were identified in the "lookback" reviews and the applicant's quality assurance program was corrected to address these problems. Given this, and the search for generic causes carried out by the IDVP for each identified design error, no further specification of discrete quality assurance failures was necessary. Further, the asserted errors the Governor claims the ITP made as part of its verification effort were essentially differences between the ITP and the IDVP in modeling and they do not have quality assurance implications.[215]

We find, therefore, that the causes for the failures of the applicant's quality assurance program and the evaluation of those errors for generic concerns have been sufficiently addressed by the applicant's verification program to provide adequate confidence that no further significant design errors exist. We reach this result even though the verification efforts did not also identify as a root cause for the design deficiencies, PG&E management's lack of commitment to quality assurance as suggested by the Governor.[216] Similarly, the staff, while in general agreement with the findings of the IDVP and ITP on the causes underlying the design errors, concluded that these causes should be more fundamentally attributed to the failure of PG&E management to recognize, at the time of the Hosgri reevaluation, the significance of the revised seismic design requirements and the attendant need to implement a well-controlled design effort.[217] Whether it was lack of commitment or lack of

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[212] App. Exh. 91, ITP Phase I Final Report, at Appendices 1C and 1D.

Based on the testimony of his expert witness, the Governor generally asserts that the verification program did not correlate basic causes to specific identified errors. Gov. PF at 63. Interestingly, the Governor's witness did not review the applicant's nonconformance reports (NCRs) even though each NCR contained a statement of the cause of each reported error. Hubbard Tr. D-2164; *see, e.g.*, Gov. Exhs. 43 and 44.

[213] App. Exh. 92, ITP Phase II Final Report, at 3-2 to -3.

[214] Gov. PF at 64-65.

[215] Cooper *et al.* fol. Tr. D-1459 at 8.8; Reedy Tr. D-1640-41. *See* p. 607, *supra*.

[216] Gov. PF at 67-68.

[217] Knight and Schierling Tr. fol. D-2906 at 4. *See also* Staff Exh. 54, Diablo Canyon QA Case Study (final) (Sept. 19, 1983) and JI Exh. 128, Diablo Canyon QA Case Study (draft) (July 1983).

awareness, PG&E's management cannot escape responsibility for a quality assurance program that initially allowed for design errors of the type and number identified at Diablo Canyon by the verification program. The evidence indicates, however, that by the late 1970's significant improvements were being made in the applicant's quality assurance program.[218] Since that time, the applicant has instigated many more changes in its quality assurance program and carried out an extensive and unparalleled design verification program.[219] The painful lessons PG&E's management has learned from the huge expediture of resources required to verify the adequacy of the Diablo Canyon design have produced a present approach to quality assurance that is much improved and currently satisfactory.[220] As it must accept responsibility for past failings, PG&E management must also be credited for the significant improvements in its quality assurance program. For this reason, the failure of the applicant's verification program to include in its list of causative factors the past failings of PG&E management toward quality assurance is not fatal and does not alter our conclusion that the root causes have been sufficiently identified.

M. In issue 8, the Governor and the joint intervenors maintain that the ITP did not timely develop and implement an adequate quality assurance program for performing its verification functions and the necessary physical modifications to the Diablo Canyon facility, and that the IDVP failed to oversee the ITP's program.[221]

From the time of its initiation in November 1981 until August 1982, the ITP carried out its design verification work under the PG&E quality assurance program. An NRC inspection in November 1981 found the structure of this program acceptable although some implementation deficiencies existed in the program.[222] When the Bechtel-PG&E team was formed in 1982, the managements of the two companies decided to use the Bechtel quality assurance program for the project. The applicant's program was not chosen because of the controversy surrounding PG&E quality assurance triggered by the suspension of its low power license. Rather, because Bechtel was the manager of the completion project and its quality assurance program had been accepted at other facilities by the NRC, the Bechtel program (*i.e.*, the Bechtel Topical Report) was adopted and appropriately modified to reflect, *inter alia*, the organizational struc-

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[218] Dick *et al.* Tr. fol. D-847 at 3-5; de Uriarte Tr. 887-88; Staff Exh. 38, SECY-82-89, Encl. 1 at 1 n.1.
[219] Anderson *et al.* Tr. fol. D-224 at 16-21; Dick *et al.* Tr. fol. D-847 at 9-11.
[220] *See* pp. 613-17, *infra*.
[221] Gov. PF at 68-73; JI PF at 37-45.
[222] Dick *et al.* Tr. fol. D-847 at 9; de Uriarte Tr. D-895; Skidmore Tr. D-3170.

ture of the ITP.[223] Under the amended Bechtel program, however, the applicant's engineering procedures were used as implementing, or second tier, procedures.[224] The modified Bechtel quality assurance program was conditionally approved by the NRC on August 2, 1982, and placed in effect on August 20, 1982. Final NRC acceptance was granted on September 22, 1982. All design modifications performed by the ITP after the August 20 date were done under the modified Bechtel program.[225]

After the ITP became a joint Bechtel-PG&E project, the relationship between the ITP and the applicant was essentially that of an architect-engineer and applicant. Before the Bechtel-based quality assurance program was put into effect, the applicant reviewed it for compliance with Appendix B and the applicant's licensing commitments. Once the program was adopted, the applicant's quality assurance department performed continuous audits of the ITP's activities.[226] The applicant also audited the IDVP contractors to ensure that each one had implemented an adequate quality assurance program to control the verification activities. Similarly, the applicant audited the IDVP-ITP interface to verify that it was adequately controlled.[227]

In order to ensure that the modified Bechtel quality assurance program was properly implemented, the ITP initially trained and indoctrinated all personnel performing quality-related activities in the requirements of the program and, while the verification was ongoing, performed further training to remedy any program weaknesses identified by audits and other oversight activities. Throughout the program, numerous additional audits of the ITP verification program were performed by the applicant's quality assurance department, the ITP's own quality assurance personnel, Bechtel's San Francisco Power Division Quality Assurance, and the IDVP.[228] And, as a result of the various audit findings, the ITP took appropriate remedial and corrective actions.[229] Similarly, the staff reviewed the ITP's quality assurance program through a series of inspections while the verification activities were in progress.[230] The results of the audits and inspections demonstrate that the ITP's quality assurance

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[223] Dick *et al.* Tr. fol. D-847 at 10, 15-16; Dick Tr. D-1016-18.
[224] Dick *et al.* Tr. fol. D-847 at 13-14; de Uriarte Tr. D-1015.
[225] Dick *et al.* Tr. fol. D-847 at 10-11.
[226] *Id.*; Skidmore Tr. D-851-52.
[227] Dick *et al.* Tr. fol. D-847 at 11-12.
[228] *Id.* at 17-19; Cooper *et al.* Tr. fol. D-1459 at 8-1 to -5.
[229] Dick *et al.* Tr. fol. D-847 at 18.
[230] Morrill Tr. fol. D-2906 at 4-5.

program was effectively implemented.[231] No serious deficiencies were identified by the audits and the staff issued no notices of violation with respect to the ITP quality assurance program. In total, the various audits revealed less than 100 findings or conditions needing correction for a project involving 1200 technical people performing design work over an eighteen-month period.[232]

The Governor and the joint intervenors charge, however, that the design verification work performed under the PG&E quality assurance program (*i.e.*, the program in effect from November 1981 until August 1982) is suspect because the PG&E program was inadequate. Additionally, they assert that deficiencies identified by audits of the ITP quality assurance program (*i.e.*, the modified Bechtel program in effect after August 1982) show that that program also was insufficient.[233] A preponderance of the evidence does not support either position of the Governor and the joint intervenors.

The Governor and the joint intervenors rely upon two reports by PG&E consultants, Project Assistance Corporation (PAC)[234] and EDS Nuclear, Incorporated (EDS),[235] to support their claim that the applicant's quality assurance program in effect until August 1982 was inadequate. Both reports are generally critical of the relationship and coordination between the basic corporate quality assurance manual and the various subordinate departmental manuals and other quality assurance documents. The two reports do not represent, however, the results of audits or evaluations of the pre-August 1982 PG&E quality assurance program.[236] Rather, each of the reports deals with a very limited review of the applicant's corporate quality assurance manual or the individual department quality control manuals.

The Commission's regulations do not require that all pertinent quality assurance or quality control documents be consolidated and integrated into a single manual or set of manuals. Under the applicant's quality assurance program none of the quality assurance and quality control manuals is self-sufficient (*i.e.*, each must be read in conjunction with other documents). Because the PG&E quality assurance program is comprised of many documents and a large number of procedures, the applicant retained PAC to review the company quality assurance manual and

[231] *Id.* at 5-6; Dick *et al.* Tr. fol. D-847 at 20; Cooper *et al.* Tr. fol. D-1459 at 8-3 to -5.
[232] Dick *et al.* Tr. fol. D-847 at 20-21.
[233] Gov. PF at 68-73; JI PF at 37-45.
[234] Gov. Exh. 35.
[235] Gov. Exh. 36.
[236] Stokes Tr. D-3147; Gouveia Tr. D-3149; de Uriarte Tr. D-3148-49, D-3173-74.

outline the scope of the work necessary to make the manual self-sufficient.[237] PAC examined the corporate quality assurance manual, which consists of 40 procedures out of some 2400 procedures that make up the entire program,[238] and made a number of findings critical of the applicant's organization.[239] But the items identified by PAC as missing from the corporate manual can be found in other documents in the program.[240] Indeed, the PAC report indicated that, within the company, complete — yet separate — quality assurance programs were being implemented by various organizations.[241] Similarly, EDS was retained by the applicant to review the individual department quality control manuals to determine the work necessary to make each manual self-sufficient and properly coordinated with the other manuals.[242] Once again, EDS was critical of the applicant's organization,[243] but the applicant's review of the EDS findings identified no violations of 10 C.F.R. Part 50, Appendix B.[244] Accordingly, these two limited reviews do not establish, as the Governor and the joint intervenors would have it, that the applicant's quality assurance program in effect from the beginning of the verification program until August 1982 was inadequate. Hence, the verification design work performed under the PG&E program is not inferentially suspect.[245] Moreover, 95 to 100 percent of the design work that resulted in modifications to the Diablo Canyon facility was performed after August 1982.[246]

[237] de Uriarte Tr. D-3148-49.
[238] Gouveia Tr. D-3149-51.
[239] Gov. Exh. 35 at 4-6.
[240] Gouveia Tr. D-3151-52; de Uriarte Tr. D-3152.
[241] Gov. Exh. 35 at 5.
[242] de Uriarte Tr. D-3149; Stokes Tr. D-3154.
[243] Gov. Exh. 36, Attachment at 1-2.
[244] de Uriarte Tr. D-3156.
[245] The Governor and the joint intervenors assert, based on the testimony of a staff witness, that the PG&E quality assurance program in effect from November 1981 until August 1982 was also deficiently implemented. Gov. PF at 69; JI PF at 37. Although there were deficiencies in the implementation of the PG&E program, staff witness Morrill pointed out that as a result of staff inspections conducted prior to the verification program the deficiencies were known by the applicant and that corrective actions were taken and largely completed by mid-1982. Because of this, the staff witness did not consider the program implementation from November 1981 to August 1982 inadequate; rather he found it deficient only in certain particulars. Morrill Tr. D-3025-26.
[246] Moore Tr. D-3157-60.

The ITP's decisions to redo certain designs without reliance on any previous review work, were made over a period spanning the date when the modified Bechtel quality assurance program was adopted. The individual decisions were made at the following times: fuel handling building — May 1982; auxiliary building — June 1982; intake structure — June 1982; piping — July 1982; raceways and heating, ventilating and air conditioning (HVAC) supports — July 1982; containment annulus — January 1983. Additionally, the decisions on all Phase II reviews (including HVAC technical reviews, electrical reviews and mechanical reviews) were made in August 1982. *Id.*

Nor do the conditions identified by the various audits of the ITP quality assurance program (*i.e.*, the modified Bechtel program in effect after August 1982) demonstrate that the program was inadequate as charged by the Governor and joint intervenors. Among others, the Governor and joint intervenors point to the twenty-four conditions identified by the IDVP in its initial audit of the ITP quality assurance program as establishing the inadequacy of the program. On the basis of that audit, and the subsequent follow-up audit of the previously identified conditions, the IDVP concluded that the ITP quality assurance program was being effectively implemented and none of the identified conditions would have an impact on the control of design for the ITP work.[247] Our review of the conditions noted by the IDVP, as well as the other audit findings relied upon by the Governor and the joint intervenors, convinces us that none of the conditions, singularly or in combination, shows that the ITP quality assurance program was inadequate. Typical of these conditions was ITP management's lack of action toward nine engineers who missed three scheduled training sessions. This condition was corrected after the initial audit and the IDVP's follow-up audit verified that the condition had been remedied. This minor deficiency and other similar ones simply do not demonstrate the program was unacceptable. Considering the extent of the ITP verification activities, such discrepancies are to be expected and the very purpose of the auditing process is to ensure that they are caught and corrected. Thus, contrary to the charges of the Governor and the joint intervenors, the ITP quality assurance program, under which the vast majority of the design verification program was performed, was adequate.

N. In issue 9, the joint intervenors[248] maintain that the applicant has failed to provide assurance of component cooling water system (CCWS) heat removal capacity and that a technical specification limiting plant operation does not provide a level of safety equivalent to compliance with GDC 44.[249]

During the course of a review of the applicant's analysis of the CCWS, the NRC staff discovered that the most limiting single failure from the standpoint of CCWS performance, concurrent with the worst design basis accident heat load, had not been considered by the applicant as required by GDC 44.[250] Rather, the assumptions incorporated in the

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[247] Cooper *et al.* Tr. fol. D-1459 at 8-1 to -8; App. Exh. 90, IDVP Final Report, Vol. III, at 5.6.4; App. Exh. 133, ITR 41, at 1-2, 11.
[248] JI PF at 45-46.
[249] 10 C.F.R. Part 50, Appendix A, GDC 44.
[250] Wermiel Tr. fol. D-2864 at 1-2 (Contention 9); Staff Exh. 55, SSER 16, at 9-5 to -7.

applicant's original analysis (including the use of a single heat exchanger) led the applicant to conclude that adequate cooling for the CCWS would be available as long as the water temperature of the ocean, the ultimate heat sink for the Diablo Canyon reactors, did not go above 70°F. With the more stringent conditions assumed by the staff, however, the maximum temperature of the ocean under which the CCWS could meet the limiting conditions would be 64°F.[251]

To overcome this problem, the applicant proposed a technical specification requiring monitoring of the ocean water temperature. When the temperature approaches the maximum allowable limit of 64°F, the normally isolated second CCWS heat exchanger will be put on line to provide the additional heat removal capability needed to maintain an acceptable CCWS temperature in the event of the design basis loss of coolant accident.[252] In the event that the second heat exchanger, a passive unit with low failure probability, is unavailable or fails, the technical specification requires that the plant be shut down.[253]

We find the applicant's proposed technical specification is sufficient to meet the requirements of GDC 44.[254] Because the applicant's CCWS technical specification was presented to us as "proposed," the Director of Nuclear Reactor Regulation must ensure that the essential operating conditions set forth in the applicant's proposal are incorporated into the plant technical specifications before permitting operation.[255]

## III. CONCLUSION

For the reasons we have discussed in Parts I and II, we find that the scope and the execution of the applicant's verification programs have

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[251] Wermiel Tr. fol. D-2864 at 1-4 (Contention 9).

[252] Connell *et al.* Tr. fol. D-487 at 37.

[253] Connell Tr. D-546, D-551.

[254] Connell *et al.* Tr. fol. D-487 at 37; Wermiel Tr. fol. D-2864 at 3-4 (Contention 9).

[255] The joint intervenors assert that the applicant's proposed technical specification is insufficient. They argue that if the recently experienced, above-normal ocean temperatures continue for long periods then, under the technical specification, the plant will have to shut down more frequently than originally contemplated. The joint intervenors then claim that each such unnecessary shutdown unacceptably challenges plant systems, thereby eroding the original safety margins of the facility. Thus, they argue the proposed limitation does not provide a level of safety equivalent to compliance with the requirements of GDC 44. *See* JI PF at 45-46. The sequence of events that must occur before shutdown is necessary is an unlikely one. The ocean temperatures must reach above-normal levels and the second heat exchanger (a passive component) must be unavailable for a period of at least six hours. Connell Tr. D-551. In these circumstances, the likelihood of any significant increase in the number of plant shutdowns because of ocean temperatures is exceedingly remote and the effect on the number of thermal cycles is inconsequential. Finally, we note that the applicant's technical specification could be amended and an additional heat exchanger added to the CCWS sometime in the future if the recent transient rise in ocean temperatures should become permanent or the facility should experience unexpected and repeated failures in the existing heat exchangers. Connell Tr. D-546.

been sufficient to establish that Diablo Canyon Unit 1 design adequately meets its licensing criteria. The applicant's verification efforts provide adequate confidence that the Unit 1 safety-related structures, systems and components are designed to perform satisfactorily in service and that any significant design deficiencies in that facility resulting from defects in the applicant's design quality assurance program have been remedied. Accordingly, we conclude that there is reasonable assurance that the facility can be operated without endangering the health and safety of the public. As a result, the license authorization previously granted to the Director of Nuclear Reactor Regulation in the Licensing Board's August 31, 1982 initial decision, LBP-82-70, *supra,* 16 NRC at 854, remains in effect with respect to Unit 1. Before exercising that authority, the Director must ensure that the applicant has adopted an appropriate technical specification for the component cooling water system.[256] In addition, before allowing commercial operation, the Director must ensure that the applicant has performed the appropriate jet impingement analyses for certain lines inside the containment.[257] Until we make our findings with respect to Unit 2, the license authorization previously granted for that unit is not effective.[258]

Our findings have been made on the basis of the record evidence in the reopened operating license proceeding. We note, however, that recent events may affect our findings. On February 14, 1984, the joint intervenors filed a second motion to reopen the record[259] citing, *inter alia,* a number of recently discovered, purported design deficiencies that they assert undermine the validity and integrity of the applicant's verification efforts and directly bear upon the issues in the proceeding. In support of their motion, the joint intervenors proffer the affidavits of several engineers who formerly worked at the Diablo Canyon site. The applicant and the staff oppose the joint intervenors' motion and have filed numerous affidavits of asserted experts rebutting joint intervenors' claims.[260] Although we have initially reviewed the motions and the responses, our assessment of the parties' filings has not been completed. In addition, Supplement 21 of the staff's Safety Evaluation Report for Diablo Canyon indicates that the staff is currently investigating a large number of recent allegations concerning the Diablo Canyon facility including

[256] *See* p. 618, *supra.*
[257] *See* p. 603, *supra.*
[258] *See* p. 582, *supra.*
[259] Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen the Record.
[260] *See* Pacific Gas and Electric Company's Answer in Opposition to Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen (Mar. 5, 1984); NRC Staff's Answer to Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen the Record (Mar. 15, 1984).

several that appear to relate to the adequacy of facility design.[261] In this regard, the staff informed us by a letter dated February 7, 1984, and again in its opposition to the joint intervenors' motion, that it is currently investigating matters relating to small bore piping at the facility that directly bear upon the issues in this proceeding. Therefore, some of these matters may require that we again reopen the record in the proceeding and hear further evidence.[262] Hence, it is possible that these findings may have to be amended or withdrawn in their entirety depending upon the nature of the new evidence.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

**Concurring opinion of Mr. Moore:**

I write separately on an additional point in order to call it to the Commission's attention. In the reopened proceeding, the joint intervenors and the Governor sought to litigate several issues involving the adequacy of the applicant's verification efforts in light of the asserted failure of the applicant's quality assurance program to comply with 10 C.F.R. Part 50, Appendix A, GDC 1. Specifically, the joint intervenors and the Governor claimed, based on the applicant's FSAR, that the applicant had no quality assurance program to assure the design of structures, systems and components that were "important to safety" within the meaning of Appendix A. Rather, they asserted the applicant only had a quality assurance program to assure the design of structures, systems and components that were "safety-related" within the meaning of 10 C.F.R. Part 50, Appendix B.

At the prehearing conference, we excluded these issues from the reopened proceeding. We ruled that the history of the Diablo Canyon

[261] NUREG-0675, Supplement No. 21, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2" (Dec. 1983).

[262] Because the joint intervenors' appeal from the Licensing Board's initial decision, LBP-82-70, *supra*, 16 NRC 756, is still pending before us and, in addition, the joint intervenors' latest motion was filed while the reopened phase of the proceeding was before us, we necessarily retain jurisdiction over the proceeding.

operating license application showed that the two terms, "important to safety" and "safety-related," had been used synonymously by the applicant and the staff, and to the extent the quality assurance criteria are currently interpreted to distinguish between the terms, such distinction would not be retroactively applied to Diablo Canyon.*

I highlight this matter because on January 18, 1984 the staff issued Board Notification 84-011 regarding the meaning of the terms "safety-related" and "important to safety." That notification contains a January 5, 1984 letter from the Director, Division of Licensing, to all operating licensees and applicants. The letter states that applicants are responsible for developing and implementing quality assurance programs that meet the requirement of Appendix A, GDC 1, for plant equipment "important to safety" as well as a program for "safety-related" equipment in accordance with Appendix B. The letter then suggests this interpretation of the regulations is not new but one that the staff has always followed. If the Director's position on this matter is now that of the Commission (including the asserted long-standing nature of the interpretation), then it would appear that the Governor and the joint intervenors must be given an opportunity to litigate the issues regarding the applicant's compliance with Appendix A.

## APPENDIX A

### Issues at Hearing in Accordance with Orders of August 26 and October 7, 1983 (unpublished)

1. The scope of the IDVP review of both the seismic and nonseismic aspects of the designs of safety-related systems, structures and components (SS&Cs) was too narrow in the following respects:
   (a) The IDVP did not verify samples from each design activity (seismic and nonseismic).
   (b) In the design activities the IDVP did review, it did not verify samples from each of the design groups in the design chain performing the design activity.
   (c) The IDVP did not have statistically valid samples from which to draw conclusions.

*Transcript of August 23, 1983 prehearing conference at D-67-68.

(d) The IDVP failed to verify independently the analyses but merely checked data of inputs to models used by PG&E.

(e) The IDVP failed to verify the design of Unit 2.

2. The scope of the ITP review of both the seismic and nonseismic aspects of the designs of the safety-related systems, structures and components (SS&Cs) was too narrow in the following respects:

(a) The ITP did not verify samples from each design activity (seismic and nonseismic).

(b) In the design activities the ITP did review, it did not verify samples from each of the design groups in the design chain performing the design activity.

(c) The ITP did not have statistically valid samples from which to draw conclusions.

(d) The ITP has failed systematically to verify the adequacy of the design of Unit 2.

3. In various situations listed below the ITP used improper engineering standards to determine whether design activities met license criteria. In some of these situations, the IDVP either used or approved the use of such improper standards or did not verify them at all.

(f) The ITP's modeling of the soil properties for the containment and auxiliary buildings was improper in that:

(i) in the soil structure interaction analysis of containment for the DE [Design Earthquake] and the DDE [Double Design Earthquake], use of boundary motion inputs to the model were improperly used;

(ii) the soil structure interaction analysis for containment for the DE and the DDE uses a seven percent damping value for rock, which is unconservative, especially for the DE;

(iii) the dynamic analyses of the containment for all earthquakes omit any analysis of uplifting of the foundation mat;

(iv) the modeling of the soil springs for the auxiliary building does not specify soil properties;

(v) in the modeling of the soil springs for the auxiliary building, the motion inputs to the lower ends of the springs does not account for all soil structure interaction phenomena that could be expected.

(o) The ITP has not demonstrated, and the IDVP has not verified, that the DCP modeling of the seismic response of the fuel handling building is proper, in that the DCP has not adequately justified the use of the translational and torsional response of the auxiliary building as input to the fuel handling building nor has it demonstrated the validity of the dynamic degrees of freedom selected.

(p) The ITP has not demonstrated, and the IDVP has not verified, that the DCP seismic model of the slabs in the auxiliary building is proper, in relation to the use of vertical and rotational springs to model the columns, and the motions used as input at the ends of the springs not connected to the slabs. In addition, in the study of the diaphragms, the ITP has not adequately accounted for the inplane flexibility of these slabs, and has not adequately demonstrated that stresses are within allowable limits at all elevations.

(q) The ITP has not demonstrated and the IDVP has not verified, that the soils analysis for the buried diesel fuel oil tanks is proper in that the values of the exponent shown in figure 14 of ITR 68 have not been demonstrated to be appropriate and the variation of shear velocity with depth is not properly justified.

(r) The ITP has not demonstrated and the IDVP has not verified that the soils analysis for the auxiliary saltwater piping and circulating water intake conduits is proper in that the selection of the modulus versus strain curve utilized is not justified.

(s) The ITP has not demonstrated and the IDVP has not verified that the seismic analysis of the turbine building is proper in that bolt bearing capacities were taken from an inappropriate source.

(t) The ITP has not demonstrated and the IDVP has not verified that the seismic analysis of the turbine building is proper in that the use of four different models for the vertical analysis has not been justified.

4. The IDVP accepted deviations from the licensing criteria without providing adequate engineering justification in the following respects:

(a) Contrary to the requirements of FSAR Section 17.1 regarding compliance of the as-built installation with the design documents, the IDVP review of the AFWS disclosed that the as-built installation failed to meet the design drawings in that (i) a steam trap on the turbine-driven AFW pump steam supply line is not provided and (ii) there are discrepancies in the arrangement of the long-term cooling water supply line.

(b) Contrary to FSAR Section 8.3.3, the electrical design does not fully comply with the commitments regarding separation and color coding.

(h) Contrary to PG&E's September 14 and December 28, 1978 licensing commitments, CRVPS equipment identified in the

FSAR as necessary to maintain control room habitability during safe shutdown has not been evaluated regarding the effects of a moderate energy pipe break.

(i) The fire protection for the motor driven AFW pump room is not consistent with the PG&E licensing commitment for fire zone separation as stated in its November 13, 1978 Supplemental Information for Fire Protection Review ("SIFPR") in that:
  (1) there is a large grated ventilation opening in the ceiling of the room;
  (2) a fire damper has gaps when it is closed.

(j) The fire protection for the AFW pump room is not consistent with the PG&E licensing commitment for cable separation as stated in its SIFPR of November 13, 1978 in that:
  (1) the pumps for the motor driven AFW pumps and the control circuitry for a flow control valve necessary for operation of the turbine driven AFW pump are located in a single fire zone;
  (2) cables for some AFW circuits are not routed in accord with descriptions in the SIFPR and four AFW circuits PG&E committed to identify and review in the SIFPR were not included in that document.

(k) Contrary to the licensing commitment set forth in its SIFPR of November 13, 1978, each of the three 4160 volt cable spreading rooms has a ventilation opening leading up to the 4160 volt switchgear rooms.

(l) Contrary to FSAR Section 3.6, possible jet impingement loads have not been considered in the design and qualification of safety-related piping and equipment inside containment.

(q) Contrary to PG&E's December 28, 1979 licensing commitment letter to the NRC, modifications to protect two Auxiliary Feedwater valves from the effects of moderate energy line breaks were not implemented.

(r) Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR Section 3.6A (NSC evaluation of pipe break outside containment), four components were identified for which high energy line cracks could cause temperatures in excess of the specification temperatures of the components.

(s) Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR, Section 3.6A (NSC evaluation of pipe break outside containment), a conduit was identified whose failure due to a high energy line crack could eliminate redundant Auxiliary Feedwater system flow.

(t) Contrary to the FSAR Section 8.3 commitment to provide switchgear buses with adequate short circuit interrupting capability, the calculated duties for circuit breakers on 4160 V buses F, G, and H were above the nameplate ratings for those buses.

(u) Contrary to single failure criteria stated in FSAR Section 3.1.1, reviews of the Auxiliary Feedwater and Control Room Ventilation and Pressurization systems identified circuit separation and single failure deficiencies. Similar deficiencies were identified in additional verification reviews, which included other safety-related systems.

5. The verification program has not verified that Diablo Canyon Units 1 and 2 "as built" conform to the design drawings and analyses.

6. The verification program failed to verify that the design of safety-related equipment supplied to PG&E by Westinghouse met licensing criteria.

7. The verification program failed to identify the root causes for the failures in the PG&E design quality assurance program and failed to determine if such failures raise generic concerns.

8. The ITP failed to develop and implement in a timely manner a design quality assurance program in accordance with 10 C.F.R. Part 50, Appendix B to assure the quality of the recent design modifications to the Diablo Canyon facility and the IDVP failed to ensure that the corrective and preventative action programs implemented by the ITP are sufficient to assure that the Diablo Canyon facilities will meet licensing criteria.

9. Contrary to General Design Criterion 44 (GDC 44) of Appendix A to 10 C.F.R. Part 50, PG&E has failed to provide adequate assurance of component cooling water system (CCWS) heat removal safety function capacity in that the maximum ocean water temperature of 64°F. is not conservative because it has already been exceeded in 1983. Furthermore a technical specification limitation which permits plant operation at reduced power levels in lieu of enlarging the capacity of the CCWS does not provide an equivalent level of safety as compliance with the requirements of GDC 44 (SSER, 16 (Aug. 1983) and September 1983 ocean water temperature readings).

## APPENDIX B — LIST OF WITNESSES

### Applicant's Witnesses

Anderson, Richard C.
- Education: B.S. Mechanical Engineering, University of California at Berkeley
- Present Occupation: An Engineering Manager for Bechtel Power Corporation now assigned as the Engineering Manager for the Diablo Canyon Project

Connell, Edward C., III
- Education: M.S. Nuclear Engineering, 1974, Purdue University
- Present Occupation: Mechanical Group Supervisor (Bechtel) Diablo Canyon Project

Cranston, Gregory V.
- Education: B.S. Nuclear Science, United States Naval Academy, Annapolis, MD
- Present Occupation: Project Engineer (Bechtel) for Unit 2 of the Diablo Canyon Project

Dick, Charles W.
- Education: M.S. Electrical Engineering, 1948, Stanford University
- Present Occupation: Project Manager (Bechtel) and member of project management team of the Diablo Canyon Project

Gouveia, Leigh A.
- Education: B.S. Mechanical Engineering, 1968, California State Polytechnic College, San Luis Obispo
- Present Occupation: Project Engineer for Project Assistance Corporation

Hoch, John B.
- Education: B.S. Mechanical Engineering, 1959, University of Idaho
- Present Occupation: PG&E Project Manager of Diablo Canyon Project

Jacobson, Michael J.
- Education: B.S. Civil Engineering, 1970, Sacramento State College

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| Present Occupation: | Project Quality Assurance Engineer (Bechtel) for Diablo Canyon Project |
| Kaplan, Stanley | |
| Education: | Ph.D Mechanical Engineering and Applied Mathematics, 1960<br>University of Pittsburgh |
| Present Occupation: | President, Kaplan & Associates, Inc. — a consulting firm specializing in risk analysis and applied decision theory |
| Kreh, Edward J., Jr. | |
| Education: | B.S. Mechanical Engineering<br>Carnegie Institute of Technology (now Carnegie Mellon University) of Pittsburgh, PA |
| Present Occupation: | Senior Consulting Engineer with SMC O'Connell and Associates of Pittsburgh, PA |
| Moore, Gary H. | |
| Education: | M.S. Mechanical Engineering, 1969<br>San Jose State University |
| Present Occupation: | PG&E Unit 1 Project Engineer of the Diablo Canyon Project |
| Seed, H. Bolton | |
| Education: | Ph.D Civil Engineering, 1948<br>Kings College, London University |
| Present Occupation: | Professor, University of California at Berkeley |
| Shipley, Larry E. | |
| Education: | B.S. Mechanical Engineering<br>United States Merchant Marine Academy, Kings Point, NY |
| Present Occupation: | Assistant Chief Engineer (Plant Design) in Bechtel's San Francisco Power Division and Technical Consultant to Diablo Canyon Project |
| Skidmore, Steven M. | |
| Education: | M.S. Nuclear Engineering, 1969<br>Stanford University |
| Present Occupation: | PG&E Manager of Quality Assurance in the Nuclear Power Generation Department |

Stokes, William J.
Education: B.S. Mechanical Engineering, 1974
Drexel University
Present Occupation: Partner, Shalako Energy Services (formerly with EDS Nuclear)

de Uriarte, Thomas G.
Education: B.S. Civil Engineering, 1967
University of California at Berkeley
Present Occupation: Senior Engineer, Quality Assurance Department, Pacific Gas and Electric Company

Vahlstrom, Wallace
Education: Electrical Engineer (degree not specified)
Present Occupation: Senior Electrical Engineer at Pacific Gas and Electric Company

White, William H.
Education: Ph.D Civil Engineering
University of Colorado
Present Occupation: Engineering Specialist with Bechtel's San Francisco Power Division — Seismic Analysis and Assistant Project Engineer in the Diablo Canyon Project

Wiesemann, Robert A.
Education: B.S. Mechanical Engineering, 1949
Case Institute of Technology
Present Occupation: Manager of Regulatory and Legislative Affairs in the Nuclear Technology Division of the Westinghouse Electric Corporation

**IDVP Witnesses**

Biggs, John M.
Education: M.S. Civil Engineering, 1947
Massachusetts Institute of Technology
Present Occupation: Professor Emeritus of Civil Engineering, Massachusetts Institute of Technology and Partner in the Consulting Firm of Hansen, Holley and Biggs, Inc.

Cloud, Robert L.
Education: Ph.D Mechanical Engineering, 1964
University of Pittsburgh

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| Present Occupation: | President, Robert L. Cloud Associates, Inc., Berkeley, CA |
| Cooper, William E. | |
| Education: | Ph.D Engineering Mechanics, 1951<br>Purdue University |
| Present Occupation: | Senior Vice President and Technical Director of Teledyne Engineering Services until 1976, now Consulting Engineer for Teledyne |
| Holley, Myle J., Jr. | |
| Education: | M.S. Civil Engineering, 1947<br>Massachusetts Institute of Technology |
| Present Occupation: | Professor Emeritus, Massachusetts Institute of Technology and Partner in the Consulting Firm of Hansen, Holley and Biggs, Inc. |
| Krechting, John E. | |
| Education: | B.S. Naval Science, 1965<br>United States Naval Academy |
| Present Occupation: | Employed by Stone & Webster Engineering Corporation — assigned as Project Engineer for the IDVP |
| Reedy, Roger F. | |
| Education: | B.S. Civil Engineering, 1956<br>Illinois Institute of Technology |
| Present Occupation: | President, R.F. Reedy, Inc., Consulting Engineers, Los Gatos, CA |
| Wray, Ronald | |
| Education: | M.S. Engineering Science, 1962<br>Rensselaer Polytechnic Institute |
| Present Occupation: | Theoretical Stress Analyst, Teledyne Engineering Services |

**Governor's Witnesses**

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|---|---|
| Apostolakis, George | |
| Education: | Ph.D Engineering Science and Applied Mathematics<br>California Institute of Technology |
| Present Occupation: | Professor, Engineering and Applied Science, University of California at Los Angeles |

Hubbard, Richard B.
Education: B.S. Electrical Engineering, 1960
University of Arizona
Present Occupation: Vice President — MHB Technical Associates, San Jose, CA

Roesset, Jose M.
Education: D.S. Structures and Soil Mechanics, 1964
Massachusetts Institute of Technology
Present Occupation: Professor, University of Texas, Austin, TX

**Joint Intervenors' Witness**

Samaniego, Francisco J.
Education: Ph.D Mathematics-Statistics, 1971
University of California at Los Angeles
Present Occupation: Professor, Division of Statistics, University of California at Davis

**Staff's Witnesses**

Altman, Willard D.
Education: Ph.D Mathematics, 1975
University of Virginia
Present Occupation: Section Chief, Quality Assurance Branch, Division of Quality Assurance, Safeguards and Inspection Programs, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission

Costantino, Carl J.
Education: Ph.D Civil Engineering, 1966
Illinois Institute of Technology
Present Occupation: Professor, Civil Engineering, City College of the City University of New York

Haass, Walter P.
Education: B.S. Mechanical Engineering, 1952
Stevens Institute of Technology
Present Occupation: Assistant to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission

Knight, James P.
Education: B.S. Mechanical Engineering, 1957
Northeastern University

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| Present Occupation: | Assistant Director for Components and Structures Engineering, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission |
| Knox, John L. | |
| Education: | B.S. Electronic Systems Engineering, 1971<br>University of Maryland |
| Present Occupation: | Senior Reactor Systems Engineer (Electrical), Power Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission |
| Kubicki, Dennis J. | |
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Christine N. Kohl,** Chairman
**Dr. John H. Buck**
**Thomas S. Moore**

**In the Matter of** | **Docket Nos. 50-329-OM&OL 50-330-OM&OL**

**CONSUMERS POWER COMPANY**
**(Midland Plant, Units 1 and 2)** | **March 30, 1984**

The Appeal Board affirms the Licensing Board's refusal to quash subpoenas aimed at employees of a nonparty to this operating license proceeding.

## RULES OF PRACTICE: INTERLOCUTORY APPEALS (NONPARTY)

A nonparty to an operating license proceeding may appeal immediately an otherwise interlocutory discovery order. *Pacific Gas and Electric Co.* (Stanislaus Nuclear Project, Unit 1), ALAB-550. 9 NRC 683, 686 n.1 (1979).

## RULES OF PRACTICE: SUBPOENAS

A board may issue a subpoena upon a showing of only "general relevance" and "shall not attempt to determine the admissibility of evidence." *See* 10 C.F.R. § 2.720; *see also* 10 C.F.R. § 2.740(b)(1).

## PRIVILEGES: FIRST AMENDMENT (THE PRESS)

That the press enjoys a qualified privilege not to reveal its sources in certain circumstances is beyond doubt. *Branzburg v. Hayes,* 408 U.S. 665, 709-10 (1972) (Powell, J., concurring); *United States v. Cuthbertson,* 630 F.2d 139, 147 (3d Cir. 1980), *cert. denied,* 449 U.S. 1126 (1981); *Silkwood v. Kerr-McGee Corp.,* 563 F.2d 433, 436-37 (10th Cir. 1977); *Carey v. Hume,* 492 F.2d 631, 636 (D.C. Cir.), *cert. dismissed,* 417 U.S. 938 (1974); *Baker v. F&F Investment,* 470 F.2d 778, 783 (2d Cir. 1972), *cert. denied,* 411 U.S. 966 (1973).

## PRIVILEGES: GENERALLY

Courts traditionally have been loath to create a new testimonial privilege or to extend an existing one, "since such privileges obstruct the search for truth." *Branzburg v. Hayes, supra,* 408 U.S. at 690 n.29. *See Herbert v. Lando,* 441 U.S. 153, 175 (1979).

## EVIDENCE: DUTY TO PROVIDE

All citizens have a "general duty . . . to provide evidence when necessary to further the system of justice." *Wright v. Jeep Corp.,* 547 F. Supp. 871, 875 (E.D. Mich. 1982). *See Branzburg v. Hayes, supra,* 408 U.S. at 688.

## PRIVILEGES: FIRST AMENDMENT (THE PRESS)

The qualified First Amendment privilege of the press has been consistently and strictly limited to those reasonably characterized as part of the media. *Compare, e.g.,* the following cases where the privilege has been recognized: *United States v. Cuthbertson, supra; Silkwood v. Kerr-McGee Corp., supra; Baker v. F&F Investment, supra; Solargen Electric Motor Car Corp. v. American Motor Corp.,* 506 F. Supp. 546 (N.D.N.Y. 1981); *In re Consumers Union of the United States, Inc. (Starks v. Chrysler Corp.),* 32 Fed. R. Serv. 2d 1373 (S.D.N.Y. 1981); *Apicella v. McNeil Laboratories, Inc.,* 66 F.R.D. 78 (E.D.N.Y. 1975); with *Wright v. Patrolmen's Benevolent Ass'n,* 72 F.R.D. 161 (S.D.N.Y. 1976).

## PRIVILEGES: FIRST AMENDMENT (SCHOLAR'S)

The "scholar's privilege" — an alleged outgrowth of the journalist's First Amendment privilege — is of doubtful validity under modern case law, at least as applied to non-scholars. *See Wright v. Jeep Corp., supra,* 547 F. Supp. at 875-76. *See also In re Dinnan,* 661 F.2d 426, 427-31 (5th Cir. 1981), *cert. denied,* 457 U.S. 1106 (1982).

PRIVILEGES: FIRST AMENDMENT (THE PRESS)

Where the courts have recognized a journalist's privilege, they have balanced "the potential harm to the free flow of information that might result against the asserted need for the requested information." *Bruno & Stillman, Inc. v. Globe Newspaper Co.,* 633 F.2d 583, 596 (1st Cir. 1980) (footnote omitted). *See Branzburg v. Hayes, supra,* 408 U.S. at 710; *United States v. Cuthbertson, supra,* 630 F.2d at 148; *Carey v. Hume, supra,* 492 F.2d at 636-39; *Solargen Electric Motor Car Corp. v. American Motor Corp., supra,* 506 F. Supp. at 550.

PRIVILEGES: FIRST AMENDMENT (THE PRESS)

The principal factors to consider in determining to give recognition to the journalist's privilege are whether the requested information is relevant and goes to the heart of the matter at hand, and whether the party seeking the information has tried to obtain it from other possible sources. *Silkwood v. Kerr-McGee Corp., supra,* 563 F.2d at 438; *Baker v. F&F Investment, supra,* 470 F.2d at 783.

RULES OF PRACTICE: PROTECTIVE ORDERS

Boards assume protective orders will be obeyed unless a concrete showing to the contrary is made. One who violates a protective order risks serious sanction. *See Commonwealth Edison Co.* (Byron Nuclear Power Station, Units 1 and 2), ALAB-735, 18 NRC 19, 25 (1983).

RULES OF PRACTICE: PROTECTIVE ORDERS

Imposition of a protective order can be a pragmatic accommodation of the need for discovery and the protection of the asserted interests of the persons against whom discovery is directed.

## APPEARANCES

**John W. Karr,** Washington, D.C., for the appellants, Government Accountability Project employees Louis Clark, Thomas Devine, Billie Pirner Garde, and Lucy Hallberg.

**David M. Stahl, Susan D. Proctor,** and **Sarah H. Steindel,** Chicago, Illinois, for the applicant, Consumers Power Company.

**Donald F. Hassell** and **Nathene A. Wright** for the Nuclear Regulatory Commission staff.

# MEMORANDUM AND ORDER

Four employees of the Government Accountability Project (GAP) have appealed the Licensing Board's refusal to quash subpoenas directed to them in this operating license proceeding. *See* LBP-83-53, 18 NRC 282, *reconsideration denied,* LBP-83-64, 18 NRC 766 (1983).[1] For the reasons stated below, we affirm the Licensing Board's decision.

## I.

In July 1982, applicant Consumers Power Company (CPC) requested the Licensing Board to issue the four subpoenas here challenged. According to applicant, GAP submitted affidavits to the NRC alleging poor quality work at the Midland facility and gave similar information to the press. Applicant further asserted the relevance of this information to its pending licensing proceeding before the Licensing Board. CPC Application for Deposition Subpoenas (July 8, 1982) at 1, 2. The Board agreed as to the general relevance of GAP's allegations to certain matters already at issue in the proceeding and accordingly issued the subpoenas. Licensing Board Memorandum of July 8, 1982 (unpublished).[2]

Applicant, however, acceded to the NRC staff's request to defer serving the subpoenas while the staff conducted its own investigation of GAP's allegations. In March 1983, applicant informed the staff of its intention to proceed with discovery, and the staff did not object. Letter from M.I. Miller to J.G. Keppler (March 22, 1983). Applicant subsequently served its subpoenas, and GAP moved to quash on essentially three grounds: (1) First Amendment privilege, deriving from GAP's

[1] For ease of reference, we will refer to the four appellants collectively as GAP. GAP is not a party to the operating license proceeding and thus may appeal now an otherwise interlocutory discovery order. *Pacific Gas and Electric Co.* (Stanislaus Nuclear Project, Unit 1), ALAB-550, 9 NRC 683, 686 n.1 (1979).

[2] *See* 10 C.F.R. § 2.720, which provides that a board may issue a subpoena upon a showing of only "general relevance" and "shall not attempt to determine the admissibility of evidence." *See also* 10 C.F.R. § 2.740(b)(1).

information gathering and disseminating functions; (2) common law privilege; and (3) estoppel, resulting from the NRC's asserted promise of confidentiality. GAP Motion to Quash Subpoenas (June 27, 1983). As GAP explained, it "offers assistance to public and private employees, private citizens and community-oriented groups who pursue illegal, wasteful, improper or negligent actions by government or corporate bodies." *Id.* at 1. Citizens groups in Midland, Michigan, thus approached GAP, seeking assistance for "whistleblowers" at the Midland nuclear plant. GAP agreed to help and, serving as a conduit, submitted to the NRC staff six affidavits (five from persons who sought to remain anonymous) alleging poor quality work and safety problems at the plant. *Id.* at 2-3. Applicant's subpoenas, in GAP's view, are designed to determine the identities of the Midland whistleblowers and thereby deter others from coming forward in the future. *Id.* at 3.

The Licensing Board denied the motion. It concluded, in agreement with both applicant and the staff, that GAP's motion was "premised on the false notion that the Applicant is seeking to expose the identity of the confidential informants." LBP-83-53, *supra,* 18 NRC at 286. *See* CPC Application for Deposition Subpoenas at 2-3, Schedule of Documents Requested at 2. It thus found it unnecessary to reach the question of privilege.[3] As for GAP's claim of estoppel, the Board noted that the Commission's "assurance . . . of nondisclosure" went only to the informants' identities. 18 NRC at 286. The Board did, however, take steps — at the staff's urging and without objection by applicant — to assure that the release of the *contents* of the affidavits will not inevitably or inadvertently lead to the disclosure of the affiants' *identities,* and so undermine GAP's credibility. Specifically, the Board entered a protective order providing that (1) the informants' identities, and any information that might reasonably lead to their disclosure, need not be provided on deposition or in the subpoenaed documents; (2) if such information is inadvertently disclosed, it shall be deleted from the transcript and not revealed by those present; (3) all information elicited is to be restricted to applicant's *counsel,* the NRC staff, intervenors, and, to the extent necessary, the Board itself; and (4) applicant, the staff, and deponents may present to the Licensing Board for resolution any dispute over what constitutes protected information. *Id.* at 289-90. *See* 10 C.F.R. §§ 2.720(f), 2.740(c). The Board also repeated the direction of its

[3] The Board did note, however, that the privileges asserted are not absolute and would require a balancing of the need for the information against the harm in revealing it. Under such a test, the Board "would not in any event quash the instant subpoenas on the basis of privilege." LBP-83-53, *supra,* 18 NRC at 288-89.

Memorandum of July 8, 1982, that the scope of the depositions and documents sought under these subpoenas be

> *limited* to "[that information] relevant to the matters already at issue in the OL/OM (including admitted contentions) proceedings." In that connection, the manner in which GAP generally obtains information would not be relevant; the manner in which it obtained particular information relevant to particular contentions or issues might be relevant.

18 NRC at 287.[4] *See* LBP-82-118, 16 NRC 2034, 2047-50, 2057-61 (1982), for a description of the pertinent "matters already at issue" in this proceeding.

Still dissatisfied, GAP asked the Licensing Board to reconsider its denial of the motion to quash. The Board denied the motion for reconsideration but elaborated somewhat on its earlier decision. It pointed out that, although it found it unnecessary to decide if either asserted privilege applied, it did in fact undertake the balancing test employed where such a privilege is found to exist. LBP-83-64, *supra,* 18 NRC at 768-69. *See* note 3, *supra.* The Board emphasized the value of the protective order imposed and found GAP's expressed apprehension of a breach of that order baseless. 18 NRC at 769-70. The Board also stressed the limits on the scope of discovery permitted under the subpoenas and provided further guidance in that regard. *Id.* at 771-72.

GAP now appeals the Licensing Board's two orders denying its motion to quash the subpoenas.[5] Both applicant and the staff urge that we affirm the Board's decision.[6]

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[4] In the same order, the Board granted a separate motion to quash the subpoenas to the extent they requested testimony and documents concerning communications between GAP and two intervenors in this proceeding. The Board based its decision on attorney-client privilege. *See id.* at 284-86. That matter is not before us in this appeal.

[5] GAP sought a stay of the Board's decision from both the Licensing Board and us. In each instance, the stay request was denied. *See* LBP-83-64, *supra,* 18 NRC at 768, 772, 773; Appeal Board Memorandum of October 6, 1983 (unpublished); Appeal Board Order of October 27, 1983 (unpublished). No party, however, has requested an expedited appeal.

When GAP failed on October 27, 1983, to produce the subpoenaed documents, applicant moved to compel and sought court enforcement of the subpoenas. CPC Motion to Compel and Application for Enforcement of Subpoenas (November 2, 1983). The GAP deponents responded that they "will not appear for depositions unless and until ordered to do so by a court of the United States." GAP Deponents' Response (November 4, 1983) at 2 n.*. *See Solargen Electric Motor Car Corp. v. American Motor Corp.,* 506 F. Supp. 546, 552 (N.D.N.Y. 1981), where the court was "greatly bothered by the unreasonable refusal of the [subpoenaed persons] to even appear at their designated depositions. . . ." The Licensing Board granted applicant's motion and asked the NRC's General Counsel to seek court enforcement of the subpoenas. Memorandum and Order of November 8, 1983 (unpublished). *See* 10 C.F.R. § 2.720(g). Apparently further action in that regard has been deferred pending resolution of the instant appeal.

[6] Applicant has moved for leave to file corrected copies of its brief. The purpose of the motion is to provide the table of contents and table of authorities omitted from its original filing more than six weeks

*(Continued)*

## II.

On appeal, GAP renews two of the three arguments it made to the Licensing Board, specifically those concerning the "privileged" nature of the information solicited by the subpoenas.[7] Its principal argument is that the subpoenas impair "its First Amendment right freely to collect information from confidential sources about the safety problems at the Midland nuclear plants." GAP Memorandum in Support of Appeal (October 20, 1983) at 4. Because it gathers information from confidential sources and passes it on to the NRC for investigation, GAP claims that, like journalists and scholars, it is entitled to the qualified privilege — footed in the First Amendment guaranty of a free press — not to disclose its sources.[8] GAP's second and independent argument is that the subpoenas violate the common law principle of confidential communications. *Id.* at 9-10. *See* 8 Wigmore, *Evidence* § 2285 (J. McNaughton rev. 1961). We find no merit to either argument.

A. As noted, GAP draws an analogy between itself and the press, contending that "[t]he growing line of cases which protects journalists and news editors in their news-gathering and editorial functions clearly protects GAP's information-gathering which serves the public interest." GAP Memorandum in Support of Appeal at 5. That the press enjoys a qualified privilege not to reveal its sources in certain circumstances is beyond doubt. *Branzburg v. Hayes,* 408 U.S. 665, 709-10 (1972) (Powell, J., concurring); *United States v. Cuthbertson,* 630 F.2d 139, 147 (3d Cir. 1980), *cert. denied,* 449 U.S. 1126 (1981); *Silkwood v. Kerr-McGee Corp.,* 563 F.2d 433, 436-37 (10th Cir. 1977); *Carey v. Hume,* 492 F.2d 631, 636 (D.C. Cir.), *cert. dismissed,* 417 U.S. 938 (1974); *Baker v. F&F Investment,* 470 F.2d 778, 783 (2d Cir. 1972), *cert. denied,* 411 U.S. 966 (1973). We can find no basis, however, for expanding the press' qualified privilege to encompass GAP's activities.

First, courts traditionally have been loath to create a new testimonial privilege or to extend an existing one, "since such privileges obstruct the search for truth." *Branzburg, supra,* 408 U.S. at 690 n.29 (plurality opinion). *See Herbert v. Lando,* 441 U.S. 153, 175 (1979). Further, all citizens have a "general duty . . . to provide evidence when necessary to further the system of justice." *Wright v. Jeep Corp.,* 547 F. Supp. 871,

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earlier. *See* 10 C.F.R. § 2.762(c). We *grant* the motion, although the proffered material came too late to be of any real value. In the future, we expect applicant's counsel, as well as the other parties, to conform their pleadings to the Commission's Rules of Practice. *See* 10 C.F.R. § 2.762(f).

[7] GAP no longer presses its "estoppel" claim. *See* p. 637, *supra*.

[8] GAP here abandons one of the First Amendment arguments it pressed before the Licensing Board — *i.e.*, that it serves as a conduit through which citizens can "petition the Government for a redress of grievances." U.S. Const. amend. I. *See* GAP Motion to Quash at 4-6.

875 (E.D. Mich. 1982). *See Branzburg, supra,* 408 U.S. at 688 (plurality opinion).

Second, despite GAP's suggestion to the contrary, the qualified privilege of the press has been consistently and strictly limited to those reasonably characterized as part of the media. *Compare, e.g.,* the following cases where the privilege has been recognized: *Cuthbertson, supra* (Columbia Broadcasting System); *Silkwood, supra* (free-lance documentary filmmaker); *Baker, supra* (journalist); *Solargen,* note 5, *supra* (television cameraman); *In re Consumers Union of the United States, Inc. (Starks v. Chrysler Corp.),* 32 Fed. R. Serv. 2d 1373 (S.D.N.Y. 1981) (publisher of *Consumer Reports*); *Apicella v. McNeil Laboratories, Inc.,* 66 F.R.D. 78 (E.D.N.Y. 1975) (publisher of *The Medical Letter on Drugs and Therapeutics*); with *Wright v. Patrolmen's Benevolent Ass'n,* 72 F.R.D. 161 (S.D.N.Y. 1976) (journalist's privilege *not* extended to bar association that conducted investigation of transfer of judge from criminal to civil bench).[9] GAP does not purport to be part of the fourth estate, nor could it, given the description of its work provided by its Executive Director. *See* GAP Motion to Quash, Affidavit of Louis Clark (June 24, 1983) at 2-3. Moreover, although it does perform some information gathering and disseminating functions on a confidential basis, that alone is not enough to convert GAP into a branch of the media. By its own account, GAP is a public interest group that offers assistance to corporate and governmental whistleblowers. It does not place information directly into the public "marketplace of ideas." *Apicella, supra,* 66 F.R.D. at 84. By no reasonable measure can it be deemed "the press" for the purpose of invoking the qualified privilege of a journalist not to reveal confidential communications.[10]

Even if GAP were within the ambit of this qualified privilege, the outcome here would be no different. Where the courts have recognized a

[9] Indeed, the plurality opinion in *Branzburg* speculated on the "practical and conceptual difficulties" of creating a broad and absolute privilege bottomed on the function of disseminating information. 408 U.S. at 703-05.

[10] GAP's reliance on a "scholar's privilege" — an alleged outgrowth of the journalist's privilege — is similarly without basis. GAP can no more be fairly characterized as part of the academic community than part of the media. More important, support for a recognized scholar's privilege cannot be found in the cases on which GAP relies.

In *Richards of Rockford, Inc. v. Pacific Gas & Electric Co.,* 71 F.R.D. 388, 389 & n.2, 390 (N.D. Cal. 1976), the court explicitly declined to decide if such a privilege exists, deciding the case on other grounds. Neither the majority nor concurring opinion in *United States v. Doe (In re Popkin),* 460 F.2d 328 (1st Cir. 1972), *cert. denied,* 411 U.S. 909 (1973), expressly recognizes a scholar's privilege. Finally, while *United States v. Doe (In re Falk),* 332 F. Supp. 938, 941 (D. Mass. 1971), might be read as according First Amendment rights to a professor, that case relied on a lower court opinion overturned in *Branzburg* and its rationale is thus suspect. On the other hand, more recent authority has clearly rejected the notion of a scholar's privilege. *See Wright v. Jeep Corp., supra,* 547 F. Supp. at 875-76. *See also In re Dinnan,* 661 F.2d 426, 427-31 (5th Cir. 1981), *cert. denied,* 457 U.S. 1106 (1982) (rejection of "academic privilege" against testifying).

journalist's privilege, they have balanced "the potential harm to the free flow of information that might result against the asserted need for the requested information." *Bruno & Stillman, Inc. v. Globe Newspaper Co.*, 633 F.2d 583, 596 (1st Cir. 1980) (footnote omitted). *See Branzburg, supra,* 408 U.S. at 710 (Powell, J., concurring); *Cuthbertson, supra,* 630 F.2d at 148; *Carey, supra,* 492 F.2d at 636-39; *Solargen, supra,* 506 F. Supp. at 550. The principal factors considered are whether the requested information is relevant and goes to the heart of the matter at hand, and whether the party seeking the information has tried to obtain it from other possible sources. *Silkwood, supra,* 563 F.2d at 438; *Baker, supra,* 470 F.2d at 783. Although the Licensing Board found it unnecessary to decide whether GAP was entitled to assert a First Amendment privilege, the Board in fact balanced these facts as if the privilege did apply.

In particular, the Board determined that the requested information is relevant to certain issues in this licensing proceeding[11] and noted applicant's inability to obtain the information elsewhere. LBP-83-53, *supra,* 18 NRC at 287, 288; LBP-83-64, *supra,* 18 NRC at 771-72. *See* LBP-82-118, *supra,* 16 NRC at 2047-50, 2057-61; CPC Application for Deposition Subpoenas at 2. *See also* Memorandum of CPC in Opposition to Appeal (December 9, 1983) at 13-14. The Board also concluded that the protective order it was imposing would eliminate the harm GAP perceived to its interest. LBP-83-53, *supra,* 18 NRC at 288; LBP-83-64, *supra,* 18 NRC at 768-69. *See* pp. 643-44, *infra.* It then weighed this factor against the others and — quite reasonably, in our view — denied the motion to quash. LBP-83-53, *supra,* 18 NRC at 288-89. *See also* LBP-83-64, *supra,* 18 NRC at 769, 771.

B. GAP also contends that the subpoenas should be quashed as a matter of the "common law of privilege." It refers us to no case authority, but relies on Wigmore's statement of the "four fundamental conditions . . . recognized as necessary to the establishment of a privilege against the disclosure of communications:"

> (1) The communications must originate in a *confidence* that they will not be disclosed.
>
> (2) This element of *confidentiality must be essential* to the full and satisfactory maintenance of the relation between the parties.
>
> (3) The *relation* must be one which in the opinion of the community ought to be sedulously *fostered.*

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[11] GAP conceded as much in its Motion to Stay Depositions (October 26, 1983) at 3.

(4) The *injury* that would inure to the relation by the disclosure of the communications must be *greater than the benefit* thereby gained for the correct disposal of litigation.

8 Wigmore, *Evidence* § 2285 (J. McNaughton rev. 1961) (footnote omitted). GAP contends that it has satisfied all four factors, and thus the subpoenas for its testimony and documents should be quashed.

We note at the outset that Wigmore used these four conditions simply as a convenient framework for discussing already recognized privileges (*e.g.*, attorney-client, spousal, government informer). *Ibid.*[12] In no respect did he suggest that new privileges should be lightly created, even by statute. Indeed, Wigmore characterizes as "[t]he sounder attitude" several reports strongly disapproving the creation of so-called "novel privileges." *Id.*, § 2286.

Be that as it may, GAP fails to meet three of the four conditions — (1), (2), and (4) — specified by Wigmore. First, the communications did not originate in a confidence that they would not be disclosed. On the contrary, GAP was requested to "act as an intermediary for the presentation of [the utility and construction workers'] information to the NRC." GAP Motion to Quash, Affidavit of Louis Clark at 4. GAP also discussed with the press some of the workers' specific allegations. CPC Application for Deposition Subpoenas, Attachment No. 1 at 2. *See* GAP Deponents' Motion for Reconsideration (September 30, 1983), Affidavit of Billie Pirner Garde (September 30, 1983) at 2, 5. Moreover, GAP's brief on appeal refers to its role in facilitating "the full and free flow of information to the NRC *and to the public.*" GAP Memorandum in Support of Appeal at 9 (emphasis added). These statements and actions belie any notion that the communications were, or were intended to be, completely confidential. Only the informants' *identities* were intended to be kept anonymous. GAP Motion to Quash, Affidavit of Louis Clark at 4, 6.[13] And, as discussed below, the Board's protective order is designed to accomplish that purpose.

By the same token, preserving the confidentiality of the information supplied to GAP could not possibly have been contemplated by — and thus essential to the relationship between — GAP and its informers. As noted, the information was intended for dissemination to at least the

[12] The way courts nave used this section of Wigmore bears this out. *See, e.g., Somer v. Johnson*, 704 F.2d 1473, 1479 n.6 (11th Cir. 1983); *Reporters Committee for Freedom of the Press v. American Telephone & Telegraph Co.*, 593 F.2d 1030, 1050 n.67 (D.C. Cir. 1978), *cert. denied*, 440 U.S. 949 (1979).

[13] Even then, the NRC explained that, while it was its policy not to divulge allegers' identities, it could not guarantee preservation of anonymity in all circumstances. GAP Motion to Quash, Affidavit of Louis Clark, Attachment 1 (Letter of J.G. Keppler to B. Garde).

NRC, while only the informers' identities were meant to be protected. The last factor, injury to the relationship from disclosure greater than the benefit to the litigation, simply reflects the balancing test employed by the courts in the journalist's privilege cases discussed above. As we explained at pp. 640-41, *supra,* the Licensing Board correctly weighed the competing interests involved here in favor of disclosure. GAP has therefore failed to make a case for according it the "common law of privilege."

C. GAP's most significant shortcoming on appeal is its total failure to address the protective order imposed by the Licensing Board. We must presume, from the very prosecution of this appeal, that GAP regards that order as deficient in some respect. Yet GAP has provided us with no explanation whatsoever of how it is inadequate. Protection of the *identities* of GAP's sources is the predominant theme of its motion filed with the Licensing Board. *See* GAP Motion to Quash at 3, 7, 8, 10, and Affidavit of Louis Clark at 4, 6. Accordingly, the most important feature of the protective order is its inclusion of precautions against even inadvertent breaches of anonymity. Disclosures are also to be limited both in scope and to only applicant's *counsel,* the NRC staff (which already has the information), intervenors (who are being counseled by GAP and are not likely to harass the informers), and the Licensing Board. Disputes over what constitutes protected material are to be presented to the Board for resolution. LBP-83-53, *supra,* 18 NRC at 289-91. GAP leaves us to ponder why these measures are not responsive to its fear of irreparable damage to its institutional integrity.[14]

The imposition of a protective order in the circumstances of this case is a sound and pragmatic action designed to facilitate essential discovery while protecting GAP's asserted interests — interests that we have found not privileged. Moreover, the Licensing Board's order is fully consistent with the approach taken by the courts, even where a qualified privilege is found to exist. *See, e.g., Bruno & Stillman, supra,* 633 F.2d at 598, where the court describes various options available, including one of the measures employed here by the Licensing Board — limiting attendance at and distribution of the depositions.[15]

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[14] The Licensing Board correctly pointed out that we assume protective orders will be obeyed unless a concrete showing to the contrary is made. *See* LBP-83-53, *supra,* 18 NRC at 287-88 and cases cited. In seeking reconsideration, GAP attempted but failed to do just that. *See* LBP-83-64, *supra,* 18 NRC at 769-70. The Licensing Board nonetheless pointed out that one who violates such orders risks " 'serious sanction.' " *Id.* at 769. *See* 10 C.F.R. § 2.713. GAP does not pursue this on appeal, and we see no basis for contradicting the Board's conclusion that its protective order is unlikely to be violated.

[15] GAP relies on *Machin v. Zuckert,* 316 F.2d 336 (D.C. Cir.), *cert. denied,* 375 U.S. 896 (1963), which involved the *government's* assertion of privilege with respect to a U.S. Air Force crash report. Although

*(Continued)*

In affirming the Licensing Board's denial of GAP's motion to quash, we do not denigrate the important role that GAP and similar organizations may play in uncovering possible wrongdoing and waste. Nor are we insensitive to informants' fears — warranted or not — of harassment and reprisal, should their identities become known. On the other hand, significant questions about quality assurance at applicant's facility have been raised in this litigation. GAP has information bearing on those issues, and applicant is entitled to learn the nature of it through reasonable discovery methods. *See Herbert v. Lando, supra,* 441 U.S. at 177; 10 C.F.R. § 2.740(b)(1).[16] We believe the Licensing Board's protective order successfully and fairly accommodates all of these competing interests.

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The Licensing Board's denial of GAP's motion to quash is *affirmed.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

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that case is not only inapposite but also probably superseded by the subsequent enactment of the Freedom of Information Act, we note that the court there found a protective order to be a useful tool in dealing with the controversy at hand. *Id.* at 340, 341.

[16] We agree with the Licensing Board's observation that it is particularly unfair to applicant and the adjudicatory system itself for GAP to reveal to the press the information "confided" to it (*see* p. 642, *supra*), while refusing to subject it to scrutiny in related litigation. *See* LBP-83-64, *supra,* 18 NRC at 770-71.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING APPEAL BOARD**

**Administrative Judges:**

**Christine N. Kohl,** Chairman
**Gary J. Edles**
**Dr. Reginald L. Gotchy**

**In the Matter of** **Docket Nos. 50-352**
**50-353**

**PHILADELPHIA ELECTRIC COMPANY**
**(Limerick Generating Station,**
**Units 1 and 2)** **March 30, 1984**

The Appeal Board affirms (1) the Licensing Board's assertion of jurisdiction over an intervenor's contentions concerning the applicant's 10 C.F.R. Part 70 application for a license to receive and store new, unirradiated fuel outdoors at the Limerick site, and (2) dismissal of the contentions for lack of basis and specificity.

## MATERIALS LICENSE UNDER PART 70: NEED

A Special Nuclear Materials License is required for a person to "receive title to, own, acquire, deliver, receive, possess, use, or transfer special nuclear material." 10 C.F.R. § 70.3. Such authorization is essentially subsumed within a license to operate a commercial power reactor, issued pursuant to 10 C.F.R. Part 50.

## MATERIALS LICENSE UNDER PART 70: NEED

If a utility wants (or needs) to receive and store new fuel before an operating license is issued, the utility must obtain a Part 70 license.

## RULES OF PRACTICE: JURISDICTION OF LICENSING BOARDS

Under the Commission's Rules of Practice, licensing boards may "preside in such proceedings for granting, suspending, revoking, or amending licenses or authorizations as the Commission may designate, and to perform such other adjudicatory functions as the Commission deems appropriate." 10 C.F.R. § 2.721(a).

## RULES OF PRACTICE: JURISDICTION OF APPEAL BOARDS

Appeal boards are delegated authority to perform the Commission's review functions in Part 50 and other licensing proceedings specified by the Commission. 10 C.F.R. § 2.785(a).

## RULES OF PRACTICE: SCOPE AND TYPE OF PROCEEDING

Under 10 C.F.R. § 2.721(a), only the Commission can define the scope of a proceeding before a licensing board, or decide that a formal adjudicatory-type proceeding should be instituted.

## ATOMIC ENERGY ACT: HEARING REQUIREMENTS FOR MATERIALS

Section 189a of the Atomic Energy Act, 42 U.S.C. § 2239a, mandates a hearing for *any* licensing action where requested by a person "whose interest may be affected." But a formal, "on the record" adjudicatory-type hearing under Section 554 of the Administrative Procedure Act (APA), 5 U.S.C. § 554 — like those conducted by licensing boards — is not required for so-called materials licenses. *See Kerr-McGee Corp.* (West Chicago Rare Earths Facility), CLI-82-2, 15 NRC 232, 244-62 (1982), *aff'd sub nom. City of West Chicago v. NRC,* 701 F.2d 632 (7th Cir. 1983). The Commission can delegate authority to adjudicate such matters informally to an agency official, such as the Director of the Office of Nuclear Material Safety and Safeguards. *See, e.g., Kerr-McGee Corp.* (West Chicago Rare Earths Facility), CLI-82-21, 16 NRC 401 (1982).

## RULES OF PRACTICE: JURISDICTION OF LICENSING BOARDS

Licensing boards may assert jurisdiction over Part 70 issues raised in conjunction with an ongoing Part 50 licensing proceeding. *See Pacific*

*Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units Nos. 1 and 2), CLI-76-1, 3 NRC 73, 74 (1976). *See also, e.g., Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 & 2), LBP-83-38, 18 NRC 61, 63 (1983); *Cincinnati Gas and Electric Co.* (William H. Zimmer Nuclear Station), LBP-79-24, 10 NRC 226, 228-30 (1979).

## ATOMIC ENERGY ACT: NOTICE REQUIREMENT FOR MATERIALS LICENSES

It is not clear what, if any, notice requirements pertain to materials license cases. *See Armed Forces Radiobiology Research Institute* (Cobalt-60 Storage Facility), ALAB-682, 16 NRC 150, 157-59 (1982).

## RULES OF PRACTICE: ADMISSIBLE CONTENTIONS

Section 2.714(b) of 10 C.F.R. requires an intervenor in a proceeding to set forth the bases for its contention(s) with reasonable specificity. Where the laws of physics deprive a proposed contention of any credible basis, the contention will not be admitted. *Compare Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542 (1980).

## RULES OF PRACTICE: RESPONSIBILITIES OF PARTIES

Parties in Commission proceedings have a duty to alert the Boards and all other parties of any significant new information related to the proceeding. *See Tennessee Valley Authority* (Browns Ferry Nuclear Plant, Units 1, 2 and 3), ALAB-677, 15 NRC 1387, 1394 (1982).

## RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS

Under *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983), all five factors enumerated in 10 C.F.R. § 2.714(a)(1) must be considered and balanced before an untimely intervention petition may be granted or a late-filed contention admitted. This is so even where a party has succeeded in making a strong showing on the first of those factors (good cause).

TECHNICAL ISSUES DISCUSSED

Criticality Potential of New Fuel;
Handling and Storage of New Fuel at the Reactor Site;
Radiation Hazard from New Fuel.

## APPEARANCES

**Robert L. Anthony**, Moylan, Pennsylvania, for intervenor Friends of the Earth.

**Troy B. Conner, Jr., Mark J. Wetterhahn,** and **Robert M. Rader,** Washington, D.C., for applicant Philadelphia Electric Company.

**Benjamin H. Vogler, Joseph Rutberg, Ann P. Hodgdon, Nathene A. Wright,** and **Michael N. Wilcove** for the Nuclear Regulatory Commission staff.

# MEMORANDUM AND ORDER

Friends of the Earth (FOE), an intervenor in this operating license proceeding, appeals the Licensing Board's March 16, 1984, memorandum and order (and related oral rulings). *See* LBP-84-16, 19 NRC 857. In that order, the Board dismissed several new contentions by FOE concerning Philadelphia Electric Company's (PECo's) application, pursuant to 10 C.F.R. Part 70, to receive and store new, unirradiated fuel outdoors at the Limerick site for several weeks.[1] Before reaching that ultimate decision, the Board determined that, despite its specific authorization to conduct hearings on PECo's *operating* license application, it had jurisdiction over related Part 70 issues. The Board also concluded that FOE's contentions were not late-filed.

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[1] The Licensing Board's dismissal of FOE's contentions (and those of LEA, another intervenor, not at issue here) opened the way to the issuance of the Part 70 license. The Board's ruling is therefore an immediately appealable order. Delivery of the fuel was scheduled for this month; hence, FOE also seeks a stay of the Board's order. After learning of the imminent issuance of the license and subsequent likely movement of the fuel, we temporarily stayed the Board's order (and thereby fuel delivery) to permit receipt of FOE's brief and to prevent FOE's appeal from becoming effectively moot. Order of March 27, 1984 (unpublished).

As explained below, we ratify the Board's assertion of jurisdiction over this matter and affirm its dismissal of FOE's contentions for lack of basis and specificity.

## I. BACKGROUND

The Licensing Board's written ruling thoroughly sets out the background of this appeal. *See id.* at 860-62. We summarize the salient points chronologically here.

In June 1983, PECo filed with the Commission a Part 70 application for a Special Nuclear Materials License.[2] The following January and February, it amended the application to reflect a proposed March 1984 delivery date for 764 fuel bundles. The fuel bundles are to remain in their shipping containers but will be stored outdoors at the Limerick site for several weeks.[3] Staff counsel served copies of the amended application on the Licensing Board and parties on February 21, 1984. In a pleading dated February 23 and filed with the Licensing Board, FOE sought to introduce what are essentially several new contentions based on the Part 70 application.[4] Because of the proposed March delivery date for the fuel, the Board requested expeditious responses to FOE's pleading from PECo and the staff. At about the same time as those responses were submitted, FOE filed what it termed an "Addition" to its earlier paper.

The Board heard oral argument from the parties and on March 6 ruled from the bench that it was not admitting FOE's contentions. *See* Tr. 7909-23. It did, however, request certain affidavits from the staff and applicant to bolster one aspect of its oral rulings.[5] After receiving them, the Board issued LBP-84-16, confirming its earlier ruling and subject only to receipt of FOE's reply affidavit. With that in hand, the Board reconfirmed its earlier oral and written rulings. Tr. 8846-48; Licensing Board Memorandum and Order of March 26, 1984 (unpublished).

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[2] This license is required for a person to "receive title to, own, acquire, deliver, receive, possess, use, or transfer special nuclear material" (*e.g.*, the fuel used in a reactor like Limerick). 10 C.F.R. § 70.3. Such authorization is essentially subsumed within a license to operate a commercial power reactor, issued pursuant to 10 C.F.R. Part 50. If the utility wants (or needs) to receive and store new fuel before that operating license is issued, the utility must obtain a Part 70 license.

[3] Though "outdoors," the stacks of containerized fuel assemblies will be sheltered by a five-sided corrugated metal box. PECo's Amended Application for Special Nuclear Material License, § 1.2.1.1 (rev. February 17, 1984).

[4] FOE is already an intervenor, involved in the litigation of several contentions concerning PECo's operating license application.

[5] In so doing, the Board emphasized that the affidavits were to address the *bases* of the contentions; it was "not talking about summary disposition." Tr. 7920. The affidavits may be properly viewed as supplemental responses to FOE's initial request to admit the contentions and its "Addition" thereto.

Apparently not certain of when it should appeal, FOE sought our intercession after the Board's initial bench ruling but before the issuance of LBP-84-16. Once all doubt as to the "finality" of the Board's opinion was removed, FOE renewed its intent to appeal, and we abbreviated the prescribed briefing schedule in light of the likely imminence of fuel movement.

## II. JURISDICTION

Under the Commission's Rules of Practice, licensing boards may "preside in such proceedings for granting, suspending, revoking, or amending licenses or authorizations as *the Commission* may designate, and to perform such other adjudicatory functions as *the Commission* deems appropriate." 10 C.F.R. § 2.721(a) (emphasis added).[6] The Commission's order directing the Licensing Board to preside in this proceeding is limited on its face to issues relating to PECo's operating license. *See* 46 Fed. Reg. 42,557 (1981).[7] Relying on Commission precedent and a common sense reading of the Rules of Practice, however, the Licensing Board concluded that it was appropriate for it to *assert* jurisdiction over FOE's Part 70 filings as well. LBP-84-16, *supra,* 19 NRC at 862-64. We agree with the Board's reasoning.

In *Diablo Canyon, supra* note 6, the Commission expressly approved the Licensing Board's assertion of jurisdiction over Part 70 issues raised in conjunction with an ongoing Part 50 licensing proceeding. The Commission noted that, under 10 C.F.R. § 2.721, it could delegate such authority to the Licensing Board. More important, it stressed that the Part 70 materials license involved there was "integral to the Diablo Canyon project." The Commission also commented that "[g]iven that Board's familiarity with the Diablo Canyon project, it made good practical sense

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6 Similarly, appeal boards are delegated authority to perform the Commission's review functions in Part 50 and other licensing proceedings specified by the Commission. 10 C.F.R. § 2.785(a). The Commission has expressly delegated us authority to exercise the review functions over the Part 70 issues raised here that it would ordinarily perform. Commission Order of March 22, 1984 (unpublished). *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units Nos. 1 and 2), CLI-76-1, 3 NRC 73, 74 (1976). At the same time, the Commission noted that it was not deciding whether the Licensing Board, in fact, had jurisdiction over these Part 70 issues, or whether there should be formal adjudication in this type of case. Commission Order of March 22, 1984, *supra,* at 2 n.1.

7 In a notice dated March 16, 1984, and served ten days later, the Chairman of the Atomic Safety and Licensing Board Panel "established" the Licensing Board below "to rule on the admissibility of Part 70 issues raised in this proceeding." That notice, of course, cannot be construed as an actual delegation of subject matter jurisdiction to the Board. As noted above, under 10 C.F.R. § 2.721(a), only the Commission can define the scope of a proceeding before a licensing board, or decide that a formal adjudicatory-type proceeding should even be instituted. *See* Commission Order of March 22, 1984, *supra,* at 2 n.1. *See also* p. 651, *infra.*

for it to hear and decide the related issues raised by the Part 70 materials license application." 3 NRC at 74 n.1.

PECo's materials license is no less "integral" to Limerick.[8] As in *Diablo Canyon,* it is necessary for PECo to receive and store new fuel assemblies in advance of the issuance of its requested operating license. It also "made good practical sense" for the Licensing Board, so familiar with the Limerick facility and engaged in hearings at the time, to rule on the admissibility of FOE's proffered contentions. The need for expeditious attention to this matter, prompted by the proposed March fuel delivery date, further validates the Board's quick and responsible action.

This is not to say that a licensing board is *required* to consider Part 70 issues, or even that it should do so in all circumstances. To be sure, Section 189a of the Atomic Energy Act, 42 U.S.C. § 2239a, mandates a hearing for *any* licensing action where requested by a person "whose interest may be affected." It is now clear, however, that a formal, "on the record" adjudicatory-type hearing under Section 554 of the Administrative Procedure Act (APA), 5 U.S.C. § 554 — like those conducted by licensing boards — is not required for so-called materials licenses. *See Kerr-McGee Corp.* (West Chicago Rare Earths Facility), CLI-82-2, 15 NRC 232, 244-62 (1982), *aff'd sub nom. City of West Chicago v. NRC,* 701 F.2d 632 (7th Cir. 1983).[9] The Commission is free to delegate authority to adjudicate such cases informally to an agency official, such as the Director of the Office of Nuclear Material Safety and Safeguards (NMSS). *See, e.g., Kerr-McGee Corp.* (West Chicago Rare Earths Facility), CLI-82-21, 16 NRC 401 (1982).[10]

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[8] Applicant argued to the Licensing Board that the Part 70 issues must concern the same matters that are being litigated in the operating license proceeding. *Diablo Canyon,* however, suggests no such requirement. In any event, FOE's contentions assertedly relate to at least two matters under litigation in the proceeding — the ability of safety-related buildings to withstand overpressures and impacts from off-site accidents, and emergency planning.

[9] *Kerr-McGee* involved a 10 C.F.R. Part 40 license for the possession, use, etc., of "source material" (*e.g.,* uranium ore). The informal hearing requirement of Section 189a, however, applies to all types of materials licenses, whether arising under 10 C.F.R. Part 30, 40, or 70. Thus, the holding of *Kerr-McGee* fully pertains to Part 70 matters.

[10] On appeal, FOE contends that Section 182b (*sic* — Section 182c) of the Atomic Energy Act, 42 U.S.C. § 2232c, requires notice of PECo's application for the Part 70 license. FOE also cites 10 C.F.R. §§ 72.34, 2.104, and 2.105 as further evidence of the Commission's obligation to provide notice of an application for a Part 70 license.

FOE is in error on all counts. Section 182c of the Act requires notice of the application for the license to operate the power plant itself — "a utilization or production facility for the generation of commercial power." 42 U.S.C. § 2232c. *See* 42 U.S.C. §§ 2014v, cc (definitions of "production facility" and "utilization facility"). It does not refer to a Part 70 materials license to receive and store special nuclear material. *See* note 2, *supra.* Nor do the Commission's Rules of Practice, 10 C.F.R. §§ 2.104, 2.105, require such notice. Section 2.104 requires notice of an application where a hearing is "required" or the Commission has found a hearing to be in the public interest. Under a court-approved Commission interpretation, Section 2.104 does not apply to materials license cases. *See City of West Chicago, supra,*
*(Continued)*

The consistent agency practice, however, is for licensing boards, already presiding at operating license hearings, to act on requests to raise Part 70 issues involving the same facility. *See, e.g., Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 & 2), LBP-83-38, 18 NRC 61, 63 (1983); *Cincinnati Gas and Electric Co.* (William H. Zimmer Nuclear Station), LBP-79-24, 10 NRC 226, 228-30 (1979).[11] In neither of these cases did the Commission intercede to terminate the Boards' action. *See also Armed Forces Radiobiology Research Institute* (Cobalt-60 Storage Facility), Docket No. 30-6931, Commission Order (October 8, 1981) (unpublished) (Commission notes ongoing proceeding for renewal of research reactor operating license and refers request for hearing on Part 30 license for same facility to a licensing board). The *Limerick* Licensing Board's assertion of jurisdiction is consistent with this practice and fully justified.

## III. THE CONTENTIONS

### A. Basis and Specificity

The precise nature of FOE's proposed contentions was not entirely clear to the Licensing Board. It identified, however, a number of areas of concern vis-a-vis the Part 70 license, reflected in FOE's original filing and subsequent "Addition": (1) the ability of safety-related buildings to withstand overpressures and impacts from offsite accidents; (2) the not yet final Independent Design Verification Program for the facility; (3) the qualification of the overhead cranes for handling fuel; (4) the incompleteness of the emergency plan; (5) natural hazards such as tornadoes and electrical storms; (6) theft and sabotage; (7) the hazard posed

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701 F.2d at 639. Although Section 2.105 lists other types of proposed action where the Commission is committed to providing notice, it does not include Part 70 materials licenses. *See id.* at 639-40. 10 C.F.R. § 72.34 is inapposite: it requires notice of a Part *72* application to license an *independent spent* fuel storage installation (ISFSI). *See* 10 C.F.R. § 72.3(m) (definition of "ISFSI"). That, of course, is not what PECo seeks here through its Part 70 application.

It is not clear whether any other statutory or regulatory provision requires notice of materials license action. *See Armed Forces Radiobiology Research Institute* (Cobalt-60 Storage Facility), ALAB-682, 16 NRC 150, 157-59 (1982) (Eilperin, concurring). As in *AFFRI,* however, it is not necessary that we resolve the issue because FOE had actual notice of the Part 70 application after February 21, 1984, when staff counsel served the Licensing Board and all parties with PECo's amended application. From this very appeal, it is evident that FOE had the opportunity to seek to litigate issues arising from the Part 70 application. It therefore has not been prejudiced, in fact, by the lack of any formal notice. *But see* p. 657 & n.20, *infra.*

[11] In *Pennsylvania Power & Light Co.* (Susquehanna Steam Electric Station, Units 1 and 2), Docket Nos. 50-387, 50-388, Licensing Board Memorandum and Order (May 20, 1981) at 28-29 (unpublished), the Licensing Board declined to assert jurisdiction over Part 70 issues *at that time* because it obviously believed it would reach an expedited decision on the operating license first, obviating the Part 70 license itself. *See* note 2, *supra.*

by a design-basis railway car explosion; and (8) possible "activation" (*i.e.*, criticality) of the fuel by an accident involving underground and overhead electrical lines. LBP-84-16, *supra,* 19 NRC at 869. The Board addressed these items collectively from two perspectives — whether the new fuel is likely to go critical, and whether the public health and safety can be threatened by the release of radioactive materials through some means not involving criticality.

As to the former, the Board emphasized that, based on its own collective knowledge, there is no credible explanation of how unirradiated fuel, as stored at Limerick, can go critical in any of the situations mentioned by FOE. It referred FOE to our general discussion of this matter in *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units Nos. 1 and 2), ALAB-334, 3 NRC 809, 817-20 (1976), and indicated that this proceeding was not intended as a forum in which to litigate the laws of physics. LBP-84-16, *supra,* 19 NRC at 869-70. On a similar basis, the Board also found no credible mechanism, not involving criticality, that would subject the public to harmful radioactive releases from the low-enriched, unirradiated fuel to be stored at Limerick. *Id.* at 870. To verify this finding, the Board obtained affidavits from the staff and applicant, as well as from FOE. These statements led the Board to confirm its earlier written ruling. Tr. 8846-48; Licensing Board Memorandum and Order of March 26, 1984, *supra.*

Further, the Board specifically addressed FOE's concern about the overhead crane.[12] It noted the staff's finding that certain lifting devices attached to the crane are "non-conforming" as to heavy weights (100,000 pounds or more). But the Board stressed that the staff does not question the crane's ability to lift together even several of the lighter new fuel containers, and PECo has explained why the crane will not be used to lift heavier loads over the new fuel storage area. LBP-84-16, *supra,* 19 NRC at 871. With respect to FOE's concern about theft and sabotage, the Board found that FOE failed to allege any specific inadequacy in the security plan and noted that it (the Board) could discern no general cause to doubt the plan's sufficiency. *Id.* at 874. The Board therefore dismissed all of FOE's contentions for failure to satisfy the basis and specificity requirements of 10 C.F.R. § 2.714(b).

FOE's purported contentions are unfocused and contain no attempt to identify with reasonable specificity the basis of the perceived risks from the temporary outdoor storage of unirradiated fuel assemblies packed in

[12] In doing so, the Board emphasized that its decision did not depend on "whether the non-conforming crane could somehow crush new fuel." LBP-84-16, *supra,* 19 NRC at 871. The Board simply sought to aid FOE's understanding of the matter.

special shipping containers. The Licensing Board quite properly dismissed them on that ground. But even though FOE's filings were thus deficient, the Board was sensitive to what it believed to be FOE's fundamental misapprehension about the delivery and temporary outdoor storage of the fuel — *i.e.*, that the fuel could somehow go critical with a corresponding risk to the public, or that through some noncriticality mechanism the fuel could release harmful radiation. The Board correctly pointed out that there is simply no conceivable, credible explanation for either to occur.[13]

Perhaps some additional elaboration will assuage FOE's concerns. As we explained in *Diablo Canyon*, ALAB-334, *supra*, for criticality — *i.e.*, a stable chain reaction — to occur, four factors must be present: (1) a sufficient supply of uranium fuel; (2) a "moderator" (usually a significant amount of water); (3) a proper geometric pattern of fuel rods within each fuel assembly, and of the fuel assemblies themselves, with the fuel/moderator ratio within certain limits; and (4) careful control of the heat produced by fission. 3 NRC at 818-19. We went on to explain how each of these factors must be controlled to maintain criticality or else the chain reaction will terminate. *Id.* at 819. Even construing FOE's contentions in a manner most favorable to FOE, we can see no way that these conditions can be achieved in the situations that FOE postulates. For example, it is simply not credible that dropping of the fuel, a tornado, electrical cables, or a nearby explosion could cause the removal of the protective packaging around enough of the fuel *and* stack it in some source of water in the required configuration, so as to achieve criticality.

FOE's fears of radiation hazard from unirradiated, noncritical fuel also are generalized and thus without basis. Moreover, even assuming the complete *absence* of the protective containers for the fuel assemblies (through unexplained means), the ceramic uranium dioxide fuel pellets, at the enrichment level involved here, would emit radiation at levels well below the dose limits set by the Commission in 10 C.F.R. Part 20. *See* Affidavit of Norman Ketzlach (March 13, 1984); Affidavit of Lubomir B. Pyrih (March 13, 1984) at 1-6; Affidavit of Paul S. Stansbury

[13] This case is thus distinguishable from *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542 (1980). There we found a *pro se* petitioner's inartfully drafted contention, asserting that a marine biomass farm would be environmentally preferable to the Allens Creek nuclear power plant, specific enough to be admitted for litigation, even though there was "appreciable room for doubt" as to its merit. *Id.* at 546-49. In this case, it is the laws of physics and the physical properties of the unirradiated fuel that deprive FOE's purported contentions of any credible or arguable basis.

(March 12, 1984).[14] The *presence* of the shipping containers, designed and licensed under 10 C.F.R. Part 71,[15] adds yet another level of protection against any perceived hazard. *See* Pyrih Affidavit at 6-12.[16]

There are other reasons as well why FOE's contentions should not or need not be admitted for litigation. (1) The ability of safety-related buildings to withstand overpressures and impacts from offsite accidents is already being litigated as contentions V-3a and V-3b in the operating license proceeding. (2) FOE has articulated no connection between the apparently ongoing Independent Design Verification Program and its concern about the temporary *outdoor* storage of unirradiated fuel pursuant to the requested Part 70 license. (3) As fully explained by the Licensing Board, the non-conformance of the overhead crane, identified by the staff, is not related to the storage of new fuel. *See* LBP-84-16, *supra,* 19 NRC at 871. (4) The Commission's regulations do not require an emergency plan to be in place for the particular activity sought to be licensed here under Part 70. *See* 10 C.F.R. §§ 70.22(i), 70.23(a)(11). (5) FOE fails to provide any specifics whatsoever concerning the alleged "risk of theft and sabotage." PECo's Part 70 application, § 1.2.1.1, as amended, states that the outdoor New Fuel Storage Area will be enclosed by an eight-foot fence, subject to 24-hour surveillance by a watchman, and illuminated at night. Further, LEA (*see* note 1, *supra*), which raised concerns about the adequacy of the security plan for this new fuel storage, has entered a stipulation with PECo whereby LEA is permitted to review that plan subject to a protective order. FOE was

[14] FOE is apparently concerned with the release of uranium oxide "dust." *See* Tr. 7908. As noted, the fuel is in ceramic pellet form. Only if removed from the fuel rod cladding and deliberately ground or cut could the pellets be transformed into "dust." *See* Ketzlach Affidavit, *supra,* at 2. FOE does not provide a credible scenario as to how this could occur. *See* Response of Anthony/FOE to Affidavits (March 19, 1984).

[15] FOE argues that the shipping containers do not conform to current NRC regulations. *Ibid.* According to applicant, the containers meet the standards in effect at the time the application to *transport* the fuel was filed (March 1982). Pyrih Affidavit at 7. Since that time, the Commission has amended 10 C.F.R. Part 71 "to make [these regulations] compatible with those of the International Atomic Energy Agency." 48 Fed. Reg. 35,600 (1983). Some substantive changes were made, but "the Commission's basic standards for radioactive material packaging remain unchanged." *Ibid.* In fact, the 1982 and current standards relevant to our inquiry here — *i.e.,* the ability of the containers to withstand certain hypothetical accident conditions — are virtually identical in all material respects. *Compare* 10 C.F.R. Part 71, App. B (1983), with 48 Fed. Reg. 35,616-17 (1983) (to be codified at 10 C.F.R. § 71.73).

To the extent FOE argues on appeal that a packaging standard other than that reflected in the pertinent Commission regulations should apply, 10 C.F.R. § 2.758(a) precludes litigation of such an issue.

[16] FOE contends that the "stringent handling" requirements for the fuel show that it is "highly dangerous." Addition to Anthony/FOE Application for Contention on New Matter (February 28, 1984). But these requirements reflect the Commission's "defense-in-depth" philosophy/policy rather than an acknowledgment that this unirradiated fuel is "highly dangerous" in the way FOE perceives it. Further, we suspect that the stringent packaging and shipping requirements for the fuel assemblies would be undertaken in large measure even in the absence of government regulation in order to protect the economic value of the cargo — in much the way delicate electronic equipment is shipped.

aware of this arrangement but apparently has chosen not to avail itself of it.

In sum, the Licensing Board's decision dismissing FOE's contentions for lack of the required basis and specificity is amply justified.[17]

**B. Timeliness**

Having affirmed the Board's dismissal of FOE's Part 70 contentions for lack of basis and specificity, it is not necessary that we consider whether the contentions were late-filed. Nonetheless, we believe some comments on this matter are in order.

The Board concluded that the criteria applied to late-filed contentions — found in 10 C.F.R. § 2.714(a)(1) and discussed in *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983) — "do not apply" in the circumstances of this case. LBP-84-16, *supra,* 19 NRC at 868. It emphasized that PECo is under the Board's standing order (since 1981) to serve the Board and all parties with any material related to the operating license proceeding. In the Board's view, *Diablo Canyon,* CLI-76-1, *supra* note 6, determined that a Part 70 license is so related. Thus, PECo's failure to serve its June 1983 Part 70 application and subsequent amendments on the parties excuses FOE from having to satisfy the five late contention criteria of 10 C.F.R. § 2.714(a)(1).[18] The Board also explained why FOE had no reason to foresee that applicant would request this particular Part 70 license. LBP-84-16, *supra,* 19 NRC at 865-68.

We certainly agree with the Board's rebuke of PECo for not at least notifying the Board and parties of the filing of its Part 70 application. Even in the absence of the standing order, PECo should have done so. *See Tennessee Valley Authority* (Browns Ferry Nuclear Plant, Units 1, 2 and 3), ALAB-677, 15 NRC 1387, 1394 (1982). PECo may not agree that its Part 70 application is "related" to the operating license proceeding; but given the Commission's 1976 *Diablo Canyon* decision and *Zimmer* (in which both PECo's counsel and the Licensing Board

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[17] In its brief on appeal, FOE discusses several matters related to testimony given just last week in the ongoing operating license proceeding for Limerick. It is not apparent how any of this relates to the Part 70 contentions FOE sought to raise, particularly insofar as FOE's principal focus was the temporary outdoor storage of fuel at the site. Moreover, assuming *arguendo* that it would be proper for us to rely here on appeal on such new information, FOE makes no attempt to explain how this testimony and other cited documents would alter the Licensing Board's dismissal of its contentions. That is, FOE fails to explain a *credible* basis for any of its scenarios by which either criticality could be achieved or a radiation hazard could occur through some noncritical means.

[18] FOE first learned of the Part 70 application when staff counsel served it with copies on February 21, 1984. *See* p. 649, *supra.*

Chairman participated), PECo has no excuse for not assuming that the *Licensing Board* would find the matters related. If in doubt, of course, the more responsible course would have been to supply the information. *Cf. Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-691, 16 NRC 897, 914 (1982), *review declined,* CLI-83-2, 17 NRC 69 (1983).[19]

The Board's criticism of the staff on this score is warranted as well. *See* LBP-84-16, *supra,* 19 NRC at 867. There are a relatively small number of plants now involved in the licensing process. Each plant has an NRC project manager, who should be aware of all licensing activity concerning the plant. Further, staff counsel are supposed to be informed of the filing of all such applications. The agency has sophisticated data processing capability. There is simply no acceptable explanation for the staff's failure to apprise the Licensing Board and parties, in a more timely fashion, of PECo's Part 70 application.[20]

We disagree with the Licensing Board, however, insofar as it concludes that the 10 C.F.R. § 2.714(a)(1) criteria and *Catawba, supra,* "do not apply" in this situation. Rather, the facts discussed above strongly establish FOE's showing of "[g]ood cause . . . for failure to file on time" — the first of the five criteria in 10 C.F.R. § 2.714(a)(1). But we believe *Catawba* requires consideration and balancing of all five factors enumerated in that provision, even where a party has succeeded in making a strong showing on the good cause factor. *See* 17 NRC at 1045-46.[21] In view of our decision affirming the Licensing Board's ultimate conclusion, however, a remand to balance the five factors would serve no useful purpose.

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[19] PECo's argument that the parties are obliged to keep abreast of the public record by reviewing the files of the Public Document Room (here in Washington, D.C.) is without merit. The Commission maintains a *Local* Public Document Room (LPDR) in the vicinity of a plant site so that parties residing nearby have *reasonable access* to all filings. We have been advised by the NRC's Division of Rules and Records that PECo's Part 70 application was never filed in the LPDR for Limerick; thus, the parties did not have reasonable access to this information. PECo's further suggestion that the parties could have obtained, pursuant to the Freedom of Information Act, a document they had no reason to expect existed is similarly specious. *See* Applicant's Answer to New Contentions (March 1, 1984) at 5 n.9.

[20] The failure of FOE to have earlier notice of PECo's Part 70 application was surely responsible for the somewhat unusual procedural course of this case, in which the time for filing pleadings before both the Licensing Board and us was shortened and a temporary stay had to be entered. Questions concerning the jurisdiction of both Boards made the handling of this matter all the more complicated and time-consuming. Fortunately, the basis of the Licensing Board's decision was such that more time and neater procedures would not have altered the outcome that we affirm here. The next such Part 70 application, however, may be different. The Commission may thus find it worthwhile to establish general procedures for handling this category of cases.

[21] The other four factors are:

(ii) The availability of other means whereby the petitioner's interest will be protected.

(iii) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.

(iv) The extent to which the petitioner's interest will be represented by existing parties.

(v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding.

The Licensing Board's decision asserting jurisdiction over and dismissing FOE's Part 70 contentions is *affirmed.*[22]

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

[22] In light of our decision on the merits, we lift our temporary stay (*see* note 1, *supra*) and deny FOE's request to stay the Licensing Board's decision and issuance of the Part 70 license.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Charles Bechhoefer,** Chairman
**Dr. James C. Lamb**
**Ernest E. Hill**

**In the Matter of**

**Docket Nos. STN 50-498-OL**
**STN 50-499-OL**
**(ASLBP No. 79-421-07-OL)**

**HOUSTON LIGHTING AND POWER COMPANY,** ***et al.***
**(South Texas Project, Units 1 and 2)**

**March 14, 1984**

The Licensing Board issues a Partial Initial Decision which resolves various quality assurance/quality control issues raised by the Commission in CLI-80-32, 12 NRC 281 (1980), together with Intervenors' contentions related to those QA/QC issues. The Board also denies a motion to reopen the record. The Board rules that, subject to possible modification in later phases of the proceeding, there is currently no basis for concluding (1) that the reasonable assurance findings contemplated by 10 C.F.R. § 50.57 cannot be made, or (2) that HL&P currently lacks managerial competence or character sufficient to preclude an eventual award of operating licenses for the facility. The Board is requiring a report in Phase II of the proceeding concerning QA/QC activities performed following the assumption of duties by a new architect-engineer/construction manager and a new construction contractor.

## ATOMIC ENERGY ACT: OPERATING LICENSES

Character and competence are fundamental requirements for an operating license applicant. They are implicit in, and hence stem from the Atomic Energy Act, specifically Sections 103 and 182a, 42 U.S.C. §§ 2133(b)(2) and 2232(a).

## OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE

There is a marked distinction between the competence and character requirements for an operating license applicant. Although the factors which comprise character or competence may overlap, they nevertheless constitute separate and distinct (and cumulative) requirements.

## OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE

Issues which may bear upon management competence include: (1) whether an applicant's staff and management have sufficient technical and managerial expertise and experience (*i.e.*, demonstrated knowledge, judgment, and skill) to construct the plant properly and operate it safely, (2) whether an applicant's staff and management are organizationally structured so as to permit and encourage the unhindered application of their expertise and experience, and (3) whether an applicant's programs and procedures require the application of that expertise and experience and are consistent with goals of the Commission's regulations and the Atomic Energy Act. That third issue may also be characterized as the adequacy of an applicant's written quality assurance/quality control program(s).

## OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE

Character is, among other things, a measure of the likelihood that an applicant will apply its technical competence to effect the Commission's health and safety (or environmental) standards.

## OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE

The character of an operating license applicant is comprised of many traits relevant to the construction or operation of a nuclear plant.

Among those traits are truthfulness and candor, the manner in which the applicant has reacted to construction noncompliances or nonconformances, its assumption of responsibility for the facility under construction, and the degree to which it attempts to stay informed about the facility.

**OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE**

In evaluating an applicant's character and competence, all relevant circumstances must be considered, including reformation of character and improvement in competence.

**OPERATING LICENSE(S): MANAGERIAL CHARACTER AND COMPETENCE**

Failure of one or more individuals to demonstrate adequate competence or character does not *per se* indicate a lack of organizational competence or character (and *vice versa*). In evaluating the competence or character of an organization, such factors as the role of particular individuals in the organization, the responsibilities they exercise, the seriousness and frequency of any deficiencies attributable to them, and the steps taken by the organization when deficiencies are discovered must be balanced.

**ATOMIC ENERGY ACT: MATERIAL FALSE STATEMENT**

The presence or absence of intent, or of knowledge of falsity of a statement, is irrelevant to the technical question of whether or not a material false statement has been made. *Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC 480, 483, 486-87 (1976), *aff'd,* 571 F.2d 1289 (4th Cir. 1978). On the other hand, such intent and knowledge are pertinent to the effect of false statements on an applicant's character.

**QUALITY ASSURANCE: REQUIREMENTS (RELATIONSHIP TO REPORTS UNDER 10 C.F.R. § 50.55(e))**

The circumstance that a deficiency was properly reported under 10 C.F.R. § 50.55(e) is not relevant to whether the deficiency represented a violation of the quality assurance requirements of 10 C.F.R. Part 50, Appendix B.

QUALITY ASSURANCE: REQUIREMENTS (SURVEYING)

The quality assurance criteria of 10 C.F.R. Part 50, Appendix B, particularly Criteria II and V, apply to construction activities such as surveying.

QUALITY ASSURANCE: REQUIREMENTS

The quality assurance criteria of 10 C.F.R. Part 50, Appendix B, control implementation as well as the establishment of a QA program. A failure in implementation may constitute a violation of Appendix B.

QUALITY ASSURANCE: REQUIREMENTS (SURVEYING)

To the extent that surveying represents a construction activity rather than a test, it is not governed by 10 C.F.R. Part 50, Appendix B, Criterion XI ("Test Control").

RULES OF PRACTICE: REOPENING OF PROCEEDINGS

A motion to reopen a record must be timely and must address significant safety (or environmental) issues. Where the record of a proceeding (or at least of a major phase thereof) is closed, the information sought to be included in the record must be material and significant — *i.e.*, to have at least the potential for altering a result which might otherwise be reached. To meet this standard, the proponent must offer new and significant factual information. The "timeliness" test is subsidiary to that of materiality or significance.

## APPEARANCES

**Jack R. Newman, Esq., Maurice Axelrad, Esq., Alvin H. Gutterman, Esq.,** and **Donald J. Silverman, Esq.,** Washington, D.C., **Finis E. Cowan, Esq.,** and **Thomas B. Hudson, Jr., Esq.,** Houston, Texas, for Houston Lighting and Power Co., *et al.*, Applicants

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**William S. Jordan, III, Esq.,** Washington, D.C., **Geoffrey M. Gay, Esq.,** Forth Worth, Texas, and **Ms. Peggy Buchorn,** Brazoria, Texas, for Citizens for Equitable Utilities, Inc., Intervenor

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# PARTIAL INITIAL DECISION
(Operating License, Phase I)

# Opinion

## I. INTRODUCTION

### A. Nature of the Proceeding (Findings 1-4)

This Partial Initial Decision is the first involving the application for licenses to operate the South Texas Project, Units 1 and 2 (STP). Information respecting the public health and safety and environmental aspects of the license application was filed in May 1978 by Houston Lighting and Power Company (HL&P), the City of San Antonio, Central Power and Light Company, and the City of Austin, Texas (hereinafter referred to collectively as the Applicants). HL&P is the lead applicant with responsibility for construction and operation of the facility. A Notice of Opportunity for Hearing was published in August 1978.

The STP is located approximately 15 miles southwest of Bay City, on the west side of the Colorado River, in Matagorda County, Texas. The plant will consist of two pressurized water reactors, each with a rated output of 1,250 megawatts of electrical power.

### B. Identification of the Parties (Finding 5)

Five petitioners originally sought intervention, and two were admitted as parties: Citizens for Equitable Utilities, Inc. (CEU), and Citizens Concerned About Nuclear Power, Inc. (CCANP). In addition, the State of Texas was admitted as an interested State pursuant to 10 C.F.R. § 2.715(c). CEU subsequently withdrew from this proceeding on June 15, 1982, subject to certain conditions. *See* our Memorandum dated June 24, 1982 (unpublished). *See also* Part III.A of this Opinion, *infra.*

### C. Procedural Posture of the Case (Findings 6-12)

Eight contentions (some with multiple subparts) were admitted. Of these, CEU and CCANP jointly sponsored Contentions 1 and 2, CCANP was the sole sponsor of Contention 3, and CEU was the sole sponsor of Contentions 4 through 8. After CEU's withdrawal, CCANP sought to adopt all of CEU's contentions. By our Memorandum and Order dated October 15, 1982, LBP-82-91, 16 NRC 1364, we permitted

CCANP to adopt Contention 4 but dismissed the remainder of the contentions sponsored solely by CEU (Contentions 5, 6, 7 and 8).

On April 30, 1980, the NRC Office of Inspection and Enforcement issued I&E Report 79-19 (Staff Exh. 46, Appendix D), which identified twenty-two noncompliances in HL&P's STP construction activities. The investigation report indicated substantial deficiencies in HL&P's construction quality assurance/quality control (QA/QC) program and cast serious doubt on HL&P's ability to manage construction of the STP. Accompanying I&E Report 79-19 was a Notice of Violation and an Order to Show Cause, requiring HL&P to set forth its reasons why safety-related construction activities should not be halted. In addition, a civil penalty of $100,000 was proposed as a result of the items of noncompliance found in 79-19. By letters dated May 23, 1980, HL&P confirmed, with minor exceptions, the findings of 79-19 and paid the civil penalty of $100,000. Beginning with its filing of July 28, 1980, HL&P responded to the tasks required by the Show-Cause Order.

On May 28, 1980, CCANP and CEU filed with the Commission requests for a hearing on the Order to Show Cause. On September 22, 1980, the Commission denied those requests but agreed with our previously expressed intent to hold an early hearing on QA/QC issues. The Commission also directed us to consider the "broader ramifications" of charges relating to HL&P's "basic competence and character." CLI-80-32, 12 NRC 281, 291-92 (1980). Therefore, on December 2, 1980, we articulated six issues (A through F) addressing the Commission's concerns. We denominated the hearing on CLI-80-32 Issues A through E (and on Contentions 1 and 2, which address QA/QC deficiencies) as Phase I of the operating license proceeding. (At that time, only two phases were anticipated.) Evidentiary hearings on Phase I commenced on May 12, 1981 and extended intermittently until June 17, 1982. Reflecting the widespread interest in this proceeding, hearings were held in Bay City, Houston, and San Antonio, Texas; limited appearance statements were invited and heard in each city, as well as at a prehearing conference held in Austin, Texas.

On September 24, 1981, the Applicants informed us that they were dismissing Brown & Root (B&R), their architect-engineer and construction manager. Later we were advised that the Applicants would also replace B&R as constructor. Bechtel Power Corp. (Bechtel) assumed the duties of architect-engineer and construction manager, and Ebasco Services Inc. (Ebasco) replaced B&R as constructor. On September 28, 1981, the Applicants further informed us that a report on B&R engineering had been prepared for HL&P by Quadrex Corporation (the Quadrex Report).

At a prehearing conference on December 8, 1981, in order to accommodate these changes, we divided the hearing into three phases. *See* Fourth Prehearing Conference Order, dated December 16, 1981 (unpublished). The topics previously included in the first phase, plus certain issues arising out of the transition from B&R to Bechtel and Ebasco, continued as Phase I. The Board also admitted for Phase I adjudication four new subparts of Contention 1 (1.8(a) through (d)). Phase II will address the Quadrex Report (including its effect, if any, on determinations reached in Phase I) and Contention 4 (hurricanes).[1] It will also include the report we are directing under CLI-80-32 Issue B, *infra,* p. 697. Phase III will address CLI-80-32 Issue F (QA for operation), Contention 3 (overpressurization), and any remaining issues.[2]

Phase I is now complete.[3] Accordingly, this Partial Initial Decision addresses and resolves CLI-80-32 Issues A through E, and Intervenors' Contentions 1 and 2.[4] For reasons hereafter spelled out, and based on the entire record, we find no basis at this time for concluding (1) that the reasonable assurance findings contemplated by 10 C.F.R. § 50.57 cannot be made, or (2) that HL&P currently lacks managerial competence or character sufficient to preclude an eventual award of operating licenses for STP. These conclusions will be subject to modification, if appropriate, as a result of our consideration of Quadrex Report issues in Phase II. In addition, we are requiring that the NRC Staff (and the Applicants and other parties if they wish) report to us during the Phase II evidentiary hearings concerning safety-related construction activities (including implementation of the QA/QC program) following the assumption of duties by Bechtel and Ebasco. We also expect that, during the consideration of Issue F (QA for operation) in Phase III, the Applicants and Staff will update (as appropriate) the testimony presented with respect to Issue C dealing with HL&P's organization for operation.

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[1] On July 14, 1983, we denied CCANP's motion seeking to add a financial qualifications contention to Phase II. LBP-83-37, 18 NRC 52, *reconsideration denied,* LBP-83-49, 18 NRC 239 (1983).

[2] On October 20, 1983, CCANP filed a motion to admit a new contention concerning soil stability. Since CCANP wishes to litigate this contention in Phase III, we have deferred ruling on it until after the issuance of this Decision.

[3] On January 10, 1983, we denied a motion by CCANP to reopen the Phase I record. We deal with and deny another such motion by CCANP during the course of this Decision. *See* Opinion, Part IV, *infra.*

[4] All of the issues and contentions dealt with by this Decision are set forth in Appendix A (unpublished), as well as in our Findings with respect to the various issues or contentions.

## II. LEGAL STANDARDS FOR DETERMINING CHARACTER AND COMPETENCE

The central focus of our inquiry in this first phase of the proceeding has been the "character and competence" of HL&P to build and operate the facility.[5] In CLI-80-32, the Commission found that many of the nonconformances and related items which gave rise to the Show-Cause Order are relevant to the "basic competence and character" of HL&P, and it directed that we "look at the broader ramifications of these charges in order to determine whether, if proved, they should result in denial of the operating license application." 12 NRC at 291-92. Reflecting this direction, questions concerning HL&P's character and competence permeate Issues A, B, C and D derived from CLI-80-32. Before addressing those questions, however, we must first delineate our understanding of character and competence and the legal standards which we will employ in determining whether HL&P possesses the requisite character and competence to be authorized to operate the STP.

Because of the importance of these standards to our Decision, we asked the parties to file pre-trial briefs on those standards. All of them did so.[6] In addition, each of the parties filing proposed findings and conclusions again addressed these legal issues.[7] We have considered all of those filings, as well as other legal authority, in formulating our views as to the legal standards for determining character and competence.

All parties appear to agree that character and competence are fundamental requirements for a license applicant. The character and competence requirements are implicit in, and hence stem from, the Atomic Energy Act.[8] All parties also concede that the Commission has not pre-

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[5] Since HL&P is the lead applicant with responsibility for construction and operation of the facility, our discussion of the character or competence of the Applicants will represent findings only with respect to the character or competence of HL&P.

[6] Applicants' Memorandum of Law on Issues Concerning Competence and Character, dated May 2, 1981; CCANP Brief on "Character," dated May 5, 1981; Citizens for Equitable Utilities Prehearing Brief, dated May 6, 1981; and NRC Staff Memorandum on Standards for Evaluating Managerial Competence and Corporate Character, dated May 6, 1981.

[7] In particular, *see* Applicants' Proposed Findings of Fact and Conclusions of Law (App. FOF), dated August 6, 1982, at 291-99; CCANP Proposed Findings of Fact and Conclusions of Law (CCANP FOF), dated September 20, 1982, at 1-19; NRC Staff's Proposed Opinion (etc.) (Staff FOF), dated October 4, 1982, at 12-28; Applicants' Reply to Proposed Findings of Fact and Conclusions of Law submitted by the Other Parties (App. Reply FOF), dated October 18, 1982, at 11-12.

[8] Section 103 states that a commercial license shall be issued to applicants "who are equipped to observe and who agree to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish." 42 U.S.C. § 2133(b)(2). Section 182a adds that "[e]ach application for a license hereunder shall • • • specifically state such information as the Commission, by rule or regulation, may determine to be necessary to decide such of the technical and financial qualifications of the applicant, the character of the applicant, • • • or any other qualifications of the applicant as the Commission may deem appropriate for the license." 42 U.S.C. § 2232(a).

cisely defined, through either rule or adjudicatory decision, the exact contours of those terms. In CLI-80-32, *supra*, it provided some guidance:

> The history of the South Texas Project — at least 12 separate NRC investigations over a 2½ year period, resulting in conferences with the licensee, several prior items of non-compliance, a deviation, five immediate action letters, and [n]ow substantiated allegations of harassment, intimidation and threats directed to QA/QC personnel and apparent false statements in the FSAR — is relevant to the issue of the basic competence and character of Houston.

12 NRC at 291. Given the lack of more detailed guidance, we have found it necessary, in applying the Commission's directive, to determine: (1) whether our evaluation of HP&L's character should be divorced from our evaluation of its competence; (2) what factors are relevant to character and to competence, and what weight should such factors be accorded; (3) what consideration should be given to reformation of character or improvement in competence; and (4) the relationship of individual and organizational character or competence.

## A. The Dichotomy

The NRC Staff and the Applicants each assert that character and competence are inextricably intertwined and cannot be evaluated separately.[9] CCANP, although it does not explicitly state how it believes character and competence should be analyzed, treats the two concepts as separate and distinct.[10] As described below, that basic approach is consistent with applicable precedent, which generally has analyzed character and competence in terms of attributes that, although overlapping in some respects, are fundamentally of a different nature. We view the differences between character and competence to be more significant than their similarities and have thus treated character and competence separately in our analysis of the instant record.

The Staff itself acknowledges that instructive case law both within and outside the NRC has addressed character and competence separately.[11] Similarly, the Commission, in CLI-80-32, *supra*, read the Atomic Energy Act as drawing a marked distinction between "competence (*i.e.*, technical)" and "character qualification" of a licensee or license applicant. 12 NRC at 291. The relationship between competence

[9] NRC Staff Memorandum (May 6, 1981) at 6; Staff FOF at 12-15; Applicants' Memorandum (May 2, 1981) at 9-10. CEU took a basic approach similar to that taken by the Staff and Applicants.
[10] CCANP Brief (May 5, 1981) at 4; CCANP FOF at 12-13.
[11] Staff FOF at 15.

(adequacy of technical qualifications) and character was well explained by the Appeal Board in *Consumers Power Co.* (Midland Plant, Units 1 and 2, ALAB-106, 6 AEC 182 (1973). In addressing the adequacy of an applicant's QA/QC program, the Appeal Board stated:

> The inquiry which the board must make is not necessarily resolved by a determination of whether, in a broad sense, the applicant and its architect-engineer are "technically qualified." A demonstration that technical qualifications do exist does not necessarily provide reasonable assurance that the QA program described in the PSAR will be faithfully fulfilled. To the contrary, as important as qualifications may be, of no less significance is the matter of managerial attitude. Unless there is a willingness — indeed, desire — on the part of the responsible officials to carry it out to the letter, no program is likely to be successful.

*Id.* at 184. Thus, under that formulation, character and competence are quite different: character is, among other things, a measure of the likelihood that an applicant will apply its technical competence to effect the Commission's health and safety standards.

We recognize that the factors which comprise character or competence may overlap. For instance, whether an applicant has developed technical ability may be relevant to and indicative of both its character and its competence. But, in our opinion, even the most technically qualified applicant should be denied a license if its character is deficient — *i.e.*, if it is shown that the applicant is unlikely to apply that technical ability adequately. Similarly, no degree of character can compensate for technical incompetence. We read the *Midland* decision as providing that an applicant must demonstrate both that it is competent (*i.e.*, technically qualified) and that it has the requisite character. Moreover, we do not believe that character can be inferred from competence, or *vice versa*. Finally, to the extent that otherwise deficient character or competence can be remedied (*see* Part II.C of this Opinion, *infra*), the remedies themselves are often quite different. We therefore view character and competence as separate and distinct (and cumulative) requirements and treat them accordingly.

### B. The Relevant Factors

The Commission's regulations do not amplify the competence requirement and make no explicit mention of character.[12] Therefore, we (as well as all of the parties) have looked to the Commission's guidance in

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[12] 10 C.F.R. §§ 50.40(b) and 50.57(a)(4) merely repeat the competence requirement, that "the applicant is technically • • • qualified" to engage in the activities for which a license is sought.

CLI-80-32 and to precedent, in order to determine what factors are relevant to character and competence. Although other decisions have addressed character and competence, the decisions have not been consistent in their terminology.[13] Accordingly, we include in our discussion of the relevant factors a definition of each term as we apply it.

### *1. Competence*

In the absence of a regulatory definition of "competence," we use the plain meaning of the term. Competence is "the quality or state of being functionally adequate or of having sufficient knowledge, judgment, skill or strength (as for a particular duty or in a particular respect)."[14] We apply this definition in accordance with the statutory mandate of Section 103 of the Atomic Energy Act, 42 U.S.C. § 2133, that applicants be "equipped to observe * * * [the Commission's] safety standards."[15]

In interpreting this statutory mandate, the Commission has pointed to a number of issues which may bear upon "management competence." In particular, it has referred to the sufficiency of staffing and resources, the quality of management, and the adequacy of organization of a utility. It has indicated that prior performance by a utility (including a comparison of its performance with industry-wide statistics) may also raise competence questions. *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit No. 1), CLI-80-5, 11 NRC 408 (1980). The Commission emphasized, however, that it has not established definitive standards by which to judge managerial competence but only has identified questions it deems pertinent to such an inquiry. *Id.* at 410.[16]

A similar scope of inquiry as to competence seems to have been envisaged by the Commission in CLI-80-32, *supra*. In particular, the Commission stressed the relevance to HL&P's competence of the history of past violations; specifically, whether those violations suggested an

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[13] In *Midland*, ALAB-106, *supra*, 6 AEC at 184, the Appeal Board used "technical qualifications" in the same sense as we use "competence," and it used "managerial attitude" as we use "character"; in *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), LBP-77-68, 6 NRC 1127, 1151 (1977), the Licensing Board used "commitment" in the same sense as we have used "character"; and in *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4), LBP-79-19, 10 NRC 37 (1979), the Licensing Board addressed both concepts under the label "capability."

[14] Webster's Third New International Dictionary 463 (unabridged ed. 1976).

[15] Competence would also extend to an applicant's ability to satisfy environmental requirements. But since CLI-80-32 (as well as Intervenors' Contentions 1 and 2) concern only safety issues, we are here considering competence only with respect to safety requirements.

[16] *See also Shearon Harris*, LBP-79-19, note 13, *supra*, 10 NRC at 56-94; *Mississippi Power & Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), LBP-74-64, 8 AEC 339, *aff'd*, ALAB-232, 8 AEC 635 (1974).

"abdication of responsibility or abdication of knowledge" and hence an organizational, programmatic or personnel deficiency. 12 NRC at 291.

To a great extent, these considerations are amenable to objective assessment. NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources" (1980), provides guidelines for an applicant's staff and management organization and experience.[17] As the Staff observes, NUREG-0731 incorporates the approach adopted by the Commission in *Three Mile Island,* CLI-80-5, *supra.*[18]

In sum, in the context of the issues before us, the appropriate inquiry is: (1) whether an applicant's staff and management have sufficient technical and managerial expertise and experience (*i.e.*, demonstrated knowledge, judgment, and skill) to construct the plant properly and operate it safely, (2) whether an applicant's staff and management are organizationally structured so as to permit and encourage the unhindered application of their expertise and experience, and (3) whether an applicant's programs and procedures require the application of that expertise and experience and are consistent with goals of the Commission's regulations and the Atomic Energy Act.

### *2. Character*

The parties all appear to recognize that the concept of character is difficult to define. *See Hall v. Geiger-Jones Co.,* 242 U.S. 539, 553 (1917). CCANP asserts that, in the absence of a regulatory or statutory direction, the term should be given its commonly understood definition (*citing Mester v. United States,* 70 F. Supp. 118, 122 (E.D.N.Y. 1947)). We agree.

Character is defined as "a composite of good moral qualities typically of moral excellence and firmness blended with resolution, self-discipline, high ethics, force, and judgment."[19] Obviously, the term is less specific than is "competence" and calls for a more subjective determination. Character comprises many traits. No trait should be considered, however, unless it is relevant to the construction or operation of a nuclear plant.[20] Therefore, a trait should only be considered if it evinces a willingness and propensity, or lack thereof, on the part of an applicant to ob-

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[17] *See also* NUREG/CR-1280, "Power Plant Staffing" (1980); NUREG/CR-1656, "Utility Management and Technical Resources" (1980).

[18] Staff FOF at 27.

[19] Webster's Third New International Dictionary 376 (unabridged ed. 1976).

[20] *See Schware v. Board of Bar Examiners of New Mexico,* 353 U.S. 232, 239 (1957).

serve the Commission's health and safety standards.[21] Indeed, our ultimate finding of fact must determine, *inter alia*, whether there is reasonable assurance that the Applicants will (*i.e.*, have the character to) observe the Commission's health and safety standards.[22]

In CLI-80-32, the Commission indicated that responsibility was a necessary trait, the abdication of which could result in denial of a license application. An applicant must retain responsibility for construction or operation of a nuclear power plant and must keep itself fully informed. In CLI-80-32, the Commission also stressed that truthfulness is of particular concern.[23] And from earlier decisions, it is clear that truthfulness contemplates not only false or misleading statements but the completeness or comprehensiveness of information provided by an applicant to the Commission. *Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC 480 (1976), *aff'd*, 571 F.2d 1289 (4th Cir. 1978).

As the Staff indicates, the trait of truthfulness or candor is particularly important given the regulatory regime relied upon by NRC.[24] The Commission has forcefully stated that

> In order to fulfill its regulatory obligations, NRC is dependent upon all of its licensees for accurate and timely information. Since licensees are directly in control of plant design, construction, operation, and maintenance, they are the first line of defense to ensure the safety of the public. NRC's role is one primarily of review and audit of licensee activities, recognizing that limited resources preclude 100 percent inspection.
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> As the Commission has stated in the past:
>
> > Our inspection system is not designed to and cannot assume such tasks [to provide full inspection of construction activities]. Rather, we require that licensees themselves develop and implement reliable quality assurance programs which can assume the major burden of inspection. *Consumers Power Company* (Midland Plant, Units 1 and 2), CLI-74-3, 7 AEC 7, 11 (1974)
> >
> > We require instead a regime in which applicants and licensees have every incentive to scrutinize their internal procedures to be as sure as they possibly can that all submissions to this Commission are accurate.

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[21] As in the case of competence, character also is reflected by an applicant's willingness or propensity to observe environmental standards. No such issue has been raised in Phase I of this proceeding.

[22] 10 C.F.R. § 50.57(a)(3).

[23] The Commission stated (12 NRC at 291 n.4):

> [T]he Commission cannot ignore false statements in documents submitted to it. Congress has specifically provided that licenses may be revoked for "material false statements," *see* section 186a of the Atomic Energy Act [42 U.S.C. § 2236(a)], and we have no doubt that initial license applications or renewal applications may also be denied on this ground, certainly if the falsehoods were intentional, *FCC v. WOKO, Inc.*, 329 U.S. 223 (1946), and perhaps even if they were made only with disregard for the truth. *Leflore Broadcasting Company v. FCC*, [636 F.2d 454 (D.C. Cir. 1980)]; *Virginia Electric and Power Co. v. NRC*, 571 F.2d 1289 (4th Cir. 1978).

[24] Staff FOF at 19.

*Petition for Emergency and Remedial Action,* CLI-78-6, 7 NRC 400, 418 (1978).

More recently, the Commission again stressed the importance to character of an applicant's truthfulness, particularly as that trait bears on information provided to Licensing Boards in adjudicatory proceedings such as this one. The Commission stated:

> A deliberate false statement or withholding of material information would warrant the imposition of a severe sanction. The time and resources committed to an adjudicatory probing of the facts of this case are evidence of our concern over allegations of this sort. Not only are material false statements and omissions punishable under Sections 234 and 186 of the Atomic Energy Act, but deliberate planning for such statements or concerns on the part of applicants or licensees would be evidence of bad character that could warrant adverse licensing action even where those plans are not carried to fruition. Moreover, we want to warn parties and their attorneys that when they engage in conduct which skirts close to the line of improper conduct, they are running a grave risk of serious sanction if they cross that line.

*Consumers Power Co.* (Midland Plant, Units 1 and 2), CLI-83-2, 17 NRC 69, 70 (1983).

There are, of course, many other traits that are pertinent, and CCANP has suggested six generalized ones against which HL&P's character should be evaluated: foresight, judgment, perception, resolve, integrity (including trustworthiness, reliability and honesty), and values (CCANP FOF at 3-5, 12). These traits are, of course, generally relevant to character. Indeed, they closely track the definition of character which we have found appropriate. But, in our view, they are so broad and ill-defined that analyzing them would give little assistance in providing answers to the questions raised by CLI-80-32.

In that connection, we note that, in applying the facts of record to determine whether HL&P possesses the requisite character, CCANP has utilized many of the same incidents or events as examples of several of the traits it enumerates. The abilities of one of HL&P's managerial personnel, for instance, are said to bear upon four of those traits: HL&P's foresight (CCANP FOF at 46), its judgment (CCANP FOF at 50-52), its perception (CCANP FOF at 66) and its values (CCANP FOF at 120-21). HL&P's utilization of this employee may be relevant to its character (as well as to its competence), but it contributes little to a meaningful analysis to find that such utilization is pertinent to one or several subsets of character. Moreover, we do not believe it is practical or necessary to attempt to enumerate all relevant traits. Were we to undertake such an exercise, we feel it would serve only to replace one label, "character," with many; it would leave unresolved the factors determinative of each trait. What is necessary is a nexus of a particular trait to particular performance standards contemplated by the Atomic Energy Act or NEPA and NRC's implementing regulations and guides.

Therefore, we adjudge HL&P's character by consideration of its past and present performance, and consider those traits, both positive and negative, that are naturally inferred therefrom. We find this approach consistent with that taken by other Boards. In particular, we scrutinize, *inter alia,* HL&P's record of compliance with the NRC regulations;[25] its response to noncompliances;[26] and, most importantly, whether HL&P made material false statements or omissions and whether it addressed questions propounded by the Staff, the parties and us with candor. This approach is also consistent with that utilized by other agencies with regulatory schemes comparable to that used by NRC. *See, e.g., FCC v. WOKO, supra.* The traits we infer from this scrutiny are, in effect, our conclusions; and the composite of these traits constitutes HL&P's character and forms the basis for our ultimate finding of fact.

## C. Reformation of Character and Improvement in Competence

Early in this proceeding, the Intervenors asserted that we should limit the first phase of this proceeding to whether HL&P's past conduct and actions in themselves indicated a lack of character or competence sufficient, without more, to warrant denial of the operating license application (in effect, the matters encompassed by Issue A). They would have thus excluded all evidence of corrective measures taken by the Applicants (*i.e.,* matters relevant to Issues B, C, D and E). We rejected that position in our Second Prehearing Conference Order (unpublished), dated December 2, 1980 (at 4-5) and denied reconsideration in our Third Prehearing Conference Order (unpublished), dated April 1, 1981 (at 8-11).[27] We reasoned that, although CLI-80-32 contemplated a determination whether past practices, in themselves, should result in a

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[25] *See, e.g., Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4), LBP-79-19, 10 NRC 37 (1979), *aff'd and modified,* ALAB-557, 11 NRC 18, CLI-80-12, 11 NRC 514 (1980). *See also Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-128, 6 AEC 399, 407 (1973); *Midland,* ALAB-106, *supra,* 6 AEC at 184 (1973); *North Anna,* LBP-77-68, note 13, *supra*; *Duquesne Light Co.* (Beaver Valley Power Station, Unit 1), LBP-76-3, 3 NRC 44 (1976); *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), LBP-73-35, 6 AEC 861 (1973), *modified on other grounds,* ALAB-226, 8 AEC 381 (1974).

[26] *See, e.g., Shearon Harris,* LBP-79-19, note 13, *supra,* 10 NRC at 51; *North Anna,* LBP-77-68, note 13, *supra*; *Consolidated Edison Co. of New York* (Indian Point Station, Unit 2), LBP-73-33, 6 AEC 751, 756 (1973), *aff'd,* ALAB-188, 7 AEC 323, 336 (1974); *Beaver Valley,* LBP-76-3, note 25, *supra,* 3 NRC at 50-51; *Zion,* LBP-73-35, note 25, *supra,* 6 AEC at 892-93, 898-99; *McGuire,* ALAB-128, note 25, *supra,* 6 AEC at 407.

[27] The Appeal Board declined requests by CCANP and CEU for interlocutory review of the April 1, 1981 Order. ALAB-637, 13 NRC 367 (1981).

denial of the operating license application, the Commission also contemplated that we explore the totality of the Applicants' performance, including matters which may mitigate the significance of adverse findings concerning prior practices.

In its proposed findings, CCANP has largely confined its factual findings to Issue A and, in essence, has ignored the Applicants' corrective actions. Although it concedes that competence can be acquired and improved by judicious hiring, it asserts that an applicant's character is far less mutable than its competence and is perhaps immutable.[28] It urges that past behavior is a true indication of present character, or at least may cast sufficient doubt on character to prevent a favorable predictive finding.

We here reaffirm our earlier rulings as to the potential importance of corrective actions or reformation to both competence and character. We will examine HL&P's competence in order to determine if there is reasonable assurance that it *can* observe the Commission's health and safety standards. Therefore, we examine its present ability. Past incompetence is relevant, of course, to the extent it may be indicative of present incompetence. Thus, if HL&P has improved its competence, it is that improved state that is determinative.

CCANP's position with respect to character is too rigid. As we previously held, both the Atomic Energy Act and CLI-80-32 contemplate that we take into account all relevant circumstances in determining character. A change in corporate management can change an applicant's character, as can education and experience. Moreover, our role in this proceeding is not to punish an applicant for past infractions.[29] Our findings (and our authority) are limited to those standards specified by the regulations, in particular 10 C.F.R. §§ 50.40, 50.57, and 51.53, and if an applicant, whose character may have been unsatisfactory in the past, demonstrates a reformed and adequate present character, then we may find that there is reasonable assurance that it will observe the Commission's health and safety standards.

We would agree with CCANP, however, that there may be *some* character defects that are so serious that they are in fact uncorrectable, at least in the absence of a "radical change in the control of [the]

28 CCANP Brief (May 5, 1981) at 4; CCANP FOF at 10-13.

29 We make no judgment here as to the appropriateness of a licensing board's denying a license application because of material false statements made directly to that Board. In such a case, the denial might be necessary to preserve the integrity of the hearing process, and imposition of such a sanction might fall under the general grant of power to presiding officers, 10 C.F.R. § 2.718. We view this issue, however, as one totally separated from our character and competence determination, and as one not raised in this proceeding.

corporation."[30] One of these defects might be evidenced by an intentional lack of truthfulness or candor condoned by management. As we have observed, the Commission in CLI-80-32 emphasized the importance of truthfulness and candor, and it explicitly pointed out that a lack of truthfulness or candor could prove disqualifying. CLI-80-32, *supra*, 12 NRC at 291 nn.4, 5. Further, the Commission cited cases suggesting that willful misrepresentations to the Commission, or representations made with disregard for their truth, could be grounds, without more, for license denial.[31]

Whether such a character defect can be attributed to HL&P, and whether such a defect, if proved, has been, or can be, reformed, are questions of fact not of law. Accordingly, we conclude that evidence of reformation of character is relevant to and may be determinative of our ultimate finding of fact.

**D. Individual vs. Organizational Competence or Character**

We might note, as a general observation, that in ascertaining whether the Applicants possess the requisite competence or character, we have had some difficulty in drawing a line between the competence or character of particular individuals and the competence or character of HL&P. As CCANP observes, all organizations must carry on their activities through individuals (CCANP FOF, ¶¶ 1.2, 4.1). It is clear to us, however, that the failure of one or more individuals to demonstrate adequate competence or character does not *per se* indicate a lack of organizational competence or character (and *vice versa*). *See, e.g.*, Tr. 9511 (Taylor). For example, if an individual employee were found to lack competence or to have demonstrated a character defect and were removed from a project, the organization would not *per se* be deemed to lack competence or character — indeed, it might then be viewed as possessing either or both of those qualities. Furthermore, particularly with respect to character, only a limited group of corporate employees may truly be regarded as exercising a sufficient degree of responsibility so as to be deemed to affect an organization's character.

In evaluating the competence or character of an organization, we must therefore evaluate such factors as the role of particular individuals in the organization, the responsibilities which they exercise, the seriousness and frequency of any deficiencies attributable to them, and the

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30 CCANP Brief (May 5, 1981) at 4.

31 FCC v. WOKO, note 23, *supra; Leflore Broadcasting Co. v. FCC*, note 23, *supra; see also Hamlin Testing Laboratories, Inc.*, 2 AEC 423, 428-29 (1964).

steps taken by the organization when deficiencies are discovered. Our final judgment as to HL&P's organizational competence or character must balance all of these factors.

## III. OPINION ON INDIVIDUAL ISSUES

### A. Introduction

The record of this proceeding consists entirely of testimony (and documentary evidence) sponsored by the Applicants or NRC Staff, documentary evidence introduced by CEU or CCANP through the Applicants' or Staff's witnesses, cross-examination by all parties, and examination by the Board of all witnesses (with the exception of the testimony of one Staff panel, which was entered into the record by stipulation). The Intervenors presented no witnesses of their own.[32] Proposed findings and conclusions were submitted by the Applicants, the NRC Staff, and CCANP.[33]

Even though CEU and CCANP were joint sponsors of certain contentions, their positions on the ultimate issues considered in this proceeding were quite disparate. CEU consistently emphasized that it was not trying to stop construction or operation of the STP but only was seeking to assure the safe and expeditious completion of the project (Tr. 782). It stressed that, although license denial was a remedy which might be necessary, the more appropriate remedy might be the dismissal of Brown & Root (Tr. 786).[34] Consistent with that position, it withdrew from the proceeding (subject to certain conditions designed to keep it informed concerning the adequacy of HL&P's QA/QC program and practices) shortly after the replacement of B&R as design engineer, construction manager and constructor.

On the other hand, CCANP consistently has taken the position that the operating license applications should be denied. It asserted that at

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[32] Prior to the withdrawal of B&R, CEU had proposed to present an expert witness concerning project organization (his direct testimony had been prefiled), and both CEU and CCANP proposed to present factual witnesses concerning B&R activities. After the announcement of B&R's replacement, CEU and CCANP declined to present any such witnesses.

[33] App. FOF, dated August 6, 1982; CCANP FOF, dated September 20, 1982; Staff FOF, dated October 4, 1982; App. Reply FOF, dated October 18, 1982. CEU withdrew prior to the conclusion of the Phase I evidentiary hearings and hence did not offer proposed findings and conclusions.

On January 14, 1984, CCANP submitted a "supplement" to its proposed findings which directed our attention to a recent ruling of another Licensing Board. The Applicants responded on January 25, 1984. We appreciate parties' calling our attention to new legal rulings which, they believe, bear significantly on not-yet-decided issues before us. (The Staff similarly did so in conjunction with CCANP's motion to reopen the Phase I record (*see* p. 715, *infra*)).

[34] To the same effect, *see also* CEU Prehearing Brief (May 6, 1981) at 4-5.

least the character, if not the competence, of HL&P should be adjudged by past acts alone; and that, given those acts, HL&P could not be trusted to operate the plant safely (Tr. 791; CCANP FOF, ¶¶ 1.33, 2.1). Reflecting that point of view, CCANP's proposed findings are for the most part limited to CLI-80-32 Issue A and ignore discussing in any detail the corrective actions which HL&P put into effect to remedy certain of its past deficiencies.

As discussed in Part II.C of this Opinion, *supra,* we disagree as a matter of law with CCANP's position that certain character traits are immutable and cannot be improved or corrected. In our view, whether or not a character trait which is deficient can be or has been reformed is a factual question. Even more so, whether or not a deficiency in competence can be or has been corrected is also a factual question. CCANP's failure to include in its proposed findings any detailed discussion of corrective actions has therefore left a gap in the material by which we must evaluate CCANP's claims.

As we spell out in more detail below, we disagree as a factual matter with CCANP's position that, based on the entire record, the Applicants at the present time lack either competence or character to a degree which would warrant license denial. This view is not based solely on our resolution of Issue A. Indeed, in certain respects, we have found that, prior to the issuance of the Show-Cause Order, HL&P's managerial competence was questionable. In addition, however, our ultimate conclusion takes into account, to a substantial degree, the corrective actions instituted by the Applicants to ameliorate their earlier deficiencies — not the least of which is the replacement of B&R with Bechtel and Ebasco. CCANP's failure to file detailed proposed findings on the corrective actions made our task more difficult; from the overall conclusions reached by CCANP, we presume it finds the corrective actions to be ineffective or inadequate, but we are left in the dark as to its reasons.

Although they would not have us base any of our conclusions thereon, the Applicants suggest that we find CCANP in default with respect to its failure to file proposed findings on certain issues — particularly some of its own contentions (App. Reply FOF at 8). We decline to take this course. In the first place, with a view to the Intervenor's limited resources, we advised CCANP that it could focus in its findings on those issues it considered most significant (Tr. 10,656-57; Memorandum and Order dated August 19, 1982 (unpublished)). Its election to forego filing detailed findings on certain issues on which it had conducted cross-examination was not inconsistent with that advice. Moreover, we never explicitly directed

CCANP to file any proposed findings — there never was any question of its desire to do so. Despite the difficulties which its filing presents, we do not wish to penalize CCANP for attempting to follow our advice.[35]

More important, however, we find it desirable for our Decision to reflect a coherent and comprehensive picture of HL&P's activities and to resolve on their merits the issues raised by CLI-80-32 (with respect to which the Intervenors' contentions are relevant). Furthermore, the Applicants' corrective actions bear on the Intervenors' contentions, as well as on the CLI-80-32 issues. For that reason, we have reviewed with great care the record on all issues, including those on which CCANP failed to submit detailed findings. We are confident that our judgments take into account all the diverse points of view bearing on matters of safety significance which the record reflects.

We turn now to the issues and contentions dealt with by this record.

## B. CLI-80-32 Issues

### *1. Issue A: HL&P's Managerial Character and Competence (Findings 13-187)*

Issue A questions whether HL&P's record of compliance with NRC requirements is so inadequate that we should determine that HL&P does not have the necessary managerial competence or character to be granted licenses to operate the STP. This is the Issue upon which CCANP has primarily focused its attention in its proposed findings. The Issue is derived from the Commission's instructions in CLI-80-32 and explicitly excludes from consideration the effectiveness of any remedial steps taken by HL&P. (Those steps are considered separately under Issue B.)

In Part II of this Opinion, *supra,* we reviewed the applicable standards for determining the competence and character of a license applicant. Under this Issue, we are applying these standards to HL&P's record of compliance, particularly in four specified areas: (1) the material false statements alleged in the Order to Show Cause; (2) the instances of non-compliance set forth in the Notice of Violation and Order to Show Cause; (3) HL&P's alleged abdication of responsibility to B&R; and (4) HL&P's alleged failure to keep knowledgeable about construction activities. The composite of our conclusions as to competence and character, respectively, in the various areas will comprise our general

[35] *Cf. Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-709, 17 NRC 17 (1983).

cribed in Part II (pp. 670-71, *supra*), we will treat HL&P's character separately from its competence. Given CCANP's emphasis on HL&P's character, we will turn first to that subject.

*a. HL&P's Character*

In our earlier discussion of the definition of character, we pointed out that it comprises many traits and that, in evaluating the character of an NRC license applicant, the relevant traits are those which evince a willingness and propensity, or lack thereof, to observe NRC regulatory standards. In the present proceeding, the most significant character traits for us to evaluate are HL&P's truthfulness and candor, the manner in which it reacted to the noncompliances or nonconformances which occurred, its responsibility, and the degree to which it attempted to stay informed about STP.

(i) We turn first to what in our view is the most important of these character traits in the context of this proceeding — *i.e.*, HL&P's truthfulness and candor. Investigation Report 79-19, and the ensuing Order to Show Cause, raised questions about HL&P's truthfulness and candor on the basis of alleged false statements in the FSAR. In CLI-80-32, the Commission indicated that statements in the FSAR, if false, would bear directly on HL&P's character (CLI-80-32, *supra*, 12 NRC at 291 n.4).

As our findings indicate, certain statements in the FSAR relating to construction techniques and tests for backfill did not in fact accurately reflect the construction and testing carried out by HL&P through its contractor, B&R. Those FSAR statements, however, were for the most part not inaccurate when written. As the Applicants and Staff have asserted, the lack of conformance with FSAR requirements should be viewed as nonconformances with specified procedures rather than as material false statements. In the limited circumstance where nonconforming performance had in fact occurred prior to the submission of the FSAR, HL&P had not become aware of the discrepancy until long after such submission (Finding 25).

In its proposed findings, the Staff asserts that there was "no intent" by HL&P to file false statements with the Commission (Staff FOF, ¶ 65). Although we agree (Finding 33), and although we are satisfied with the Applicants' explanation that HL&P had no knowledge of the deviating construction practices at the time the relevant portions of the FSAR were submitted (Finding 25), we wish to point out that the presence or absence of intent — or, indeed, of knowledge by HL&P of falsity — is irrelevant to the technical question of whether or not a material false statement has been made. *North Anna*, CLI-76-22, *supra*, 4 NRC at

483, 486-87. On the other hand, intent and knowledge are pertinent to the question of HL&P's character which is before us. Since we are not presiding over an enforcement proceeding in which HL&P is alleged to have made material false statements but, rather, are considering alleged material false statements only in the context of HL&P's character, we find it appropriate to consider intent to falsify and knowledge of falsity as ingredients in our character determination. Given the lack of intent to submit false FSAR statements, and HL&P's lack of knowledge that certain statements were in fact inaccurate, we find that the statements in question do not reflect adversely on HL&P's character.

In considering the honesty and candor of HL&P, we have not limited our inquiry to the alleged false statements in the FSAR but have also inquired into HL&P's record for being open and candid with the NRC Staff. We were most impressed by the testimony of various Staff witnesses who had interacted with HL&P and who regarded the company as exemplary in its practice of keeping the Staff fully informed on topics of interest. H. Shannon Phillips, the Staff's Resident Inspector at STP during a substantial portion of time covered by the Phase I record (from 1979 until 1982), expressed great confidence in HL&P's candor and truthfulness — *i.e.,*

> Management was not deceptive in any way, form or fashion during the [79-19] inspection or after the inspection [Tr. 9854].
>
> [HL&P's] record of identifying and reporting construction deficiencies * * * was open and honest, and probably was better than any other utility that I've been at. * * * They got us the information, even if it was going to be detrimental to them [Tr. 9855].
>
> They were cooperative, probably the most open licensee that I've ever dealt with [Tr. 9516].

Further, various Staff witnesses noted that HL&P was generally responsive to Staff inquiries and anxious to keep the Staff informed about the project (Findings 58, 61, 164, 168, 192).

Similarly, we were also impressed by the willingness and desire expressed by HL&P managerial witnesses to communicate with NRC about project developments. *See, e.g.,* Finding 162. In similar fashion, HL&P's willingness to have a CEU representative participate in an annual independent audit of the STP QA/QC program is also representative of HL&P's openness and candor. Not only did it lead to CEU's withdrawal from the proceeding but, in addition, enhances our view of HL&P's character.[36]

CCANP has dealt with the truthfulness and candor aspects of character under the heading of "Integrity" (CCANP FOF, ¶ 7.0). It focuses almost entirely upon the study of alternative forms of QA/QC organization which HL&P undertook (through Bechtel, its consultant) in response to Item 1 of the Show-Cause Order. That Order required a review by an experienced, independent management consultant of the advantages and disadvantages of various alternative forms of QA/QC organizational structure including, as a minimum, five specified forms. With regard to that study, CCANP accuses HL&P of a "deliberate attempt to deceive the Commission" (CCANP FOF, ¶ 7.3.19, at 111). The gist of CCANP's accusation is that HL&P witnesses represented that no alternative offered significant advantages over the five which HL&P was directed to study in the Show-Cause Order whereas, in fact, a sixth alternative was preferable and certain others were not even reviewed (CCANP FOF, ¶¶ 7.3.2, 7.3.11, 7.3.12). Further, CCANP claims that HL&P's contract with Bechtel unreasonably restricted Bechtel's study, in effect precluded a recommendation in favor of certain options specified in the Show-Cause Order (as well as certain other options) and assured that the conclusion of the study would favor the form of organization already in place at STP (CCANP FOF, ¶¶ 7.3.3-7.3.16).

In our view, these claims are not well founded. They are based on statements taken out of context and, in fact, amount to a distortion of the record when viewed as a whole. We agree with the Applicants' analysis of these claims, appearing in ¶¶ 57-66 of their reply findings. We need stress only that the study in question analyzed the five forms suggested in the Show-Cause Order and, in addition, a sixth form which amounted to a variant of one of the five — and which, as CCANP concedes, HL&P adopted in part (CCANP FOF, ¶ 7.3.2). Mr. John M. Amaral, Manager of Quality Assurance of Bechtel Power Corporation, who headed the study, confirmed that Bechtel was not constrained in the manner in which it conducted its study. HL&P explicitly indicated that it gave Bechtel a "blank check" in performing that study. Finding 201. The Staff accepted the study as fulfilling the Show-Cause Order requirement. Moreover, we were most impressed with the knowledge

[36] The terms upon which CEU withdrew from the proceeding are set forth in an exchange of correspondence between HL&P and CEU, which was provided the Board by the Applicants' letter of June 14, 1982. We approved CEU's withdrawal, subject to the agreed-upon conditions, on June 15, 1982. Tr. 10,384; Memorandum dated June 24, 1982. We commend both CEU and the Applicants for the responsible manner in which they settled their differences.

and forthrightness of Mr. Amaral. Nothing in his testimony, or in that of other HL&P witnesses who addressed the study, gives us any cause to believe that HL&P was not being honest and forthright in its testimony on the study.[37]

Apart from its claims concerning the Bechtel study, CCANP has advanced two other grounds for questioning HL&P's truthfulness and candor. CCANP first claims that HL&P failed to meet a commitment made in response to the Notice of Violation (CCANP FOF, ¶ 7.3.17). We were hampered in resolving this claim by CCANP's failure to have raised it during the evidentiary hearings. The evidence cited by CCANP (Staff Exh. 64 (I&E Report 80-18, at 4, Item A.19)) indicates that HL&P failed to meet a deadline for taking certain actions with respect to HL&P's and B&R's audit programs, but we fail to see how it even suggests that there was any intent by HL&P to deceive NRC. Certainly the inspector responsible for I&E Report 80-18 did not perceive any. The statement cited by CCANP as being untruthful is HL&P's representation in its response to the Show-Cause Order that HL&P and B&R had substantially revised and improved their audit programs (Staff Exh. 48 (Licensee's Response to Order to Show Cause) at 9-2). HL&P's representation is not inconsistent with the Staff's conclusion in I&E Report 80-18 that progress had been made (Staff Exh. 64 (I&E Report 80-18, at 5)), although certain details of that improvement set forth by the Applicants in the Show-Cause Order response may not have been completely accurate. Given CCANP's failure to raise the claim at a time when witnesses could have addressed it, we decline to consider it as affecting HL&P's character.

The other claim by CCANP regarding HL&P's truthfulness and candor consists of alleged inconsistencies in Mr. Don D. Jordan's testimony concerning reasons for assigning Mr. George W. Oprea full-time to the STP (CCANP FOF, ¶ 7.3.18). We do not regard the statements as necessarily inconsistent but only as elaborations of earlier statements.

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[37] We need not dwell long on the merits of the study or of any particular form of organization. The testimony of both the Applicants and Staff indicated that no one organizational form is *per se* superior or preferable to several of the other forms which were analyzed. A number of forms could be successful in particular circumstances. Tr. 1911-12, 1930-31 (Amaral); Tr. 9528-29 (Crossman); Tr. 9943-45 (Hayes, Shewmaker, Phillips). Furthermore, as a result of the shift from one contractor (B&R) to two (Bechtel and Ebasco), the form of organization currently in effect (*see* Issue D) is vastly different from the form recommended by the Bechtel study, which was a variant of the form in effect prior to the Show-Cause Order. The current form bears some resemblance to that recommended in the prepared testimony of a CEU expert witness (Prepared testimony of Richard H. Hubbard). (CEU did not in fact present Mr. Hubbard as a witness. *See* note 32, *supra*.) Finally, CCANP has advanced no objection to the current form of organization. We regard the questions raised by CCANP to the form of organization to be moot for all practical purposes.

In sum, we find no basis for determining that HL&P was anything other than open and frank with the NRC Staff and this Board. We regard the evidence on this question as enhancing HL&P's character and not as detracting from it in any way. The only caveat is the matter concerning HL&P's promptness in forwarding the Quadrex Report to the Staff and this Board — a matter which is to be considered in Phase II. Our findings and conclusions concerning HL&P's truthfulness and candor are, of course, subject to modification (if necessary) as a result of the Phase II hearings.

(ii) The next indicator of character which we find pertinent is reflected by the manner in which HL&P responded to the many noncompliances or nonconformances (including infractions, deficiencies and deviations) which occurred prior to, and to some extent resulted in, the Order to Show Cause. In terms of a character trait, the manner in which HL&P responded to noncompliances or nonconformances may be depicted as the willingness or desire of corporate officials to carry out a QA program "to the letter." *Midland,* ALAB-106, *supra,* 6 AEC at 184.

The evidence on this question is not uniform. It is clear to us that HL&P, when considered as an entity, demonstrated a strong willingness and desire to carry out a successful QA program. But it is just as clear to us that, at least during the period of time prior to the Show-Cause Order, HL&P was not entirely successful in translating its desires to reality. There was never a complete breakdown of the QA program (Findings 86, 140, 155). But in many areas, noncompliances or nonconformances up to the severity level of infractions (*see* Finding 49, *infra*) kept recurring, despite the efforts of many HL&P and B&R officials to upgrade project performance. Reports of incidents of harassment of QC inspectors by construction personnel, and other indications of poor morale of QC inspectors, kept surfacing, despite extensive efforts by HL&P and B&R officials to deal with those incidents and to upgrade morale (Findings 66, 74-75, 223, 381-398). Moreover, the Staff's SALP report for the period July 1, 1980-June 30, 1981 reflected continuing difficulties by HL&P in responding to various nonconforming conditions (Finding 223).

We will recount just a few of HL&P's efforts which bear on its corporate character. As we have demonstrated, it has been open and aboveboard in its relationship with NRC. It has promptly and adequately reported nonconforming conditions, even conditions which have a potential for being nonconforming but have not as yet been so demonstrated. It has acted promptly in trying to correct nonconforming conditions. It has frequently discussed with NRC its proposals for correcting nonconforming conditions prior to its putting those proposals into effect. During NRC's 79-19 Investigation, after early preliminary reports of numerous nonconformances in many areas, HL&P began corrective actions well before the NRC had completed its investigation and issued its report.

HL&P also initiated a number of procedural measures to counteract the incidents of harassment of QA/QC personnel. Although incidents of harassment were not unique to the STP, their seriousness caused HL&P to go beyond what most utilities have in place and to require B&R to adopt a written procedure for resolving disputes of this sort (Finding 393). Furthermore, HL&P never limited the resources available to the QA/QC program to less than what QA/QC officials claimed they needed. In fact, HL&P's and B&R's QA/QC program utilized considerably more personnel than the average program or — indeed — what a well-run program should have utilized (Findings 118, 141).

Finally, and most significantly, HL&P took the important step of replacing its less-experienced architect-engineer-constructor with two considerably more-experienced organizations. HL&P did not take this action for the express purpose of upgrading its QA/QC program. But, in our view, that action is likely to have that effect. In the words of the Staff, the action represents "the most extreme corrective action possible" (Staff Exh. 131 (I&E Rept. 81-36, at 7). One Staff witness regarded it as "a testimony to [HL&P's] character" (Finding 125).

As the Commission in CLI-80-32, and as the Appeal Board in ALAB-106 (*Midland*), have recognized, a history of nonconforming conditions may reflect a lack of character of a license applicant. CCANP takes this position particularly with respect to the incidents of harassment (CCANP FOF, ¶¶ 6.21-6.28). Here, however, the record shows that the history of nonconforming or noncomplying conditions (including the incidents of harassment) was caused not by a lack of corporate character but, instead, by inexperience on the part of both HL&P and its contractor, B&R.

HL&P had early notice of problems arising out of its utilization of B&R. For example, to consider only design engineering — the area which eventually led to B&R's dismissal as architect-engineer/construction manager — Mr. Jordan testified as to HL&P's expectation that around 50 percent of the design engineering work would have been completed at the time of NRC's award of the construction permit whereas, in fact, only about 8-9 percent of the engineering was actually complete at that time (Finding 60). Moreover, given this notice, HL&P should have taken steps earlier than it did to correct the problems which were apparent (Tr. 9524 (Crossman)). Although this delay is perceived by CCANP as a product of deficient character (CCANP FOF, ¶¶ 6.1, 6.31-6.33), we find that it more credibly may reflect a facet of HL&P's inexperience. In our view, in the days prior to the Show-Cause Order, HL&P was not sufficiently knowledgeable to realize that major corrective actions were needed or to ascertain what those corrective actions should be. Finally, to the extent that the failure of HL&P to react sooner may be attributed to a character deficiency, the strong steps taken by HL&P to correct its inexperience (the effectiveness of which we shall treat under Issue B, *infra*) in our view counterbalances any character deficiencies which HL&P may have demonstrated.

CCANP claims that merely trying (without succeeding) is not sufficient to support an award of operating licenses (*see, e.g.*, CCANP FOF, ¶¶ 2.17, 2.19, 2.27, 2.29, 2.42). We agree. But if by this claim CCANP means to assert — as we believe it does — that attempts to achieve quality should not be taken into account in evaluating character (irrespective of the degree of success of those attempts), then we must demur from that position. In our view, attempts to achieve quality are pertinent to character. Their degree of success must be taken into account, but not necessarily in terms of character, in determining whether operating licenses should be awarded (a subject we shall reach again in our consideration of Issues B and E).

In short, an applicant's sincere attempts to correct deficiencies may be viewed as favorable from a character standpoint irrespective of success. If licenses were to be denied because of the failure to take adequate corrective actions, the denial would not necessarily be premised upon the applicant's lack of character. In that context, we view the strong attempts made by HL&P to improve its own performance — particularly its extensive replacement of key managerial personnel with persons of greater nuclear experience and its eventual replacement of B&R with Bechtel and Ebasco — as strongly indicative of favorable character.

(iii) The next character trait which we find pertinent for evaluating HL&P's character is that of responsibility. In particular, did HL&P remain responsible for STP QA/QC activities or did it abdicate its responsibilities to B&R?

We begin by agreeing with the Staff (Finding 114) that, although HL&P may delegate the authority for conducting a QA/QC program to contractors and subcontractors, it remains responsible for the program. We also agree with the witnesses for both the Applicants and Staff that, at least at upper management levels, HL&P did not abdicate responsibility to B&R for the QA/QC program (Findings 114, 118-120).

On the other hand, at lower levels, HL&P did not exercise effective control prior to the Show-Cause Order in areas such as auditing (Finding 116). We attribute the lack of effective control to inexperience and excessively long chains of command rather than to abdication of responsibility. HL&P's willingness to remedy those deficiencies, and the steps it took to effectuate those remedial steps, convince us that HL&P did not abdicate responsibility for STP to an extent which would reflect on its character or which would derogate from its eligibility for operating licenses.

In its view, CCANP believes that HL&P's manner of exercising responsibility for STP is reflected by the lack of familiarity of HL&P management officials (particularly Messrs. Jordan and Oprea) with certain details arising out of the STP; by HL&P's failure to remove certain personnel (especially Mr. Richard A. Frazar, the QA manager); by HL&P's over-reliance on B&R; and by HL&P's failure to recognize B&R's inadequacies at an earlier date (CCANP FOF, ¶¶ 5.24.[1]-5.24.8, 5.25.2, 5.29, 6.8). All of these deficiencies, in CCANP's view, reflect HL&P's lack of character.

We do not agree with CCANP's analysis, even though some of the facts it relies on are accurate. For example, the record citations which are said to demonstrate a lack of familiarity with details on the part of Messrs. Jordan and Oprea do in fact reflect that neither officer was aware of every single project detail. Nor would we expect them to be. In our view, both of them have been exposed to a level of detail commensurate with their corporate positions (*see* App. Reply FOF, ¶ 36). What we fault them for is not their lack of awareness of details but their lack of understanding of the facts which they had before them. This represents in our view a defect in competence rather than character. The circumstance that Messrs. Jordan and Oprea attempted to improve their competence in QA matters, as in attending the Crosby College seminar (Finding 215), reflects favorably upon their character (as well as that of HL&P).

HL&P also tolerated deficiencies in personnel for too long a period of time, and it over-relied to some extent on B&R, particularly during the early stages of the project (Findings 115, 132). But these deficiencies are also traceable to lack of experience. When Mr. Jerome H. Goldberg was finally hired to head HL&P's nuclear construction program, he brought his many years' experience to bear in acting quickly to upgrade the project. Within 3 months of his assuming his duties, HL&P had commissioned the Quadrex Report and, at Mr. Goldberg's initiative, had begun consideration of alternatives to B&R (Finding 224). Given the magnitude of the changes which proved to be necessary, it is understandable, if not entirely acceptable, for HL&P to have waited as long as it did to act. In short, we regard HL&P's responsibility as not reflecting adversely on its character, although raising questions as to its competence. Neither with respect to character or competence, however, would HL&P's responsibility disqualify it from receiving operating licenses.

(iv) The other trait which bears on HL&P's character is the degree to which it attempted to stay informed about STP. The facts relevant to this question are to some extent the same as bear upon HL&P's exercise of responsibility: HL&P received a large quantity of information about the STP but was unable to assess the significance of much of it. Part of this problem stemmed from the lack of an adequate system of trending nonconforming conditions during the period prior to the 79-19 Investigation (Finding 128). The excessively long communications lines between personnel on site and upper management officials prior to Investigation 79-19 also resulted in HL&P management officials not being aware as rapidly as they should have been of various developments at the STP: management was informed of individual problems which were occurring but was not given either the causes or sufficient detail to ascertain the causes (Findings 96, 129, 132).

CCANP points to HL&P's alleged lack of knowledge as a deficiency in "perception" and hence as a character defect (CCANP FOF, ¶ 5.0 *et seq.; see also* CCANP FOF, ¶ 2.36.9). The gist of CCANP's claim is that HL&P never really perceived the difference between building a nuclear and a fossil-fired plant and, for that reason, was unable to deal successfully with the QA/QC requirements inherent in constructing a nuclear plant. We note in particular CCANP's claim that HL&P failed to perceive signals of developing trouble (CCANP FOF, ¶ 5.21).

In some respects, CCANP's observations are well founded. HL&P did not, prior to the 79-19 Investigation, recognize the quality differences between the construction of nuclear and fossil-fired plants. When it saw problems developing, it often failed to react effectively. Problems continued to recur. But every expert witness addressing this question attributed the recurring problems not to a lack of character but to a lack of experience on the part of both HL&P and B&R. We find no reason to disagree with that assessment.

(v) To sum up the facets of HL&P's character which we find pertinent, HL&P has been open and candid with the NRC. It demonstrated those same qualities in its relationship with CEU (which resulted in CEU's withdrawal from the proceeding). It has done its best — although not always with success — to deal with the many QA/QC problems it faced. It has assumed responsibility for all aspects of the STP. Although it perhaps at first left too much responsibility to B&R, it remedied that situation and became more involved with the project. It also exposed itself to great quantities of project information, although it was not always sufficiently knowledgeable to react properly to that information. Its shortcomings in this area are attributable in large part to lack of experience. In conclusion, HL&P has not demonstrated character deficiencies which would preclude the Applicants from being granted operating licenses for the STP. Subject to the outcome of the Phase II hearings on the Quadrex Report, we have reasonable assurance that HL&P has sufficient character for the Applicants to be granted operating licenses.

*b. HL&P's Competence*

In our discussion of the meaning of competence (Part II.B.1, *supra*), we pointed to three lines of inquiry which are appropriate for evaluating an applicant's competence — *i.e.*, (1) whether the applicant's staff and management have sufficient technical and managerial expertise and experience; (2) whether that staff and management are organizationally structured so as to permit and encourage the unhindered application of their expertise and experience; and (3) whether the applicant's programs and procedures require the application of that expertise and experience and are consistent with regulatory goals. No party has raised any question with respect to the third line of inquiry — which constitutes in effect an inquiry into the adequacy of the written QA programs of HL&P and its contractors. All witnesses addressing the programs (as they existed both before the Show-Cause Order and as subsequently modified) considered them as in compliance with applicable regulatory requirements, and we see no reason to disagree. (*See,* in particular, Findings 112, 143, 264, 268.)

It is in the implementation of the HL&P and B&R QA programs where difficulties have arisen. We turn our attention, therefore, to the first two lines of inquiry.

(i) The first line of inquiry involves expertise and experience. Both of these characteristics are necessary contributors to an applicant's technical competence. HL&P and B&R each employed many talented individuals with adequate or more-than-adequate technical expertise (Finding 118). Some of those individuals testified before us. Where necessary, HL&P and B&R were also willing to hire consultants or subcontractors. However, to some extent, B&R did not have sufficient depth of expertise prior to the 79-19 Investigation (Findings 59, 103, 111, 142). Moreover, the rapid turnover of B&R managerial personnel accentuated B&R's lack of experience — *e.g.*, it took 3-6 months for a General Manager to become fully effective, and the occupants of that position averaged only 8 months' tenure between 1977 and 1981 (Findings 105-106, 111). But it was in the area of experience in nuclear design and construction where both HL&P and B&R were lacking prior to the 79-19 Investigation.

The lack of experience of both HL&P and B&R in our opinion produced synergistic effects. As Mr. Goldberg observed, B&R's manner of carrying out its QA/QC responsibilities was typical of a first generation nuclear project (Finding 141). Or, put another way by the Staff, B&R's serving as construction manager, architect-engineer and constructor was a "very ambitious program, especially when you don't have much experience" (Tr. 10,707 (Gilray)). Thus, despite extensive efforts to correct nonconforming conditions, many of the conditions continued to recur. These continued recurrences led to enlargement of the QA/QC staff, to the point where it became excessive for the amount of construction and design work in progress (Findings 118, 141).

One of the most pointed reflections of HL&P's and B&R's lack of experience was the continuing reappearance of incidents of harassment of QC personnel by construction personnel. B&R and/or HL&P investigated each of the incidents which were reported. B&R took steps to punish the persons responsible for such harassment, and HL&P and B&R instituted enhanced procedures to settle disputes which might arise between QC and construction personnel (Finding 393). Nonetheless, reports of low morale of QC inspectors, resulting in part from their clashes with construction personnel, continued to surface (Findings 223, 397).

We view the existence of the incidents and of low QC inspector morale to be in part the result of lack of managerial experience with projects involving QA/QC requirements such as those attendant to nuclear construction. As various witnesses pointed out, clashes among construction workers (including QC personnel) are to some extent to be expected, given the nature of the work and the characteristics of persons engaged in it. But the continued reappearance of clashes and the persistence of low morale reflect management's inadequate experience in constructing facilities subject to nuclear QA/QC requirements.

In short, the record clearly demonstrates that HL&P (and B&R), prior to the 79-19 Investigation, lacked one of the important elements of technical competence: experience. Experience, by its very nature, however, is obtainable by several means, including the hiring of experienced personnel or even by the mere passage of time (*i.e.*, the more time one spends on a project, the more experience one acquires). HL&P hired more-experienced personnel, and its involvement in STP by itself provided a degree of experience. Although Issue A excludes consideration of corrective actions, we do not believe we can fairly evaluate HL&P's competence to complete and operate the STP without taking into account the qualifications and experience of the personnel who actually will be engaged in those tasks. Therefore, we are evaluating under Issue A both HL&P's prior experience and the existence of newly acquired experience such as that possessed by Mr. Goldberg. When HL&P's own increased experience is coupled with the additional experience provided by Bechtel and Ebasco, we believe that HL&P has remedied the lack of experience which has plagued the STP. The effectiveness of the newly acquired experience, however, remains for consideration under Issue B (where, *inter alia,* we conclude that further supplementation of the record in this regard is necessary).

(ii) The other line of our inquiry respecting HL&P's competence is the adequacy of the STP organizational structure. As Mr. Amaral and other witnesses pointed out, the extended lines of communication were a prime source of project difficulties (Findings 96-98, 111-112). To that extent, HL&P also lacked technical competence. But, as in the case of experience, a fair evaluation of this aspect of competence can only be given if updated organizational communication lines are taken into account. HL&P greatly shortened the lines of communication — it transferred its QA Manager to the site and enabled him to report directly to the Executive Vice President, Mr. Oprea (Finding 129). In our view, HL&P has thus alleviated the line-of-communication problem which it faced. Under Issue B, we will consider the effectiveness of the changes.

(iii) Taking into account the three lines of inquiry which we regard as pertinent to HL&P's competence, we find no problems with either the present or past written QA programs. We find that HL&P had sufficient technical capability to support its QA program but that it lacked adequate experience prior to the 79-19 Investigation. We also find that, prior to the 79-19 Investigation, HL&P lines of communication were overly extended to a degree which compromised management's ability to deal with problems efficiently. Combining all these factors, we can only conclude that HL&P's competence prior to the 79-19 Investigation was questionable in certain respects (Findings 150, 182).

HL&P took extensive steps to upgrade its experience, and it greatly shortened the lines of communication between its management and persons on site. Without regard to the effectiveness of those measures, we conclude that HL&P's past questionable competence was not of such magnitude as to preclude the eventual award of operating licenses. Had changes not been instituted, there would have remained a serious deficiency in HL&P's competence, possibly sufficient to warrant the denial of operating licenses. Changes were, however, made. Whether operating licenses should be awarded will thus depend, at least insofar as managerial competence in construction is concerned, on the effectiveness of those changes, as reflected in our conclusions with respect to Issues B and D.

### 2. *Issue B: Adequacy of HL&P's Remedial Actions (Findings 188-226)*

Issue B inquires whether HL&P has taken sufficient remedial steps to provide assurance that it now has the managerial competence and character to operate the STP safely. In effect, this issue requires an evaluation of the effectiveness of the numerous steps taken by HL&P to correct the deficiencies identified in Issue A.

We begin by disagreeing with the implication expressed by the Applicants that HL&P's competence was always adequate but that, taking into account remedial steps, it became even better (App. FOF, ¶¶ 336, 362, 363; App. Reply FOF, ¶ 101). Although we agree (for reasons set forth below) that HL&P's competence has greatly improved over what it was prior to the Show-Cause Order, we are not prepared to state that it was always adequate.

The most significant of the deficiencies was the lack of adequate nuclear experience on the part of both HL&P and B&R. HL&P took steps to remedy that deficiency by adding significantly to the experience available to the managerial personnel responsible for the STP.

Most noteworthy in our view was HL&P's hiring in October 1980 of Mr. Goldberg as a Vice President in charge of nuclear construction. Mr. Goldberg brought many years of nuclear experience to the project (Finding 209). He has employed that experience well: in early 1981, shortly after he joined HL&P, the company's executive management commissioned the Quadrex Report, and Mr. Goldberg himself alerted HL&P to the need to consider changes in its major contractor, B&R (Finding 224).

Another example of increased experience is represented by Mr. Joseph W. Briskin, Manager, Houston Operations, who was hired since the Show-Cause Order to direct the work of HL&P's project management team, including engineering, procurement, project control services, accounting and project administration. He has had over 20 years' experience in project control and project management, including 10 on nuclear projects (Oprea, *et al.*, ff. Tr. 1505, at 52).

HL&P has further increased its nuclear experience through its new Corporate QA Manager for STP and Project QA Manager, both of whom bring substantial experience to STP (Findings 211, 213). HL&P also substantially strengthened its QA/QC organizations, at first through the utilization of personnel supplied by the Management Analysis Corporation (MAC) (Finding 212). Thereafter, HL&P hired permanent employees to replace MAC personnel retained on an interim basis (*id.*). In hiring QA/QC personnel generally, Mr. Goldberg indicated that HL&P had recently concentrated on employees who had had substantial prior experience (Tr. 10,480 (Goldberg)).

And, perhaps most important, HL&P replaced B&R with two organizations with far more experience in nuclear design and construction, Bechtel and Ebasco. In our view, these changes should correct the deficiencies in nuclear experience to which many witnesses (representing both the Applicants and Staff) alluded.

Beyond those personnel changes, HL&P's shortening of its organizational lines of communication *per se* alleviated one of the sources of problems which previously existed. On paper, at least, HL&P clearly solved or at least mitigated its pre-existing organizational deficiencies. We have little record evidence, however, by which to judge the effectiveness of the shortened communications lines.

HL&P's responses to all of the various Show-Cause Items have been deemed by the Staff to be satisfactory. HL&P has continued to be open and above-board in its relations with the Staff. Moreover, we repeat that HL&P's offer to permit CEU to participate in an independent evaluation of the QA/QC program (which resulted in CEU's withdrawal from this proceeding) is commendable in every respect and constitutes a confirmation of testimony attesting to HL&P's openness and candor.

We also were impressed by the increasing involvement of HL&P management officials in QA/QC activities, including the attempts by management to increase their visibility in such involvement. Although the testimony on this subject could not relate experiences under the new contractors, we accept HL&P's expressed statements of intent to continue to become actively involved in QA/QC activities.

We here observe that the record includes extensive evidence concerning modifications or revisions to B&R procedures as a result of the Show-Cause Order, many of which resulted from HL&P's direction. *See, e.g.*, testimony concerning improvements in the procedures for processing nonconformance reports and detecting significant trends, and for controlling field changes. Oprea, *et al.*, ff. Tr. 1505, at 64-66, 95-99, 103. Because the revised procedures are not likely to be followed by Bechtel or Ebasco, we are not reviewing the adequacy of these procedures (except to the extent necessary to resolve certain of the Intervenors' contentions). Their adoption, however, strengthens our confidence that HL&P has the necessary character and competence to complete construction of the STP in a satisfactory manner.

As we have also found, however, the record does not reflect that HL&P has been totally successful in eradicating all its difficulties. For example, the SALP report for the period ending June 30, 1981 reflects continuing problems in responding to deficiencies and continuing reports of harassment and intimidation (Finding 223). Moreover, CCANP has observed, with respect to whether HL&P has taken sufficient remedial steps to improve its competence since the Show-Cause Order, that the "functions of architect-engineer, construction, and QA/QC have changed so recently that there is no record from which to judge the adequacy of that competence." It concedes that Bechtel and Ebasco might have "the best possible" reputations but points out that "this proceeding is not a construction permit proceeding." CCANP concludes that, at best, the issue of remedial measures regarding technical competence must remain unanswered. CCANP FOF, ¶ 10.3.2; *see also* CCANP FOF, ¶ 1.43.

Those observations are to some extent accurate. The record does reflect the initial performance of some recently acquired HL&P personnel (such as Mr. Goldberg). But the information concerning Bechtel and Ebasco is indeed limited to their experience on other projects, both nuclear and non-nuclear. The QA/QC programs of these companies are basically standardized programs previously approved by the Staff but modified in certain details to take into account the facts of STP. The Staff review of the new organization and the applicable QA/QC programs was largely (and of necessity) a review of Bechtel's and Ebasco's experience on other projects plus a reliance on prior Staff generic approval of the basic QA/QC programs.

CCANP is correct in its suggestion that, in an operating license proceeding such as this one, there should be more than reputation by which to evaluate the competence of the architect-engineer and/or constructor. Any evaluation at the operating license stage should take into account more than the type of information by which competence is judged during the construction permit review. CCANP is also correct in its observation that, insofar as Bechtel and Ebasco are concerned, the record here thus far includes only the type of information as would be available during a construction permit review — indeed, the same type of information as was reviewed (with respect to HL&P and B&R) at the construction permit stage of this proceeding.

The answer to this seeming deficiency in the record is, however, not the one suggested by CCANP — *i.e.*, to evaluate HL&P's competence solely on the basis of past performance on this project (CCANP FOF, ¶¶ 1.33, 2.1). For that course would ignore what undoubtedly is one of the most significant developments bearing on the construction of the

STP — the replacement of B&R by Bechtel and Ebasco. Instead, we are reaching only a preliminary evaluation of the competence of HL&P, Bechtel and Ebasco based on HL&P's performance since the Show-Cause Order and Bechtel's and Ebasco's reputations. We are similarly reaching only a preliminary conclusion concerning the effectiveness of corrective actions.

In addition, to enhance the record concerning the on-the-job performance of Bechtel and Ebasco, as well as up-to-date performance by HL&P, we are hereby requiring a report to us by the Staff concerning the performance of HL&P, Bechtel and Ebasco at STP since the close of the Phase I record. This report is to be presented during the Phase II evidentiary hearings and is to encompass (although not necessarily be limited to) such matters as the effectiveness of Bechtel and Ebasco procedures in areas which have been the subject of Phase I litigation, violations (if any) of applicable requirements, nonconformances (particularly, although not limited to, the civil structural area), altercations (if any) between construction and QC personnel, and SALP evaluations. The foregoing report should include the Staff's evaluation of the adequacy of the Applicants' implementation of their QA/QC program for construction. Although this report requirement is directed at the Staff, other parties (including the Applicants) are invited to supplement or comment upon that report or provide their own reports, if they wish.

### *3. Issue C: Character and Competence to Operate the STP (Findings 227-249)*

Issue C questions whether, in light of HL&P's planned organization for operation and the alleged deficiencies in its management of construction (including its past actions or lack of action, revised programs for monitoring the activities of its architect-engineer-constructor and those matters set out in Issues A and B), there is reasonable assurance that HL&P will have the competence and commitment to operate the STP safely.

As background for this issue, we begin with our resolution of Issues A and B: namely, that HL&P was not lacking in character to any significant degree and that its construction competence, although to some extent questionable prior to the Show-Cause Order, appears to have improved sufficiently so as not to preclude our making the reasonable assurance findings of 10 C.F.R. § 50.57. This resolution is of course subject both to our future consideration of the Quadrex Report and to the implementation review which we have directed under Issue B.

In evaluating the Applicants' proposals for operation, the Staff expressed its opinion that management of facility construction presents a more challenging problem than management of plant operation. The Staff also notes that HL&P's prior experience in operating power plants is substantially greater than in constructing them. The Applicants rely on these opinions (App. FOF, ¶ 374), whereas CCANP disagrees (CCANP FOF, ¶ 2.5). We decline to determine, or to base any of our conclusions on, the degree of challenge to management presented by construction or operation. Both must be managed adequately, and both present difficult, but different, challenges. Moreover, HL&P's operating experience thus far relates solely to non-nuclear facilities. In effect, we regard these matters as not helpful to our determination under this Issue as to whether HL&P has the character and competence to operate the STP safely.

Considering the stage of construction completion of the STP at the close of the Phase I record, HL&P's plans for operation are well under way (Findings 229, 243). Since the issuance of I&E Report 79-19, HL&P's upper management has been intimately involved with construction activities at the STP. It appears to be aware of plant status with a mind toward transition from construction activities to plant operation, and it has made substantial progress in formulating organizational plans for such operation. Based upon the testimony and our observation of witnesses from HL&P's upper management who are to be responsible for plant operation, we have reasonable assurance that HL&P is dedicated to safe plant construction and operation. Further, HL&P apparently intends to ensure that this objective remains paramount in the minds of its employees. Key positions within the plant operations staff are already filled with individuals possessing appropriate qualifications. That staff has been engaged in writing procedures and participating in transition and start-up activities.

The NRC Staff in its review of HL&P's plans for operation has concluded that HL&P's planned management and operating organizations meet the requirements of the applicable NRC rules and regulations. Although this review of necessity was preliminary in nature, we find no reason at this time to disagree. We anticipate, however, that at a time closer to operation the Applicants will update information bearing upon the organization and personnel for operation, and the Staff will review the updated information. At the time we consider Issue F (QA for operation) during Phase III, we would expect that the Applicants and Staff will supplement the record with such updated information.

CCANP submitted no detailed proposed findings and conclusions on Issue C. Instead, it merely referenced its conclusions on Issues A and B and asserted that, since Issue A must be answered in the affirmative and Issue B in the negative, Issue C must also be answered in the negative (CCANP FOF, ¶ 11.1). Since our conclusions on Issues A and B do not parallel those of CCANP, we could not, and do not, accept CCANP's conclusions as a basis for our holdings on this Issue.

For these reasons and those more fully set forth in our findings, we conclude that there is now reasonable assurance that HL&P will have the competence and character, as well as the requisite commitment to safety, to operate the STP safely. This conclusion is based solely on the preliminary information currently of record and will be subject to any updated information added to the record in Phase III.

### *4. Issue D: Adequacy of Current Construction QA Programs (Findings 250-272)*

Issue D, as admitted in the Second Prehearing Conference Order (December 2, 1980), questions (1) whether HL&P's and B&R's construction QA/QC organizations and practices meet the requirements of 10 C.F.R. Part 50, Appendix B, and (2) whether there is reasonable assurance that the QA/QC program will be implemented so that construction of STP can be completed in conformance with the construction permits and other applicable requirements. Subsequent to the admission of that issue, B&R was replaced by Bechtel and Ebasco as architect-engineer/construction manager and construction contractor respectively. The Board advised the parties that it would consider the adequacy of the QA/QC program as modified by the change in contractors. Fourth Prehearing Conference Order (unpublished) at 3-4 (December 12, 1982). Issue D has therefore been modified by replacing references to "Brown and Root (B&R)" with references to "Bechtel/Ebasco." In addition, we also advised the parties that we would consider in Phase I the related question of the adequacy of the modified organizational framework for continued construction, including consideration of plans for design, a review of past problems, project construction, and HL&P management involvement (*id.* at 4).

HL&P's most current QA program is essentially the sum of three programs: the updated Staff-approved QA program for the HL&P quality assurance-related activities and the separate QA programs of the two current principal contractors, Bechtel and Ebasco. The HL&P portion of the QA program provides for an improved QA organization with increased authority and responsibilities for surveillance by HL&P personnel during the day-to-day design and construction activities. Bechtel for its part has committed to apply its Staff-approved quality assurance topical report, as modified to meet its assigned architect-engineer and construction manager functions. Similarly, Ebasco has committed to apply its Staff-approved quality assurance topical report, as modified to meet its function as the constructor.

We agree with the opinions expressed by witnesses for both the Applicants and Staff to the effect that the new QA organization, representing an additional layer of QA review not present when B&R had both construction and construction-manager roles, is beneficial. Additionally, both Bechtel and Ebasco have extensive nuclear experience in the functions to which they have been assigned at the STP — more so than did B&R. Moreover, we are impressed with the results of the Staff's preliminary review of both organizations, which indicates that they are selecting individuals with considerable qualifications and experience to manage their responsibilities at the STP. Finally, HL&P itself appears to be seeking and attracting highly qualified individuals to run its construction program, including the QA/QC aspects of that program. In short, the program appears likely to be superior both to that utilized prior to the Show-Cause Order and to the program as it evolved after the Show-Cause Order but prior to the replacement of B&R by Bechtel and Ebasco.

Neither through cross-examination nor in its proposed findings did CCANP succeed in refuting the extensive direct evidence offered by the Staff and Applicants on Issue D. Indeed, CCANP's proposed findings on this issue did not discuss any of the testimony bearing on the adequacy of the construction organizations. As in the case of Issue C, they merely referenced CCANP's conclusions on Issues A and B and asserted that, since Issue A must be answered in the affirmative and Issue B in the negative, Issue D must also be answered "no" (CCANP FOF, ¶ 12.1). Since we do not agree with most of CCANP's conclusions on Issues A or B, we could not for those reasons accept CCANP's conclusions as a basis for our holdings on this Issue.

Accordingly, for the detailed reasons set forth in our findings, we find that the present QA/QC organizations and practices for the STP meet the requirements of 10 C.F.R. Part 50, Appendix B, and that there currently is reasonable assurance that they will be implemented so that construction of STP can be completed in conformance with the construction permits and applicable regulatory requirements. We note that, insofar as this finding deals with implementation of the QA/QC program for construction, it is a preliminary finding which will be subject to the Phase II report which we are requiring (described under Issue B, p. 697, *supra*).

### 5. *Issue E: Adequacy of Existing Structures (Findings 273-316)*

Issue E inquires as to whether there is reasonable assurance that the structures now in place at the STP (and referred to in Sections V.A(2) and (3) of the Order to Show Cause) conform to the construction permits and applicable regulatory requirements; and, if not, whether HL&P has taken steps to assure that such structures are repaired or replaced as necessary to meet such requirements.

The Show-Cause Order, and Investigation Report 79-19, pointed to numerous deficiencies in the areas of structural backfill, concrete placements and voids, and welding. As a result, HL&P conducted a comprehensive verification program in those areas. Through the verification program, additional deficiencies were identified in the Category I structural backfill. In addition, voiding was detected in some concrete structures and problems were identified in AWS and ASME welding.

No evidence was developed in the record to indicate that any structure or compacted backfill is now inadequate for its intended function. With respect to voids detected, they were properly grouted and retested for adequacy (*see* Contention 1.2, *infra*). Welds were reexamined and necessary corrective action has been performed. Extensive evidence was developed to indicate HL&P performed a comprehensive verification program relative to existing structures and took adequate corrective action where deficiencies were detected. Analyses were performed which established reasonable assurance that concrete or welds which were inaccessible and possibly deficient would not present a safety hazard.

In cross-examination, the Intervenors did not refute any of the extensive direct evidence entered by the Applicants and Staff on Issue E. In its proposed findings (CCANP FOF, ¶¶ 13.1-13.9), CCANP does not address the evidence of record on any of the three sub-issues comprehended by Issue E — namely, adequacy of structural backfill, concrete verification, and welding verification. Instead, CCANP reiterates its view of earlier "widespread noncompliance" in the implementation of the QA/QC program (particularly, instances of harassment of QC inspectors, high turnover of inspectors, and the Staff's inability to locate certain former inspectors for questioning) and concludes (CCANP FOF, ¶ 13.9):

> In light of the history of the project and the difficulty of answering all doubts, the in-place condition of the structures must be considered indeterminate. Tr. 5541.

For that conclusion, CCANP cites the opinion of HL&P's former QA manager, who expressed the view (in response to a hypothetical question) that if a QC inspector had signed off an inspection without actually performing the inspection, the condition of the work would be "indeterminate" (Tr. 5541 (Frazar)). Although Mr. Frazar's conclusion is entirely appropriate, its premise is not generally established in this record, except in extremely limited circumstances. Indeed, the evidence of record is largely to the contrary.

By its terms, Issue E questions only the adequacy of specified in-place structures and components, and the Applicants' proposed findings and conclusions on Issue E are so-limited. The Staff, however, has asked us also to conclude that there is reasonable assurance that future backfill work, concrete work and welding activities will meet applicable requirements (Staff FOF, ¶¶ 163, 172 and 185). The Staff provided a basis only with respect to the concrete work — *i.e.*, the "numerous improvements in the procedures for placing concrete" (*id.* at ¶ 172). We decline to reach this conclusion, both because it is beyond the scope of Issue E and because the record is not sufficient to support it. The improved procedures discussed at great length by various witnesses were those adopted during B&R's tenure on the project. Indeed, HL&P confirmed that, although the quality goals would remain the same, B&R's procedures would not continue to be used and Bechtel or Ebasco procedures would be substituted (Finding 225). The record at this time does not reflect the procedures which are to govern backfill work, concrete work, or welding activities.

For the foregoing reasons, we have adopted findings on Issue E with respect to in-place structures (but not with respect to future work) submitted by the Staff or Applicants and are rejecting CCANP's proposals. We find that, as of the close of the Phase I record, there is reasonable assurance that the structures in place at the STP are in conformity with applicable regulatory requirements.

## C. Intervenor Contentions

Contentions 1 and 2 allege that there is no reasonable assurance that STP can be operated safely, because of deficiencies in construction and in the construction Quality Assurance/Quality Control (QA/QC) program. The contentions enumerate these deficiencies and assert that, as a result, the findings required pursuant to 10 C.F.R. § 50.57(a)(1) and (2) (that STP conforms with its construction permits, the Atomic Energy Act, and NRC regulations, and can be operated in conformity with the operating license application, the Atomic Energy Act, and NRC regulations) cannot be made. Further, the contentions assert that the particular deficiencies violate specified criteria of 10 C.F.R. Part 50, Appendix B.

The contentions raise two separable issues: (1) whether each particular, enumerated deficiency by itself demonstrates a nonconformity with the construction permits or NRC regulations (including 10 C.F.R. Part 50, Appendix B) and prevents a finding of reasonable assurance that STP can be operated safely; and (2) whether the deficiencies, when aggregated, are indicative of an overall construction QA/QC program that is or was so defective that there can be no reasonable assurance that STP has been constructed adequately and can be operated safely. The latter and broader issue, however, is completely subsumed by the CLI-80-32 issues, particularly D and E.

Recognizing that the broader issue is incorporated into the CLI-80-32 issues, CCANP declined to submit proposed findings on all but subpart 7(e) of Contention 1 and on all of Contention 2; it explains:

> Given the relative unimportance of most of the intervenor contentions in light of the larger issues in this proceeding, no findings are offered for Contentions [1.]1, 2, 3, 4, 5, 6, 7a, 7b, 7c, or 7d.
>
> * * *
>
> Findings on Contentions [1.]8a through 8.d . . . would also not contribute materially to the record.

CCANP FOF at 134.

Therefore, we could treat those contentions as abandoned and not address them. For, in an operating license proceeding, we need address only matters in controversy among the parties. *See* 10 C.F.R. § 2.760a. Nevertheless, because the specific allegations contained in Contentions 1 and 2 are pertinent to the CLI-80-32 issues, we have, in our discretion, made findings and conclusions. *See Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-123, 6 AEC 331, 332-33 (1973). However, to avoid redundancy, we are addressing Contentions 1 and 2 narrowly and making findings and conclusions only as to each alleged deficiency standing alone; we have treated the adequacy of the overall QA/QC program in our findings and opinion on the CLI-80-32 issues.

As we have previously pointed out, the Intervenors presented no witnesses in support of their contentions (*but see* note 32, *supra*). Each of the contentions was addressed by witnesses of the Applicants and Staff. The Intervenors introduced some documentary evidence.

*1. Contention 1.1 (Findings 318-326)*

Contention 1.1 alleges that a surveying error resulted in the eastern edge of the Mechanical-Electrical Auxiliary Building (MEAB) being constructed 1 foot short, in violation of 10 C.F.R. Part 50, Appendix B, §§ X, XI. The Applicants admitted the error and acknowledged that it arose from poor surveying practices. In addition, the Applicants conceded that there was no procedure for or inspection of actual surveys at the time the error was made, although they denied that the absence of a survey inspection procedure violated 10 C.F.R. Part 50, Appendix B, Criteria X or XI.

The Applicants' and Staff's uncontroverted testimony, and answers to questions posed by the Board, however, clearly demonstrate that the particular surveying error did not create a safety hazard; the equipment layout inside the MEAB was redesigned, still complies with the applicable safety criteria, and creates no difficulty with operation, inspection, maintenance, or replacement of this equipment. The Applicants' uncontroverted testimony also clearly demonstrates that the poor surveying procedures that were the root cause of the error were corrected and (during B&R's remaining tenure on the project) were generally adequate. Finally, the error was properly reported to the Staff pursuant to 10 C.F.R. § 50.55(e).

The Staff takes the position that we need not reach the allegation that the surveying error violated Appendix B, Criteria X and XI, inasmuch as the error was "properly reported" pursuant to 10 C.F.R. § 50.55(e) and "resolved through the provisions of that regulation" (Staff FOF, ¶ 196). Notwithstanding our agreement as to the propriety of HL&P's reporting of the surveying error, we nevertheless do not view that factor as relevant to whether a violation of Criterion X or XI took place or to the desirability of our resolving that allegation of Contention 1.1. Obligations under 10 C.F.R. Part 50, Appendix B, are different from obligations under 10 C.F.R. § 50.55(e) — although a violation of the latter provision might well reflect a violation of 10 C.F.R. Part 50, Appendix B, particularly Criterion X, XVI or XVII. *Texas Utilities Generating Co.* (Comanche Peak Steam Electric Station, Units 1 and 2), LBP-83-81, 18 NRC 1410, 1414 (1983).

For their part, the Applicants assert that the surveying error did not violate either Criterion X or XI of Appendix B. With respect to Criterion X, they claim that it imposes no requirements on surveyors, that inspection of surveying activities is impracticable and that other verification methods are generally adequate. As for Criterion XI, they claim that surveying is a basic activity rather than a test and so is not governed by Criterion XI, which establishes requirements for testing. For both Criteria, the Applicants also cite Staff inspection reports and testimony which concluded that there was no violation. App. FOF, ¶ 262.

As the Applicants acknowledge (*id.*), Appendix B governs various aspects of surveying. But they construe its applicability to surveying very narrowly, limiting it to review of surveying procedures, calibration of instruments and occasional audits (Peverley (Contention 1.1), ff. Tr. 7826, at 8; Tr. 7967 (Peverley)). We read Appendix B, insofar as it applies to surveying, as considerably more encompassing. It establishes, *inter alia,* QA standards for activities affecting the construction of structures important to safety (including, here, the MEAB). Specifically, the pertinent requirements "apply to all activities affecting the safety-related functions of those structures" (Appendix B, *Introduction*). Surveying is a construction activity. The Appendix B requirements have been aptly described as a "sensible, integrated regulatory system." *Comanche Peak,* LBP-83-81, *supra,* 18 NRC at 1413. *Per force,* those requirements are not to be narrowly construed (*id.* at 1413-15). In that regard, it is noteworthy that surveying activities are not excluded, either explicitly or by inference, from any of the Criteria which would otherwise cover various surveying activities. Moreover, the Appendix B regulations control the implementation as well as the establishment of a QA program. *Midland,* ALAB-106, *supra,* 6 AEC at 184. Insofar as an activity such as surveying is concerned, a QA plan must "provide control over activities affecting the quality of the identified structures * * *," and those activities are to be "accomplished under suitably controlled conditions." 10 C.F.R. Part 50, Appendix B, Criterion II. Further, there must be an inspection program for "activities affecting quality * * * to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity" (*id.*, Criterion X).

Inasmuch as surveying is a construction activity and not a test as contemplated by Criterion XI ("Test Control"), we agree with the Applicants that Criterion XI does not govern the surveying practices covered by this contention. But, contrary to the Applicants' position, it is our view that the surveying in question is an activity covered by Criterion X ("Inspection"), as well as by Criterion II ("Quality Assurance Program"). We see no basis for the Applicants' position that the inspection requirements of Criterion X are limited, insofar as surveying is concerned, to the review of procedures, the calibration of instruments and the annual auditing of records. Rather, we construe Criteria II and X as requiring a program either for the inspection of survey activities affecting safety structures or, alternatively, for assuring that mistakes are not made or, when made, are promptly detected and corrected.[38] Furthermore, the program is required to be successfully implemented. HL&P conceded that the surveying procedures utilized for the MEAB constituted poor surveying practices and, as part of its corrective actions, adopted procedures (through B&R) which would generally qualify under the standards we view as applicable (Finding 325; also Tr. 7891 (Peverley)). Moreover, the Applicants admitted that a surveyor's work could be checked by redoing the survey (Peverley (Contention 1.1), ff. Tr. 7826, at 8; Tr. 7969 (Peverley)). Far from being "impractical," as claimed by the Applicants, we would judge such a practice to be a sound surveying practice which should be carried out by independent surveyors (*i.e.*, by surveyors other than those who perform the initial survey).

We note that the requirement of independence, as set forth in Criterion X, does not necessarily require that a resurvey be accomplished by members of a QA/QC organization. Moreover, we would find Criterion X to be satisfied by a program which required the resurvey of selected structures (perhaps on a spot-check basis), together with the other ameliorative measures adopted by HL&P and B&R. The program adopted by HL&P and B&R included such selected resurveys. There should also be a QC check of the records of all surveys, to assure that the appropriate procedures were in fact followed.

Accordingly, we conclude that the absence of a survey inspection procedure (or of a procedure for assuring that survey errors would not be made or, when made, would be promptly detected and corrected) constituted a violation of Appendix B, Criterion X (as well as Criterion II); however, the subsequently adopted procedures (which were in effect during B&R's tenure at STP) are generally adequate. Further, we conclude that there was no violation of Appendix B, Criterion XI. Finally, we conclude that the surveying error does not prevent our making the findings required by 10 C.F.R. § 50.57.

### 2. *Contention 1.2 (Findings 327-337)*

Contention 1.2 alleges that, as a result of a field construction error, extensive voids exist in the reactor containment building (RCB) walls,[39] in violation of Appendix B, §§ IX and X.

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[38] Criterion X specifically permits an alternative to "inspection of processed material or products" in certain circumstances. We construe the Criterion as permitting alternatives to other inspection requirements contained therein, in appropriate circumstances. *See* Tr. 7967-69 (Peverley).

[39] The RCB walls are constructed in circumferential portions called "lifts." A lift is generally 10 feet high. The concrete forming a lift may be the result of more than one pouring. Each such pouring and *(Continued)*

Again, Applicants admitted that a large number of voids had been discovered in the RCB walls of both units. However, the Applicants' and Staff's uncontroverted testimony clearly demonstrated that, once voids were discovered, the Applicants conducted an extensive investigation to locate all voids; and that the RCB walls were adequately repaired by filling voids with grout that is as strong as or stronger than the surrounding concrete. Further, the RCB walls will be tested prior to operation. In addition, the Applicants revised the construction procedures to eliminate the root causes of the voids. (The revised procedures were applicable throughout B&R's tenure on the job.)

Accordingly, the Board finds that there is no residual safety problem and that the voids, now repaired, do not prevent the findings required by 10 C.F.R. § 50.57. With respect to the alleged violations of Criteria IX and X of 10 C.F.R. Part 50, Appendix B, we disagree with the Staff (Staff FOF, ¶ 205) — for the same reasons as described under Contention 1.1, p. 704, *supra* — that we need not reach this question because of adequate reporting under 10 C.F.R. § 50.55(e). We also disagree with the Applicants (App. FOF, ¶¶ 273, 275) that permitting the concrete pouring under inappropriate circumstances does not demonstrate a violation of applicable procedures or of Appendix B. Inasmuch as implementation of a QA/QC program is clearly covered by Appendix B (*see* discussion, pp. 705-06, *supra*), a failure in implementation such as produced the concrete voids in our view constituted a violation of Criterion IX (and, in addition, Criterion II). The failure of QC personnel to have discovered certain voids also constituted a violation of Criterion X.

We must add that the presence of extensive voids in the RCB walls is in our opinion but another reflection of the lack of experience of B&R personnel in carrying out large nuclear construction projects, with the significant QA/QC requirements which must be followed on such projects. The Applicants' investigation revealed that the voids were localized in areas where the structural arrangements (*e.g.*, presence of reinforcing steel bar) were complex. Contributing to the voids in such areas were such factors as access and visibility limitations during concrete pours, insufficient vibrations of the poured concrete, and equipment malfunctions and the delays associated therewith. For example, repeated pump failures caused at least one pour to extend almost 20 hours, leading to marked worker fatigue; further, there was inadequate lighting to

the portion of the lift it forms is called a "placement." On the interior of the RCB walls is a 3/8-inch carbon steel liner that provides a leak-tight membrane for the containment. Construction of the corresponding portion of the liner precedes each placement. *See* Tr. 6536-43.

perform that particular pour after dark. Significantly, the voids were initially discovered not by QC personnel (who had checked the pours in question) but by construction personnel cleaning up after the pour (Finding 332).

In our discussion of CLI-80-32 Issue A, *supra,* we have pointed to B&R's (and HL&P's as well) lack of experience as contributing to the problems which arose during the early years of the STP construction. The voids in the RCB walls are but a prime example. Two of the Applicants' witnesses (Messrs. Joseph F. Artuso and Charles M. Singleton) conceded as much (Finding 333). A constructor with adequate experience would never have allowed a concrete pour on a safety structure like the RCB to be carried out under the conditions reflected in this record. Moreover, an adequate QC operation would have detected the deficiencies which occurred.

### *3. Contention 1.3 (Findings 338-345)*

Contention 1.3 alleges that a field document relating to cadweld[40] inspections had been lost, in violation of Appendix B, §§ VI and XVII.

The Intervenors never identified which field document they claim was lost. However, the document in question apparently was a field sketch, FSQ-030, which should have specified the exact location in the reactor containment building of cadwelds 28H31 through 28H44. FSQ-030 was not lost but in fact was never prepared.

We are convinced by the Applicants' uncontroverted testimony and the Staff's exhibits that these cadwelds were properly inspected and that there is no need to know their exact locations. Knowing the exact location of a batch of cadwelds is only necessary if cadweld test splices show that the batch might be defective; with respect to cadwelds 28H31 through 28H44, there is no such concern. Moreover the approximate location of the cadwelds is known; and if it were necessary, the cadwelds could be found. Accordingly, we find that the lack of the field sketch does not prevent the findings required by 10 C.F.R. § 50.57.

As for the alleged violation of Criteria VI and XVII of Appendix B, the Staff states that we need not reach this question, given the corrective action taken by HL&P. But it concedes that the failure to issue and control document FSQ-030 "would appear to be a violation of Criterion VI." Staff FOF, ¶¶ 211, 213. The Applicants base their proposed finding

[40] A cadweld is a connector used to join two pieces of reinforcing steel bar or to connect a piece of reinforcing steel bar to a structural member. For a further description of the process, *see* Murphy, *et al.* (Contentions), ff. Tr. 6522, at 24-26.

of "no violation" on a failure of proof that a document had been "lost" (App. FOF, ¶¶ 279-280). Although we agree that no document had been "lost," we find the failure to prepare document FSQ-030 violated Criterion VI, and that the failure to have a document like FSQ-030 among the project QA records violated Criterion XVII. For reasons stated earlier, however, there was no safety significance to the particular violations in question.

### *4. Contention 1.4 (Findings 346-354)*

Contention 1.4 alleges that membrane seals[41] in the containment structure are damaged, indicating a violation of Appendix B, §§ X, XV and XVI.

The record shows that an NRC investigation into allegations similar to Contention 1.4 failed to substantiate those allegations. Furthermore, though documentation indicates instances where the membrane seal was damaged during construction, that documentation also indicates that such damage was detected by inspection and repaired. Finally, the Applicants' uncontroverted and unimpeached testimony demonstrated that the membrane seal is a secondary and redundant method of protecting against groundwater seepage; even if there remains undetected damage to the membrane seal, the damage would not cause a safety hazard.

Therefore, we conclude that the damage to the seal, which has been repaired, does not prevent the findings required by 10 C.F.R. § 50.57. We also agree with the Applicants (App. FOF, ¶ 290) and Staff (Staff FOF, ¶ 218) that no violation of Appendix B, §§ X, XV and XVI, occurred with respect to membrane seals.

### *5. Contention 1.5 (Findings 355-359)*

Contention 1.5 alleges that steel reinforcement bars (rebar) are missing from parts of the containment structure, in violation of Appendix B, §§ X, XV and XVI.

The record shows that there have been two NRC investigations into these allegations, neither of which uncovered any irregularities. Moreover, the Applicants' uncontroverted and unimpeached testimony indicated that any instances where rebar had been omitted were the subject of documentation, either an NCR or a field request for engineering action (FREA), and a design change and engineering review.

[41] A waterproofing seal, a laminated sheet material consisting of rubberized asphalt bonded to a polyethylene sheet, is placed around the STP containment building surfaces that are below grade.

Accordingly, we conclude that the allegation in this contention does not prevent the findings required by 10 C.F.R. § 50.57. We also agree with the Applicants (App. FOF, ¶ 295) and Staff (Staff FOF, ¶ 224) that no violation of Appendix B, Criteria X, XV or XVI occurred due to missing rebar in the containment structure.

### *6. Contention 1.6 (Findings 360-367)*

Contention 1.6 alleges that there are cadwelds in the STP facility that cannot be verified with regard to compliance with Appendix B, in violation of §§ IX and X of Appendix B.

Cadweld documentation has been the subject of several NRC investigations and reports. The investigations revealed documentation deficiencies — in particular, missing records — but did not substantiate any allegations of falsification of cadweld records. The investigations also revealed quality control deficiencies, which were rectified by reinspecting cadwelds, retraining inspectors and cadwelders, and revising procedures.

HL&P also identified cadweld record deficiencies, and it and B&R established a task force to review all cadweld records. The task force was able to verify inspection of all but 40 of over 36,000 cadwelds.

We do not find the absence of documentation for these forty cadwelds has a significant impact on the proper construction or safe operation of the facility. The Applicants' unimpeached testimony demonstrated that it is very unlikely that even one of the unverified cadwelds would not meet tensile strength requirements; and even if there were instances where cadwelds did not meet requirements, that failure would be offset by the STP design conservatism.

Accordingly, we conclude that the documentation deficiencies do not prevent the findings required by 10 C.F.R. § 50.57. With respect to the alleged violations of Appendix B, we decline to follow the Staff recommendation that we not reach that question (Staff FOF, ¶ 231). Contrary to the view of the Applicants (App. FOF, ¶ 285), we view the document deficiencies (even though insignificant from a safety standpoint) as at least technical violations of Appendix B, Criteria IX and X (and Criterion VI as well).

### *7. Contention 1.7 (Findings 368-399)*

In Contention 1.7, CCANP makes five allegations, (a) through (e), each of which is said to represent a QC deficiency and to violate 10 C.F.R. Part 50, Appendix B, §§ III and IX. In the first, 1.7(a), CCANP claims that QC inspectors were unable to verify that design changes were executed in accordance with the purposes of the original design. The uncontroverted testimony, however, clearly demonstrated that QC inspectors were not required to verify that changes were executed in accordance with the purposes of the original design. That function belonged to Design Engineering, and was accomplished by the Applicants' design change procedures. The QC inspectors' function was to verify that construction was performed in conformity with the appropriate design documents (as amended) and in accordance with appropriate procedures.

In Contentions 1.7(b) and (c), CCANP alleges that design changes were being approved by persons with inadequate knowledge. CCANP's total failure to prosecute these portions of its contention left the Applicants in the difficult position of being required to prove a negative, *i.e.*, the absence of improper design change approvals. The Applicants assumed that the allegations in 1.7(b) and (c) stemmed from an onsite design change approval procedure implemented in 1978. The procedure allowed construction pursuant to a design change, in advance of formal review and authorization by Design Engineering. The Applicants' uncontroverted and unimpeached testimony, however, demonstrated that *all* design changes were subject to review and ultimate authorization by the responsible and qualified Design Engineer. In addition, an NRC investigation into a similar allegation found no evidence that unqualified persons were approving design changes.

In Contention 1.7(d), CCANP alleges that numerous pour-cards had been falsified. The Staff, however, introduced investigative reports, none of which substantiated the allegation. Moreover, the Applicants conducted a thorough audit of concrete field documents, and this effort also produced no indication of pour-card falsification.

Accordingly, we conclude that the Applicants have rebutted each of the allegations contained in Contentions 1.7(a)-(d). We also conclude, with respect to those contentions, that no violations of Appendix B, Criteria III or IX have been demonstrated. (*See* App. FOF, ¶¶ 299, 301 and 303, and Staff FOF, ¶¶ 238, 242, 245, and, insofar as it covers Contention 1.7(d), ¶ 252.)

Finally, in Contention 1.7(e), CCANP alleges that there has been (and continues to be) harassment of QC inspectors, in the form of assaults and threats of bodily harm from construction workers and the firing of QC inspectors; and that, as the result of friction between QC inspectors and construction personnel, QC inspectors decided to play cards rather than perform their inspections. As we previously observed, this is the only one of its contentions on which CCANP submitted proposed findings. Those findings cover only the allegation concerning past and continuing harassment. In that regard CCANP claims that the contention "has been proven beyond any doubt" (CCANP FOF, ¶ 14.1). It cites the Staff's Show-Cause Order (Staff Exh. 46). Under the aegis of Issue A, however, CCANP also asserts that the instances of harassment and intimidation were chronic and reflect a deficiency in HL&P's character (CCANP FOF, ¶¶ 3.9, 6.22-6.28, 7.3.9).

The record as a whole does reveal that there was friction between QC inspectors and construction personnel, and that there were incidents of harassment of and threats against QC personnel. However, the record refutes the allegation that as a result QC inspectors elected to play cards and not perform their inspections. This allegation was made by a former B&R inspector after he was discharged for allegedly soliciting a bribe. A subsequent NRC investigation of the allegation revealed that there had been card games only during lunch or during periods of low construction activity, and that these card games did not interfere with the QC inspectors' pursuit of their duties. These investigative results were consistent with a B&R investigation of the allegation, and with the direct testimony before this Board of two of the QC inspectors who were alleged to have been among the card players (one of whom appeared at the Board's request).

As we have pointed out in discussing Issues A and B, incidents of harassment of QC inspectors are not unique to STP. Nevertheless, the incidents of harassment of QC inspectors at STP were frequent enough to represent a serious indictment of B&R's managerial competence. *See, e.g.,* Findings 62, 64, 74-75, 223, 381-397. As to whether the incidents of harassment represent violations of Appendix B, Criteria III and IX, the Applicants have submitted no findings on this question, and the Staff finds no violation (Staff FOF, ¶ 252). We conclude that the incidents do not represent violations of Criteria III and IX, as alleged, although they do represent violations, in our opinion, of the general implementation requirements of Criterion II. Although B&R (assisted by HL&P) took steps to eliminate the harassment, the record does not reflect whether, if it had remained on the project, B&R would likely have succeeded in doing so. The recurrence over the course of several years of incidents of harassment, notwithstanding attempts to eliminate them, create certain doubts in this regard.

With B&R no longer on the job, we trust that HL&P and its new contractors will take such steps as are necessary to prevent the recurrence of incidents of harassment of QC inspectors. The past records of Bechtel and Ebasco on other projects, to the extent addressed by witnesses before us, do not appear to reflect that they have encountered significant difficulties of this type. Nor would the record, as it stands, preclude our making the "reasonable assurance" findings of 10 C.F.R. § 50.57, given the corrective actions taken by HL&P and the generally favorable track records of the new contractors. Because the Phase I hearings were concluded prior to the performance of significant safety-related work by the new contractors, however, we have no record of actual performance on which to found any prediction in this regard. For that reason, as previously discussed (p. 697, *supra*), we are requiring the Staff (and the Applicants and other parties if they wish) to provide a report to us during the Phase II hearings which covers, *inter alia*, any further incidents of harassment or intimidation which may have surfaced since Bechtel and Ebasco assumed their duties.

### *8. Contention 1.8 (Findings 400-413)*

In Contention 1.8, CCANP makes four allegations derived from the investigative results of I&E Report 81-28 (Staff Exh. 124). Each of the allegations asserts a violation of one or more criteria of 10 C.F.R. Part 50, Appendix B.

In Contentions 1.8(a) and 1.8(b), CCANP alleges that the investigative results of Allegation 1 of I&E Report 81-28 indicate that HL&P failed to assure prompt corrective action of an access engineering problem and that HL&P management does not have a consistent policy on the issuance of stop-work notices. However, in I&E Report 81-28, the NRC inspectors concluded that HL&P had violated no NRC requirement with regard to that allegation. Moreover, the Applicants' unimpeached and uncontroverted testimony demonstrates that HL&P acted decisively and promptly to correct the access engineering problem. The same testimony also indicates that HL&P's QA stop-work procedures are adequate.

In Contention 1.8(c), CCANP alleges that the results of Allegation 2 of I&E Report 81-28 evidenced a lack of managerial commitment to observing NRC regulations. However, the NRC inspectors concluded that HL&P had not failed to meet an NRC requirement; rather, an HL&P manager had made a confusing statement at a meeting in an attempt to address a QC problem, and had resolved the confusion in a subsequent letter of clarification. This conclusion was corroborated by the Applicants' uncontroverted and unimpeached testimony.

In Contention 1.8(d), CCANP alleges that the investigative results of Allegation 4 of I&E Report 81-28 evidence a failure on the part of HL&P management to implement effectively a QA program. Again, however, the investigation of the allegation revealed no instance where HL&P failed to meet an NRC requirement.

Accordingly, we conclude that the investigative results of I&E Report 81-28 do not support CCANP Contention 1.8 but, rather, rebut it. Therefore, the investigative results do not prevent a finding pursuant to 10 C.F.R. § 50.57. Nor do they support any violation of Appendix B.

### 9. *Contention 2 (Findings 414-425)*

In Contention 2, CCANP alleges that NRC Inspection Reports 77-03 and 78-08 indicate that construction records have been falsified, in violation of Criteria VI and XVII of Appendix B.

I&E Report 77-03 does substantiate the falsification of construction records by an employee of a subcontractor. A Pittsburgh Testing Laboratory (PTL) employee falsified records to show tests on concrete that were not performed. However, the Applicants' uncontroverted and unimpeached testimony and the NRC inspection report demonstrate that HL&P discovered the falsification first and promptly notified the NRC, that the PTL employee was promptly discharged, and that other tests and a review by the subcontractor negated any safety consequences that might have resulted from the nonperformance of the tests.

I&E Report 78-08, the second report referred to in CCANP Contention 2, does not address any instance or allegation of falsified construction documents. That report addressed an instance where a QC inspector had marked a record print to indicate a completed inspection when in fact he had only completed a partial inspection. However, the inspection report and the Applicants' uncontroverted and unimpeached testimony clearly demonstrated that the incident was not an instance of deliberate falsification, but rather was a misinterpretation based on an ambiguity in the QC procedures.

It is also possible that CCANP meant to refer to I&E Report 78-09; this inspection, although it addressed alleged construction document falsifications, did not substantiate such allegations.

Accordingly, the Board finds that the inspection reports referred to in Contention 2 do not prevent findings pursuant to 10 C.F.R. § 50.57. Although the falsification of construction records by the PTL employee constitutes a violation of 10 C.F.R. Part 50, Appendix B, Criterion VI, there was no culpability in this regard by HL&P or B&R management. Indeed, that one instance was swiftly and effectively identified and rectified by HL&P and B&R.

## IV. MOTION TO REOPEN PHASE I RECORD

On August 8, 1983, CCANP filed a motion to reopen the Phase I record. In responses dated August 23 and 29, 1983, the Applicants and NRC Staff, respectively, each opposed the motion. (The Staff supplemented its response by letter dated September 12, 1983, which directed our attention to a pertinent Appeal Board opinion issued on August 31, 1983, subsequent to the Staff's filing of its response.) For reasons set forth in this portion of our Opinion, we agree with the Applicants and Staff that the information sought to be added to the record is essentially cumulative and, accordingly, that the Phase I record should not be reopened.[42]

A. CCANP seeks to reopen the record to include assertedly new information bearing on allegations of harassment and intimidation of B&R QC inspectors by B&R construction personnel, and alleged alteration or falsification of QC documents by B&R employees. We have dealt extensively with these general subjects in our rulings on Issues A and B and Contentions 1.7(d), 1.7(e) and 2. The material for which CCANP now seeks to reopen the record is that associated with an investigation conducted by the Commission's Office of Inspector and Auditor (OIA). Specifically, it seeks to include information in the record with respect to three areas:

(1) the concerns which led OIA to conduct its investigation;
(2) the interaction between OIA and the Department of Justice; and
(3) the OIA report, including attachments.

In the latter regard, CCANP points to at least one altercation mentioned by OIA which, it claims, was not encompassed by testimony in this proceeding, together with several of OIA's conclusions.

In opposing CCANP's motion, the Applicants stress that the material sought to be added to the record by CCANP is to a large degree cumulative of evidence already in the record and hence is not significant enough to warrant a reopening of the record. They also suggest that, at least with respect to document falsification, the motion may not have been submitted in a timely fashion (Applicants' response at 3 n.**). For its part, the Staff does not contest either CCANP's timeliness or the significance of the issues involved. It opposes reopening the record on the ground that the material involved would not lead to a different result in the proceeding.

[42] Given the conclusion we are reaching, we have deemed it appropriate to await issuance of this Partial Initial Decision to provide our ruling on CCANP's motion.

B. As all parties recognize, and as we earlier pointed out in our Memorandum and Order (Denying CCANP's Motion to Reopen Record), dated January 10, 1983 (unpublished), the proponent of a motion to reopen a record bears a heavy burden. It is well established that the motion must be timely and must address significant safety (or environmental) issues. *Id.* at 2-3, and cases cited therein. Furthermore, where the record of a proceeding (or at least of a major phase thereof) is closed, the information sought to be included in the record must be shown to be material and significant — *i.e.*, to have at least the potential for altering a result which might otherwise be reached. *Id.*[43] To meet this standard, the proponent must offer new and significant *factual* information relating to the issue in question. *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), ALAB-644, 13 NRC 903, 994-95 (1981).

The "timeliness" test is clearly subsidiary to that of materiality or significance. As the Appeal Board has observed:

> A board need not grant a motion to reopen which raises matters which, even though timely presented, are not of "major significance to plant safety" * * *. By the same token, however, a matter may be of such gravity that the motion to reopen should be granted notwithstanding that it might have been presented earlier * * *.

*Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-138, 6 AEC 520, 523 (1973) (citations omitted); *see also Midland,* LBP-83-50, note 43, *supra,* 18 NRC at 249.

In conclusion, we are considering this motion in terms of the materiality and significance of both the issue and the information sought to be included in the record, and to a subsidiary degree, the timeliness of the motion.

C. 1. We need not dwell long on either the timeliness of CCANP's motion or the significance of the issue which the information in the reopened hearing would address.

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[43] If an initial decision on a particular question had been issued, a motion to reopen the record would be denied if the material sought to be added to the record were not susceptible of altering the result previously reached. *Louisiana Power & Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-753, 18 NRC 1321, 1324, 1328 (1983); *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit No. 2), ALAB-486, 8 NRC 9, 21 (1978). On the other hand, if the record were closed on a particular issue but not on the portion of the proceeding in which the issue was included, the susceptibility of altering the result would not be pertinent, since no result would yet have been reached (or, indeed, even proposed by the parties). *Consumers Power Co.* (Midland Plant, Units 1 and 2), LBP-83-50, 18 NRC 242, 247-49 (1983). Here, with the record closed on the portion of the proceeding with respect to which information is being proffered, and with proposed findings on the question already submitted, it is appropriate to consider (in the context of the materiality or significance of the information in question) whether the additional information would alter the result we would reach in its absence.

As long ago as March 1981, prior to the commencement of evidentiary hearings, CCANP sought information through discovery concerning the OIA investigation, and we denied such discovery because of questions as to our authority to obtain information from an NRC office which reports directly to the Commission and is not technically involved in licensing (Tr. 707-13). Even though CCANP may have become aware of some of the information concerning which it seeks to reopen the record prior to June 1983 (when it states it first became aware that the OIA report had been released), we note that the significance of the information in question would likely be different if only isolated segments of that information were relied upon. We also note that, although the OIA report is dated October 10, 1980 and was apparently released to the public in March 1983, neither the Board nor the parties were apparently made aware of such release. Additionally, the approximately 6 weeks' elapsed time between CCANP's discovery of the OIA file and the filing of its motion to reopen the record is not excessive, considering the efforts required to locate and obtain certain documents referenced in the OIA file. We agree with the Staff that the motion to reopen should not be considered as untimely. We regard the timeliness criterion for reopening to have been satisfied.

Furthermore, no party contests the significance of the issues to which the OIA report relates. Indeed, the questions of alleged harassment and intimidation of QC inspectors and of alleged document falsification are central to many of the matters considered in this Decision. The criterion for reopening which requires that the issue be significant is thus also satisfied.

2. With respect to the significance and materiality of the OIA report and related documents to the questions before us, we must evaluate the information as to which CCANP seeks to reopen the record in light of the record as it currently stands. There is no question that, in the course of our taking evidence with regard to the allegations of harassment and intimidation of QC inspectors, and with respect to alleged document falsification, the information now proffered by CCANP would have been both relevant and, for the most part, material and not unduly repetitious. *See* 10 C.F.R. § 2.743(c). At this stage, however, there is much evidence in the record on these topics. What might have been material at one point in this proceeding may no longer be so.

The first of the three categories of information for which CCANP seeks to reopen the record is identified as the concerns which led OIA to conduct its investigation. CCANP characterizes these documents as "useful" (motion at 4, 6). If we were to admit the OIA report into the evidentiary record, we agree with CCANP that the information concerning how and why the report was initiated would be relevant and perhaps useful as background to establish the context in which the information in the report is to be viewed. But the documents in question (included as a part of Exhibit 2 of CCANP's motion) do not appear in themselves to include factual information material to the issues or contentions before us. They therefore do not constitute significant and material factual information necessary for the record to be reopened. *Diablo Canyon,* ALAB-644, *supra.*

The second category of information for which CCANP seeks to reopen the record concerns the interaction between OIA and the Department of Justice. CCANP relies in particular on letters from the Department of Justice to HL&P and B&R (included in motion, Exh. 2) in which the Department assertedly concluded that "criminal violations were committed at STNP" and that these acts were "merely symptomatic of an overall pattern of neglect" on the part of HL&P and B&R; the Department also observed that HL&P and B&R were "on notice that any further such violations would be attributed to the two companies and their responsible officers" (motion at 4). CCANP claims that these letters establish a level of seriousness of the QA violations not previously documented and, as such, are relevant to the question of HL&P's character (Issue A) (motion at 6).

As pointed out by the Applicants, the Department of Justice opinions related to document falsification, and concerned the actions of lower level employees of B&R or its subcontractors. Although not a part of the basis for our ruling, we note that the very same documents recognize HL&P's efforts to rectify the Department's concerns with respect to safety, compliance and reporting problems (letter from Julian Greenspun, DOJ, to Earl J. Silbert, dated June 19, 1981, included in motion, Exh. 2).

We have dealt extensively with the allegations of document falsification and harassment or intimidation of QC inspectors elsewhere in this Opinion, and we have reached our own conclusions with respect to them. The conclusions of the Department of Justice (or of OIA) on these questions in any event could not be given much weight by us, at least in the absence of testimony by DOJ (or OIA) representatives as to the basis for their conclusions. *See Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit No. 1), ALAB-738, 18 NRC 177, 195 (1983). Moreover, as spelled out in more detail below, the conclusions of the DOJ (and of OIA) apparently are derived from the OIA report which, in turn, is based in large part on factual events which are already covered in the record of this proceeding, upon which our own conclusions are based. For those reasons, the opinions of the Department of Justice (or of OIA) would not constitute the type of factual information necessary to warrant a reopening of the record. *Diablo Canyon,* ALAB-644, *supra.*

The final and most important category of information which CCANP seeks to add to the record is the OIA report itself (motion, Exh. 2). That report, dated October 10, 1980, generally covers many of the same allegations as were dealt with in I&E Report 79-19 (Staff Exh. 46, Appendix D). OIA investigators interviewed various Region IV personnel — Messrs. Karl Seyfrit, William G. Hubacek, William Crossman, William Seidle, and Ramon Hall. All of these individuals with the exception of Mr. Seyfrit (Regional Director during the period covered by the OIA investigation) appeared as witnesses in this proceeding. OIA investigators also interviewed a number of B&R QC inspectors (including former B&R QC personnel), and seven executives, managers and supervisors of HL&P and B&R. The version of the OIA report which we received from CCANP does not identify these persons, but it appears that at least some of them have testified during the Phase I hearings.[44]

The OIA report reflects that, in addition to I&E Report 79-19, OIA reviewed I&E reports containing allegations received by NRC from May 1978 through March 1979 depicting program deficiencies, as well as selected earlier reports which included similar allegations. All of the I&E reports upon which the OIA report relied (which are attached to the OIA report) are also included in the record of this proceeding.[45] Indeed, for that reason, CCANP does not include them in its motion to reopen the record (motion at 2).

In addition, the OIA report reviewed the FBI report of investigation, the transcript of an October 1979 CBS television report on the STP, as well as an audio tape in which an unidentified former B&R employee made certain allegations concerning alleged deficiencies at the STP to a representative of one of the Intervenors in this proceeding. It is our

[44] The OIA report has excised from it the names of all individuals interviewed other than NRC personnel, together with information which might tend to identify the anonymous individuals.

[45] I&E Reports (Docket Numbers 50-498 and/or 50-499) 77-03 (Staff Exh. 1), 77-07 (CCANP Exh. 7), 77-08 (Staff Exh. 4), 78-04 (CCANP Exh. 9), 78-07 (CCANP Exh. 10), 78-09 (Staff Exh. 7), 78-12 (Staff Exh. 8), 78-13 (Staff Exh. 9), 78-14 (Staff Exh. 12), 78-15 (Staff Exh. 13), 78-16 (Staff Exh. 11), 78-17 (Staff Exh. 16), 79-01 (Staff Exh. 17), 79-03 (CCANP Exh. 12), 79-04 (Staff Exh. 20), 79-05 (Staff Exh. 23), 79-09 (Staff Exh. 26), 79-12 (Staff Exh. 22), 79-13 (Staff Exh. 27), and 79-14 (Staff Exh. 32). Two additional I&E reports (77-01 and 79-11) are also listed in the index to the OIA report but have not been offered (or admitted) into evidence in this proceeding. The OIA report makes no specific reference to the substance of these two I&E reports.

belief that this tape is one of those about which we had extended discussions with CCANP earlier in this proceeding (*e.g.*, Tr. 430-36, 475, 535-43).

As we have earlier stressed, reopening of the record would depend in large part on whether CCANP was proffering significant and material information not yet a part of the record. Most if not all of the information in the OIA report or its attachments is apparently already in the Phase I record. Although CCANP does not precisely define which documents or information in the OIA report it would seek to introduce into evidence, it mentions five items as being "of importance" (motion at 4-5). Three of them (numbers 1, 2, and 4) are various OIA conclusions, all based on other material in the report. For reasons previously outlined, these conclusions are not the type of factual information for which a reopening of the record would be warranted.

Item number 3 to which CCANP refers represents the OIA reports of interviews with personnel who assertedly provided information beyond that contained in I&E Report 79-19. CCANP cites observations by two QC inspectors concerning the morale of QC inspectors and the practice of not performing required inspections. While we cannot pinpoint the exact source of these observations (because of the anonymity policies of OIA), it appears that these allegations are comparable to those already in the record and dealt with in this Decision (*see* Findings 64, 66, 69, 74-75, 122-123, 194, 381-398).

The last item in the OIA report to which CCANP refers is an apparent altercation over quality between a QC inspector and a project engineer which occurred on March 7, 1979. The Applicants acknowledge that the specific allegation "may not have been addressed in Phase I" (response at 8). In contrast, the Staff characterizes this item as the only instance where CCANP is pointing to new factual evidence but claims that the fight between the inspector and the project engineer was in fact included in the testimony of record (Staff response at 7, *citing* Warnick, *et al.*, ff. Tr. 8032, at 13-14, 33-34, and Staff Exh. 20 (I&E Rept. 79-04, at 7)). We agree with the Staff that the asserted altercation was indeed the subject of Phase I testimony, and we have dealt with that fight in Finding 387, *infra*.

In sum, none of the items in the OIA report to which CCANP refers constitutes new significant, material factual information. The information from the report cited by CCANP would not be sufficient to warrant a reopening of the record.

D. We have considered the items of information referenced by CCANP not only individually but collectively to determine whether the record should be reopened. We have concluded that the information at this stage would either be cumulative or not of a type which could lead us to reopen the record. We are therefore denying CCANP's motion to reopen the Phase I record. Nor do we view the particular information, in itself, as sufficiently significant or material to warrant including it in a later phase of this proceeding. In that regard, the information relates almost exclusively to B&R's performance of its contractor services. We disagree with CCANP's conclusion that, at least insofar as HL&P is concerned, the information could change our view of HL&P's character. Although the information might bolster our view of HL&P's former questionable managerial competence, it would not do so to a degree which would offset our view that HL&P has taken adequate steps to improve its competence.

At an earlier stage of Phase I, however, the information would have been useful. At least it would have circumscribed the quantity of then-undiscovered information about which all parties expressed an interest and thus could have served to remove some of the uncertainties concerning the testimony which was adduced. We regret that the Commission did not see fit to release the OIA report at an earlier date, when it could have proved useful in the litigation of certain issues before us.

## V. CONCLUSION

In conclusion, we perceive this project as one which, although in trouble at an earlier date, has now likely "turned the corner." The STP QA/QC program had serious implementation problems prior to the issuance of I&E Report 79-19 and the accompanying Order to Show Cause and Notice of Violation. The problems were traceable, in our opinion, not to character defects on the part of HL&P but, rather, to deficiencies in two key ingredients of competence: experience with respect to design and construction of large nuclear facilities such as STP on the part of both HL&P and B&R, and long lines of communication which resulted in HL&P's inexperienced management failing to receive the information it required to operate a successful QA/QC program. Indeed, with both HL&P and B&R lacking the necessary experience, the result was a synergistic magnification of the problems which resulted.

The changes effectuated by HL&P in our view have alleviated the communications problem and have the likelihood of greatly enhancing the experience to be made available to the STP.[46] They accordingly

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[46] In this respect, we regard the record on corrective actions here to be considerably more persuasive than the Licensing Board in the *Byron* proceeding apparently fouhd to be the case there. *See Commonwealth Edison Co.* (Byron Nuclear Power Station, Units 1 and 2), LBP-84-2, 19 NRC 36 (1984).

should result in upgraded QA/QC performance at STP. Because the record does not include much information on project performance with the referenced changes, this assessment can only be preliminary at this time. The report which we are directing to be made in Phase II will likely make clear whether our preliminary expectations in fact prove realistic.

At this point, we wish to reiterate that the Intervenors in this proceeding raised serious QA/QC issues, both in submitting well-based contentions for litigation and in requesting a hearing on the Order to Show Cause — a request which, although denied by the Commission, resulted in the inclusion in this proceeding of additional important matters. They initially raised these issues long before the NRC recognized their significance and for this they are to be commended.

On the other hand, we must also observe that dealing with CCANP's claims presented considerable difficulty, for several reasons. In the first place, CCANP conducted lengthy cross-examination on technical issues but filed few proposed findings with respect to the technical information it elicited — even where that information appeared to support certain of its claims. Given the extensive amount of time utilized by such cross-examination, we would have expected CCANP to provide us with greater assistance in ascertaining the import of the information, particularly where it related to issues upon which CCANP filed proposed findings. In determining the extent of cross-examination to be permitted in later phases of this proceeding, we will consider carefully the probable productivity of such cross-examination.

More important, although we permitted — indeed encouraged — CCANP to select the issues which it wished to stress, we were somewhat dismayed to see CCANP in its proposed findings attempting to focus all the information in the proceeding upon which it was relying solely on the character aspect of Issue A, even where the information bears little if any relationship to HL&P's character and considerable relevance to other issues such as HL&P's competence.

Finally, as we indicated earlier, CCANP's failure to indicate any deficiencies it perceived in HL&P's various corrective actions may have left a gap in the record. We recognize CCANP's general legal position that reformation or corrective action is not possible and should not have been considered, at least in Phase I. Taking into account our early rejection of this position, and our acceptance of Issue B as a Phase I issue, it would have been helpful if CCANP had explained why it believed particular corrective actions were ineffective or inadequate (assuming that to be its position).

Notwithstanding the difficulties presented by CCANP's manner of presenting its claims to us, we have reviewed with great care the entire record of this proceeding, including the proposed findings of fact and conclusions of law submitted by the various parties. Based on the foregoing Opinion, the Findings and Conclusions on which it relies, and this entire record, it is our opinion that HL&P is not now deficient in character and has not demonstrated character deficiencies which would warrant denial of operating licenses; and that HL&P's competence, while questionable prior to the Staff's 79-19 Investigation and the issuance of the Show-Cause Order and Notice of Violation, was not so deficient as to preclude, without more, the award of operating licenses. Moreover, that competence appears to have substantially improved. We now have reasonable assurance that structures which are complete and work which has been performed comply with applicable regulatory requirements, and that future work activities (including implementation of the QA/QC program for construction) will be carried out satisfactorily. These conclusions are, of course, subject to the outcome of later phases of this proceeding, particularly the Phase II report we are directing under Issue B, p. 697, *supra*.

This Opinion is based upon, and incorporates, the Findings of Fact and Conclusions of Law which follow. Any proposed findings or conclusions submitted by the parties which are not incorporated directly or inferentially in this Partial Initial Decision are rejected as being unsupportable in law or in fact or as being unnecessary to the rendering of our Decision.

## Findings of Fact and Conclusions of Law

### I. FINDINGS OF FACT

#### A. Jurisdiction and Parties

1. This Partial Initial Decision involves the application of Houston Lighting and Power Company (HL&P, the project manager), the City of San Antonio, Central Power and Light Company, and the City of Austin, for licenses to operate the South Texas Project, Units 1 and 2 (STP).

2. The STP is located approximately 15 miles southwest of Bay City, on the west side of the Colorado River, in Matagorda County, Texas. The plant will be comprised of two pressurized water reactors,

each with a rated core power level of 3800 megawatts thermal and a net electrical output of 1250 megawatts.

3. Construction permits for the STP were issued by the Nuclear Regulatory Commission (NRC) on December 22, 1975. 41 Fed. Reg. 831 (1976). HL&P hired Brown and Root, Inc. (B&R) as architect-engineer, construction manager, and constructor.

4. On August 2, 1978, the NRC published a "Notice of Receipt of Application for Facility Operating Licenses"; of the availability of the Final Safety Analysis Report and Environmental Report; and of the "Consideration of Issuance of Facility Operating Licenses, and Opportunity for Hearing." 43 Fed. Reg. 33,968 (1978).

5. Five petitioners sought intervention, and two were admitted as parties: Citizens for Equitable Utilities, Inc. (CEU) and Citizens Concerned About Nuclear Power, Inc. (CCANP). In addition, the State of Texas was admitted as an interested State. LBP-79-10, 9 NRC 439 (1979). CEU withdrew from this proceeding on June 15, 1982, subject to certain conditions (Tr. 10,384). *See* our Memorandum dated June 24, 1982 (unpublished).

6. Eight contentions (some with multiple subparts) were admitted. CEU and CCANP jointly sponsored Contentions 1 and 2; CCANP was the sole sponsor of Contention 3; and CEU was the sole sponsor of Contentions 4 through 8. Memorandum and Order dated August 3, 1979 (unpublished). Contention 1 was later supplemented. Fourth Prehearing Conference Order dated December 16, 1981 (unpublished). After CEU's withdrawal from this proceeding, we granted CCANP's request to adopt Contention 4; we denied its request to adopt Contentions 5 through 8 and dismissed those contentions. Memorandum and Order dated October 15, 1982, LBP-82-91, 16 NRC 1364.

7. On April 30, 1980, the NRC Office of Inspection and Enforcement (I&E) issued I&E Report 79-19 (Staff Exh. 46, Appendix D), identifying twenty-two noncompliances in STP construction and the STP construction Quality Assurance/Quality Control (QA/QC) program. The Director of I&E also issued a Notice of Violation and an Order to Show Cause why construction of the STP should not be stopped pending certain actions. In addition, a civil penalty of $100,000 was proposed as a result of the items of noncompliance found in I&E Report 79-19 (Staff Exh. 46, Appendices A and B; *see also* Finding 68, *infra*). By letters dated May 23, 1980, HL&P admitted the validity of most of the findings of Investigation Report 79-19 and acknowledged and paid the civil penalty of $100,000 (Staff Exhs. 47, 90). On July 28, 1980, HL&P responded to the Show-Cause Order and committed to satisfying the requirements of that order (Staff Exh. 48). During the course of this hearing, we have

considered in great detail the methods selected by HL&P to meet those requirements.

8. On May 28, 1980, Intervenors CCANP and CEU filed with the Commission requests for a hearing on the Show-Cause Order. Earlier, by Memorandum dated March 10, 1980 (unpublished), we had proposed that our hearing on Contentions 1 and 2 (which involved QA/QC matters of the same sort as later became the subject of the Show-Cause Order) be held on an expedited basis "so that, if corrective action is required, it may be undertaken as early as possible in the construction schedule." We reiterated that view in our Memorandum of August 1, 1980 (unpublished). By Memorandum and Order dated September 22, 1980, the Commission denied the requests for a hearing on the Show-Cause Order but agreed with our previously expressed intent to expedite a hearing on QA/QC issues. The Commission also directed us to consider the "broader ramifications" of charges relating to HL&P's "competence and character." CLI-80-32, 12 NRC 281, 291-92 (1980).

9. By our Second Prehearing Conference Order, dated December 2, 1980 (unpublished), we articulated six issues (Issues A through F) addressing the Commission's concerns set forth in CLI-80-32. We denominated the early resolution of CLI-80-32 Issues A through E and Intervenor Contentions 1 and 2 as Phase I of our proceeding. The Phase I hearing was noticed in the *Federal Register* on April 9, 1981. 46 Fed. Reg. 21,289 (1981). On April 13, 1981, the Staff filed a Partial SER (NUREG-0780), assessing HL&P's management capability and the QA program for operations. Phase I evidentiary hearings commenced on May 12, 1981.

10. On September 24, 1981, the Applicants informed us that B&R had been dismissed as architect-engineer and construction manager. Bechtel Power Corporation (Bechtel) replaced B&R in those capacities. The Applicants further informed us, on September 28, 1981, that a report on B&R engineering had been prepared for HL&P by the Quadrex Corporation (the Quadrex Report). Later, on November 5, 1981, the Applicants advised us that they were unable to reach agreement with B&R for B&R to continue as constructor. On February 16, 1982, the Applicants notified us of the selection of Ebasco Services, Inc. (Ebasco) to replace B&R as project constructor.

11. At a prehearing conference on December 8, 1981, we divided the hearing into three phases (rather than the two phases which had previously been contemplated). Phase I comprised the topics previously included in the first phase, plus certain issues arising from the transition from B&R to Bechtel and Ebasco. The Board also admitted four further subparts to Contention 1 (Contentions 1.8(a)-(d)). Phase II is to address

the Quadrex Report (including its effect, if any, on determinations reached in Phase I) and Contention 4 (concerning hurricanes). It will also include the report we are directing under CLI-80-32 Issue B, p. 697, *supra*. (By Memorandum and Order dated July 14, 1983, we denied CCANP's motion to add a financial qualifications contention to Phase II. LBP-83-37, 18 NRC 52, *reconsideration denied*, LBP-83-49, 18 NRC 239 (1983).) Phase III will address CLI-80-32 Issue F (operations QA), Contention 3 (overpressurization), and any remaining issues. Fourth Prehearing Conference Order, dated December 16, 1981 (unpublished).

12. Evidentiary hearings on Phase I were held during the weeks of May 12 and 18, June 1, 15 and 22, July 20, and September 14, 1981, and January 19, February 9, and June 15, 1982. The record was closed on June 17, 1982 (Tr. 10,722). (By Memorandum and Order dated January 10, 1983 (unpublished), we denied a motion by CCANP to reopen the Phase I record. In Part IV of the Opinion section of this Decision, *supra*, we are denying another such motion by CCANP.)

### B. Findings on CLI-80-32 Issues

#### *Issue A: HL&P's Managerial Character and Competence*

13. Issue A states:

> If viewed without regard to the remedial steps taken by HL&P, would the record of HL&P's compliance with NRC requirements, including:
>
> (1) the statements in the FSAR referred to in Section V.A(10) of the Order to Show Cause;
>
> (2) the instances of noncompliance set forth in the Notice of Violation and the Order to Show Cause;
>
> (3) the extent to which HL&P abdicated responsibility for construction of the South Texas Project (STP) to Brown & Root; and
>
> (4) the extent to which HL&P failed to keep itself knowledgeable about necessary construction activities at STP,
>
> be sufficient to determine that HL&P does not have the necessary managerial competence or character to be granted licenses to operate the STP?

Because of the segmentation of this issue into four subparts, our findings will treat each of the subparts *seriatim* and, thereafter, include our general findings with respect to the managerial character and competence questions which permeate the entire issue.

*(1) Alleged False Statements in the FSAR*

14. Sections III and V.A(10) of the Order to Show Cause, referenced in the first subpart of this Issue (*see* Finding 13), indicate that there were "apparent false statements" in Sections 2.5.4.5.6.2.4 and 2.5.4.5.6.2.5 of the Final Safety Analysis Report (FSAR) (Staff Exh. 46, Order to Show Cause at 11, 17).

15. The second and third paragraphs of Section 2.5.4.5.6.2.4 in the May 1978 FSAR require conduct of certain soil tests during backfill operations, as follows:

> At least one relative density test (ASTM D 2049) and one gradation test (ASTM D 422) were performed for every fourth field test to ensure compatibility between field and laboratory tests.
>
> Whenever fill or backfill was placed during a work shift, at least one field test and one laboratory relative density test were conducted during the shift, provided that the compaction operation was completed in some area.

A review of data from the testing laboratory on December 8, 1979, revealed that equipment failure had prevented the conduct of any relative density tests since November 17, 1979. The equipment was replaced on January 7, 1980. During the entire period, backfill operations continued and several sets of field tests were conducted without performing the specified relative density tests, contrary to the FSAR requirement. Further, testing laboratory personnel failed to document and correct this nonconforming condition. This is the basis for Noncompliance No. 3 as set forth in the April 30, 1980 Notice of Violation. FSAR Section 2.5.4.5.6.2.4; Staff Exh. 46, Appendix D (I&E Rept. 79-19, at 64); *id.*, Appendix A (Notice of Violation at 6); Shewmaker, *et al.*, ff. Tr. 9576, at 14-24, 48; Pettersson/White, ff. Tr. 6162, at 9-10; Staff Exh. 48, at 2-33 to 2-36.

16. The first paragraph of § 2.5.4.5.6.2.5 in the May 1978 FSAR requires inspection during placement of backfill:

> The testing agency provided continuous inspection of the placement of all backfill material and tested the material in the field for degree of compaction. The inspectors observed the type of material, lift thickness, operation of compaction equipment, and all other pertinent material or construction conditions affecting the quality of work and compliance with the specifications. The frequency of testing and selection of test locations for placed material were according to the requirements identified in these six categories: * * *

Also, 10 C.F.R. Part 50, Appendix B, Criterion XVII requires that "[s]ufficient records shall be maintained to furnish evidence of activities affecting quality." In I&E Investigation Report 79-19, Allegation 3d indicates that the lift thickness and number of passes of compaction equipment were not documented (Staff Exh. 46, Appendix D, at 64-65). That information was needed to assure systematic placement and compaction of backfill material. This missing documentation is listed as Noncompliance No. 5 in the April 30, 1980 Notice of Violation. Staff Exh. 46, Appendix A, at 7; Shewmaker, *et al.*, ff. Tr. 9576, at 21, 25, 48; Pettersson/White, ff. Tr. 6162, at 10-11; Crossman, *et al.*, ff. Tr. 10,010, at 18; Tr. 9921-25 (Hayes, Shewmaker).

17. Based on information in I&E Report 79-19, the Order to Show Cause (Staff Exh. 46, at 11) summarized this situation in the following statement:

> During the review of backfill installation and testing activities two apparent false statements in the FSAR were identified regarding test and observation work actually performed. (Sections 2.5.4.5.6.2.4 and 2.5.4.5.6.2.5.)

Subsequently, in Section V.A(10) of the Order to Show Cause (*id.* at 17), HL&P was in effect directed, *inter alia,* to

> verify or correct if necessary, the FSAR statements contained in Section 2.5.4, Stability of Surface Materials, especially Section 2.5.4.5, Excavations and Backfill.

18. HL&P supplied direct testimony on this question through a panel consisting of C. Bernt Pettersson, B&R Assistant Project Civil Engineer for STP, and Jon G. White, HL&P Licensing and Technical Coordinator for STP. Mr. Pettersson had been in charge of soils work on the STP and was responsible for developing the FSAR section in question. The panel described how the FSAR sections were prepared and reviewed to assure consistency with design documents and compliance with regulatory requirements. They also described studies conducted in response to the Show-Cause Order, their findings, and the basis for and nature of subsequent changes made in the FSAR. Pettersson/White, ff. Tr. 6162.

19. Staff testimony on this issue was given by a panel consisting of Robert E. Shewmaker, H. Shannon Phillips and D.W. Hayes (Shewmaker, *et al.*, ff. Tr. 9576). During the period referred to in the testimony, Mr. Shewmaker had been Senior Structural Engineer, Division of Reactor Construction Inspection, Office of Inspection and Enforcement. He served as headquarters liaison for the special investigation that culminated in preparation of I&E Report 79-19, the Notice of Violation, the Order to Show Cause, and the Proposed Imposition of Civil Penalties (Staff Exh. 46). Mr. Phillips was the Region IV Resident

Reactor Inspector at STP from August 1979 until 1982 and served as a member of the special investigating team. He has had over 12 years' experience in quality-assurance-related activities. Mr. Hayes was Chief, Engineering Support Section I, Reactor Construction and Engineering Support Branch, Office of Inspection and Enforcement, Region III, and was assigned to head the special investigating team. He has had reactor inspection experience throughout his employment with AEC/NRC since 1970 and has had other nuclear experience since 1948. Shewmaker, *et al.*, ff. Tr. 9576, at 1-2 and attached Professional Qualifications. The panel discussed concerns about STP construction identified during the special investigation, summarized the enforcement action, addressed questions raised in CLI-80-32 Issue A, as well as responding to other issues and contentions before the Board.

Another NRC Staff panel discussed I&E activity after the Order to Show Cause, including matters related to this Issue. This panel consisted of William A. Crossman, Ramon E. Hall, William G. Hubacek, H. Shannon Phillips, Dan Paul Tomlinson, J.I. Tapia, and Richard K. Herr. Mr. Crossman was Chief, Section 3, Reactor Projects Branch, Office of Inspection and Enforcement, Region IV, and supervised the personnel inspecting nuclear power plants in Region IV, including STP. Mr. Hall was Chief, Systems and Technical Section, Office of Inspection and Enforcement, Region IV, and was responsible for supervision of engineering specialist inspectors in Region IV, including STP. Mr. Hubacek is currently retired. During periods relevant to this proceeding he was a Reactor Inspector, Office of Inspection and Enforcement, Region IV, and was responsible for project inspection of facilities under construction in Region IV, including STP. Mr. Phillips' position and experience are described above. Messrs. Tomlinson and Tapia were Reactor Inspectors, Engineering and Materials Section, Office of Inspection and Enforcement, Region IV. Mr. Herr was a Senior Investigator, Region IV. Crossman, *et al.*, ff. Tr. 10,010, at 1-3; Tr. 9997 (Herr, Hubacek).

20. The FSAR was issued in May 1978 (Tr. 6207 (White)). Accordingly, the statements therein, including the subsections forming the bases for Noncompliances Nos. 3 and 5 of the Notice of Violation, were written many months before the observations in I&E Report 79-19, in late 1979, that led to concern about the possible "false statements." Tr. 6207-10 (White, Pettersson).

21. HL&P asserts that the FSAR statements concerning relative density tests were true when written and that the incident set forth in Noncompliance No. 3 should be viewed as an instance where personnel did not adhere to specified procedures in field activities (a nonconformance), instead of a false statement suggesting intent to

deceive. HL&P concluded that, in the absence of the Order to Show Cause and Notice of Violation, the natural course of events would have been to report this as a nonconformance instead of changing the FSAR. Pettersson/White, ff. Tr. 6162, at 9-10; Tr. 6188-91, 6205, 6208-10, 6216 (Pettersson, White).

22. During the approximately 2 months in which the vibratory equipment required for the relative density test was not available, the samples to be used in those tests were collected and saved. Those samples subsequently were tested when the defective equipment had been replaced and it was found that the required relative densities had been met in the backfill. Technically, this is proper since the test results are not affected by the passage of time. During an inspection on June 23-26, 1980, it was verified that a backup vibratory head and spare mold for measuring relative density were available on site and Nonconformance No. 3 was closed. Pettersson/White, ff. Tr. 6162, at 10; Staff Exh. 46 (I&E Rept. 79-19, at 64); Shewmaker, *et al.*, ff. Tr. 9576, at 20; Tr. 9929-30 (Shewmaker); Pettersson, *et al.*, ff. Tr. 5796, at 25; Staff Exh. 63 (I&E Rept. 80-17, at 4).

23. The Applicants and Staff concluded that there was no safety significance in the fact that no relative density tests were performed between November 17, 1979, and January 7, 1980. Pettersson/White, ff. Tr. 6162, at 15-16; Crossman, *et al.*, ff. Tr. 10,010, at 11. No evidence to the contrary was presented.

24. The Staff viewed the principal significance of the gap in relative density determinations to be in the possibility that it could reflect a false statement. That possibility required evaluation because of its potential effect on NRC's confidence in HL&P's reliability and truthfulness. Accordingly, the Staff directed HL&P to verify or correct the FSAR statements in Section 2.5.4. That same possibility constitutes the basis for the Commission's directions to this Board, in CLI-80-32, to review and evaluate this matter to determine whether material false statements had been made in the FSAR. Upon completion of the investigations, the Staff concluded that the FSAR statements were not false. Crossman, *et al.*, ff. Tr. 10,010, at 12; Tr. 10,040 (Tapia); Tr. 9862-63 (Shewmaker).

25. The only evidence to the contrary was the Applicants' admission that certain instances of deviation from FSAR requirements in fact occurred prior to the May 1978 issuance of the FSAR. The Applicants explained that they had not become aware of the deviations until their in-depth review was undertaken in May/June 1980, approximately 2 years later. Tr. 6209-10 (Pettersson). We find this explanation to be credible and not to undermine the basic thrust of the Staff's conclusion as to the truthfulness of the FSAR statements when made.

26. As a result of its assessment, the NRC Staff permitted HL&P to revise the FSAR statements. The necessary revisions appeared in FSAR Amendments 12, 17 and 18, submitted on September 15, 1980, April 14, 1981, and May 1, 1981, respectively. FSAR § 2.5.4.5.6.2.4, at 2.5.4-56 (May 1, 1981); Crossman, *et al.*, ff. Tr. 10,010, at 13.

27. The laboratory relative density tests for the structural backfill had been performed, on the average, at least once for each four field tests and HL&P engineers and consultants determined that such frequency was adequate. Accordingly, the second paragraph of FSAR § 2.5.4.5.6.2.4 was amended to broaden the FSAR criteria from one "for every fourth field test * * *" to one "on the average for every four field tests * * *." The changes make the FSAR conform with actual practices in the field, causing it to read as follows:

> One relative density test (ASTM D 2049) and one gradation test (ASTM D 422) were performed on the average for every four field tests in the palnt [sic] area to ensure compatibility between field and laboratory tests, for structural backfill in the plant area and for the ECW structures. For the ECW piping backfill, including the backfill immediately around the ECW piping within the plant area and at the ECW structures, a minimum of one relative density test and one gradation test was performed on the average for every seven field tests.

Shewmaker, *et al.*, ff. Tr. 9576, at 19-20; Pettersson/White, ff. Tr. 6162, at 14; Wilson/Kirkland, ff. Tr. 2697, at 14-15, 19-22; Staff Exh. 48, at p. 2-36; FSAR § 2.5.4.5.6.2.4, at 2.5.4-56 (May 1, 1981).

28. FSAR Section 2.5.4.5.6.2.4 also was revised to change the time limit on testing samples. The change was necessary because, for samples taken near the end of a shift, there could be too little time to return to the laboratory and conduct the test during that shift, as required by the 1978 FSAR. The new statement requires, instead, that in each shift in which work is completed the field test must be conducted and the sample for the laboratory test collected. The laboratory work on that sample then could proceed in some following shift. The third paragraph of that subsection now reads:

> Whenever fill or backfill was placed during a work shift, at least one field test was conducted during the shift and a sample for laboratory relative density testing was obtained, provided that the compaction was completed in some area.

Tr. 6074-75, 6124-25, 6196-98, 6202-03 (Pettersson); FSAR § 2.5.4.5.6.2.4, at 2.5.4-56 (May 1, 1981); Staff Exh. 48, at p. 2-36.

29. Item of Noncompliance No. 5 was based on lack of documentation of the lift thicknesses and number of passes with compaction equipment during backfill operations, which was interpreted as violating the

"continuous inspection" requirement of 1978 FSAR § 2.5.4.5.6.2.5 (Pettersson/White, ff. Tr. 6162, at 10-11). In a July 1980 inspection, it was verified that QA procedures had been changed to require documentation of loose lift thickness and number of roller passes, as well as the roller pattern used. Based on this modification, Item of Noncompliance No. 5 was closed. Staff Exh. 65 (I&E Rept. 80-19, at 2-3); Crossman, *et al.*, ff. Tr. 10,010, at 18; Pettersson, *et al.*, ff. Tr. 5796, at 25.

30. Section 2.5.4.5.6.2.5 of the FSAR subsequently was modified in Amendments 12 and 17 to clarify the inspection procedure and read as follows:

> The testing agency provided QC inspection of the backfill, the placement and testing of the material in the field for degree of compaction. The QC inspectors observed the type of material, lift thicknesses, operation of compaction equipment, and all other pertinent material or construction conditions affecting the quality of work and compliance with the specifications. The QC inspectors noted conformance with the limiting criteria of the specification and construction procedure for structural backfill and reported the acceptability of the operation. The frequency of testing and selection of test locations for placed material were according to the requirements identified in these seven categories: * * *

Shewmaker, *et al.*, ff. Tr. 9576, at 21; Pettersson/White, ff. Tr. 6162, at 14; FSAR § 2.5.4.5.6.2.5, at 2.5.4-56a (Sept. 15, 1980) and 2.5.4-57 (April 14, 1981).

31. Item V.A(10) of the Order to Show Cause (Staff Exh. 46) required verification or correction of the statements in FSAR Section 2.5.4. The HL&P review, and Staff concurrence, resulted in changes in the two FSAR subsections, as indicated. The rest of the Section specified in the Order to Show Cause was verified to be correct as stated originally. Item V.A(10) was closed out during NRC Inspection 80-17. Staff Exh. 113 (I&E Rept. 81-16, at 6); Crossman, *et al.*, ff. Tr. 10,010, at 13, 50.

32. The question of "false statements" in the FSAR has been examined thoroughly in extensive direct evidence presented by the Applicants and Staff, as well as several hundred pages of transcript covering examination of witnesses by all parties and the Board. It has been the Applicants' position that the matter should be viewed as a nonconformance with specified procedures, rather than false statements in the FSAR that had been prepared about 18 months earlier. Testimony of the Staff is consistent with that position. The Staff found no question of HL&P trying to confuse anybody or any willfulness on HL&P's part to include anything in the application that subsequently was not done in the field. It was simply a matter of failing to conform with two out of a

very large number of QA/QC control documents. Tr. 9862-63 (Shewmaker). The one deviation was unintended (Finding 25). No witness for either the Applicants or Staff differed with that position, either in direct testimony or in response to cross-examination, and no evidence to the contrary was developed by the Intervenors. *See* Findings 15-31, above and citations therein.

33. The Board finds that the FSAR statements that were the origins for concern expressed in the Order to Show Cause and CLI-80-32 were not generally false when made. We do not regard the single deviation mentioned in Finding 25 to be significant or to detract from our general finding of lack of falsity. There is no evidence that there was either intent by the Applicants to deceive the Commission or disregard for the truth.

34. We find that the incidents and remedial actions in connection with subpart A(1) of CLI-80-32 Issue A do not reflect adversely upon HL&P's character and, while pertinent to HL&P's or B&R's competence, are not sufficient to determine that HL&P does not have the necessary managerial competence to be granted licenses to operate the STP.

*(2) Noncompliances Set Forth in Notice of Violation and Show-Cause Order*

35. Issue A(2) asks whether HL&P's compliance record with NRC requirements, when considered alone and without regard to remedial steps, would be sufficient to determine that the company lacks the managerial competence or character to be granted operating licenses. *See* Finding 13, *supra*. We consider that, to be meaningful, our evaluation must include consideration of the number and severity of violations, any significant patterns that may exist in the occurrences, prior knowledge or involvement of management in them, ability of management to learn from those experiences, the willingness and attitude of HL&P officials in responding to NRC observations and enforcement actions, and the promptness and nature of those responses. All of those aspects relate to the ability of management to deal with problems inherent in constructing and operating a nuclear power plant and can reflect its integrity in dealing with NRC, including truthfulness, candor, commitment to safety, and willingness to shoulder its responsibilities as a licensee. Additional information in those regards can be obtained by examining the details and effectiveness of the specific remedial actions actually implemented, which will be considered to the extent pertinent in our consideration of Issue B.

36. The Notice of Violation and Order to Show Cause review, and are based on, noncompliances that were identified both before and during I&E Investigation 79-19 (Staff Exh. 46). Also, CLI-80-32, *supra,* expresses the Commission's concerns about activities and performance spanning several years of the project (12 NRC at 291). Accordingly, witnesses for HL&P and the NRC Staff, and this Board, have addressed the broad range of noncompliances covering the entire period from initiation of the South Texas Project through completion of Investigation 79-19 and thereafter until December 31, 1981.

37. Don D. Jordan, President and Chief Executive Officer of HL&P, presented direct testimony on the company's commitment to safe construction and operation of the STP and some of the steps taken by its management to assure that STP meets all applicable regulatory requirements. Jordan, ff. Tr. 1223. Mr. Jordan was cross-examined by all parties and the Board. Tr. 1224-1503.

38. HL&P presented direct testimony on company experience in STP construction and actions taken in response to the Order to Show Cause through a panel consisting of George W. Oprea, Jr., Joseph W. Briskin, Richard A. Frazar, and John M. Amaral (Oprea, *et al.,* ff. Tr. 1505). At the Board's request, Mr. Edward A. Turner joined the panel (Tr. 3360). Mr. Oprea, Executive Vice President of HL&P and a member of the HL&P Board of Directors, testified on implementation of the QA program at STP and HL&P management reaction to the Show-Cause Order. Mr. Briskin, the STP Manager, Houston Operations, was responsible for directing work of the engineering, procurement, project control services, accounting and project administration activities located in Houston. He testified on the organization and function of the Task Force, headed by him, that reviewed I&E Report 79-19 and the Show-Cause Order and developed the HL&P responses to it. Mr. Frazar was Manager, South Texas Project Quality Assurance and testified with respect to changes made in the STP administrative controls in response to the Show-Cause Order and changes in the STP QA program before and after issuance of the Show-Cause Order. Mr. Amaral, the Manager of Quality Assurance of Bechtel Power Corporation, described Bechtel's audit and recommendations concerning alternative organizational structures for the STP QA program and the bases for those recommendations. Mr. Turner was Group Vice President, Power Plant Engineering & Construction-Fossil, of HL&P. He was responsible for engineering and construction of all HL&P generating plants, including STP, from 1972 to 1976 and 1978 to 1980. Oprea, *et al.,* ff. Tr. 1505, at 1-5, 52-53, 77, 118-19; Tr. 3382-86 (Turner). There was extensive cross-examination

of the witnesses by all participants and the Board (Tr. 1543-2298, 3360-3630, 5063-5544).

39. HL&P also presented direct testimony on organization, QA activities, management involvement and responses to NRC enforcement actions by B&R, through a panel consisting of Dr. Knox M. Broom, Jr. and Raymond J. Vurpillat (Broom/Vurpillat, ff. Tr. 3646). Dr. Broom was Senior Vice President of the B&R Power Group and Assistant to the Group Vice President. His responsibilities included supervision of the Quality Assurance (QA) Department of the Power Group, which had responsibility for the B&R QA Program for the South Texas Project (STP). Mr. Vurpillat joined the B&R organization in August 1980, as Manager of the Power Group QA Department and was responsible for all QA programs implemented within the Group, including that at STP. At the Board's suggestion, this panel was joined by Mr. Stephen H. Grote, Senior Vice President, Operations, for the B&R Power Group, who was responsible for project management services, including cost engineering, scheduling, estimating, material management and project control, and contracts and proposals for all projects in the Power Group, including STP. From April 1979 until May 1981, he served as B&R executive sponsor of the STP, with responsibility for client communication and accountability above the project level. Broom/Vurpillat, ff. Tr. 3646; Tr. 4341-44 (Grote). There was extensive cross-examination of the panel by all participants and the Board. Tr. 3659-3871, 3894-4108, 4132-5055.

40. Staff direct testimony on activities leading up to the Order to Show Cause was presented by a panel consisting of William C. Seidle, William A. Crossman, William G. Hubacek, Robert G. Taylor and H. Shannon Phillips (Seidle, *et al.*, ff. Tr. 9205). During the period relevant to this testimony, Mr. Seidle was Chief of the Reactor Construction and Engineering Support Branch (RCESB), Region IV, and was responsible for implementing programs of inspection, investigation and enforcement for nuclear power plants in Region IV, including STP. At that time, Mr. Crossman was Chief, Projects Section, RCESB, and was responsible for supervising the project inspectors for plants under construction in Region IV, including STP. Mr. Taylor was a Construction Project Reactor Inspector from 1976 to 1978 and was responsible for inspecting nuclear power plants under construction in Region IV, including STP. Mr. Hubacek's and Mr. Phillips' positions and experience are described in Finding 19, *supra*. Seidle, *et al.*, ff. Tr. 9205, at 1-3 and Statements of Educational and Professional Qualifications. There was extensive cross-examination of this panel by other participants and the Board (Tr. 9208-9561).

41. Staff testimony on I&E Report 79-19 and the Order to Show Cause was presented by the Shewmaker panel, identified earlier in Finding 19, *supra*. Shewmaker, *et al.*, ff. Tr. 9576. Cross-examination of the panel is at Tr. 9578-9980.

42. Staff testimony on inspection and enforcement activities at STP since Investigation 79-19 and the Order to Show Cause was presented by the Crossman panel, all members of which have been identified earlier in Finding 19, *supra* (Crossman, *et al.*, ff. Tr. 10,010). Cross-examination of this panel is at Tr. 10,011-10,119.

43. It is fundamental in the NRC regulatory program that the reactor licensee is fully responsible for designing, constructing, testing and operating its facility in accordance with requirements imposed by the Commission. A vital feature of that program, which is of special importance in these proceedings, is the requirement for the licensee to comply with Quality Assurance Criteria for Nuclear Power Plants set forth in Appendix B to 10 C.F.R. Part 50. Seidle, *et al.*, ff. Tr. 9205, at 6.

44. The quality assurance program described in Appendix B envisions a pyramid control system, the base of which requires detailed inspection and test programs by the licensee or its contractors. All safety-significant actions must be accomplished by craftsmen using approved procedures and verified through up to 100 percent inspection by the licensee's onsite quality control personnel. It is this level of verification of procedures implementation that results in accept/reject decisions on specific items of equipment, construction activities, systems, technician or operator actions and procedures. *Id.* This is the level at which HL&P and B&R quality control inspectors were intended to function.

45. At the next level up in this pyramid system, the licensee must include audits that oversee and test the adequacy of the detailed quality control tests and inspections referred to above. Results of the audit are reported to licensee management, which then makes program corrections when needed and feeds appropriate changes back to the lower level through training or modifications in procedures or other programmatic improvements. This feedback system is designed to assure and enhance the reliability of the program in verifying that all safety-significant actions have been considered and implemented properly. *Id.* at 7. This is the system level at which HL&P and B&R audit programs were intended to function.

46. At the upper level of this system, licensee management must provide adequate organizational independence of QA/QC personnel from construction scheduling and costs, competent and adequate manpower to carry out the quality assurance and quality control programs, and policy guidance to the licensee's and contractors' organization and

personnel. *Id.* This is the level at which HL&P QA management was intended to function in providing programmatic direction to lower QA/QC echelons and the contractors.

47. The function of the NRC's Office of Inspection and Enforcement (I&E) is to seek reasonable assurance that the licensee's programs meet NRC regulatory requirements. To that end, it performs selective inspections, which are not aimed at verification of individual components, actions or procedures, but rather to evaluate functioning of the above management-control system. *Id.* at 7-8. In NRC Region IV, the Reactor Construction and Engineering Support Branch (RCESB) is responsible for carrying out those activities.

48. The RCESB has conducted inspections and investigations of the HL&P STP program since about 1974 and throughout the period of time covered by the Phase I record.[47] Some of the "inspections" are routine in nature and initiated by NRC personnel to review construction and other activities at the site for comparison with NRC criteria, as part of a program intended to identify problems and prevent them from attaining serious safety significance. Others have been conducted in response to problems reported to NRC by the licensee in accordance with NRC regulations. "Investigations," on the other hand, are initiated in response to allegations of program irregularities received by the Staff by telephone, letters or other means. Because most of these investigations involve allegations about safety matters, the threshold for initiating them is very low. "Inquiries" sometimes are conducted to obtain specific information desired by other NRC offices, NRC management or Congressional officials. Results of the inspections and investigations are written up subsequently as "I&E Reports." *Id.* at 8-9; Tr. 9352 (Seidle); Tr. 10,358 (Herr); Tr. 10,363 (Phillips).

49. Instances where the licensee fails to meet regulatory requirements are recorded as "noncompliances," which were categorized during the period in question at three levels of severity: violations, infractions and deficiencies. A violation was the most severe and was issued when the fabrication, construction, testing or operation of a Safety-Related Category I system was such that its function or integrity was lost. An infraction was less serious in that the system was impaired, but not lost. A deficiency was a noncompliance in which the threat to health, safety or interest of the public was remote and included such

[47] In April 1982, the NRC announced a forthcoming reorganization of its investigative functions, centralizing the conduct of "investigations" in a new Office of Investigations (OI). *See* NRC Announcement No. 38, dated April 20, 1982.

items as failure to follow procedures, and posting or labeling requirements which were not serious enough to constitute infractions. A fourth category, which was not viewed as a noncompliance, was a "deviation" and covered instances in which the licensee failed to keep a promise concerning controls or procedures, but the commitment was not a regulatory requirement. Licensees were required to take appropriate corrective actions for noncompliances and to report them to the NRC. Subsequent NRC inspections were made to assure that the corrective actions had been implemented properly, after which the noncompliance was closed out. Seidle, *et al.*, ff. Tr. 9205, at 10-11.

50. NRC enforcement is based on assessing deviations from the program of the specific facility under consideration. In carrying out its enforcement activities, the NRC does not attempt to compare licensee performance with that at any other facility. Tr. 9469 (Crossman).

(a) Noncompliances Before Investigation 79-19

51. During the approximately 6 years of STP activities preceding I&E Investigation 79-19, the Staff conducted a total of sixty-seven NRC-initiated inspections and eleven investigations in response to allegations of defects in construction or procedures received from various sources. Seidle, *et al.*, ff. Tr. 9205, at 10 and Appendix A. These resulted in issuance of: no "violations," thirty "infractions" and three "deficiencies," for a total of thirty-three items of noncompliance with NRC requirements or Applicants' specifications. In addition, eight "deviations" from controls or procedures to which the Applicants had committed to the NRC but which were not required by NRC were cited. *Id.* at 10-11 and Appendix B.

52. The eleven investigations did not result in the substantiation of most of the allegations that caused them to be initiated. This result can be attributed in part to the fact that the threshold for initiating an investigation is low because most investigations involve allegations concerning safety matters. Tr. 10,363 (Phillips). Eight of the eleven investigations produced no noncompliances. The remaining three resulted in three infractions for failing to follow specified methods in cadwelding procedures, inspections and record-keeping, one infraction for failing to follow specified procedures for release of a Stop Work Notice, and one deviation for failure to record the identity of the person making a supplemental entry in a QA record. Staff Exh. 17 (I&E Rept. 79-01, Notice of Violation); Staff Exh. 32 (I&E Rept. 79-14, Notice of Violation); Seidle, *et al.*, ff. Tr. 9205, at 31-33, 38-39, 52-53 and Appendices A and B.

53. HL&P took corrective actions for each of the noncompliances. Subsequent examinations by the Staff approved the resolution of each of them. Seidle, *et al.*, ff. Tr. 9205, at 33-35, 39-41, 56-57; Staff Exhs. 15, 16, 18, 19, 33, 34 and 35.

54. The remaining twenty-six infractions, three deficiencies and seven deviations originated in the sixty-seven inspections initiated by the NRC Staff between November 1973 and November 1979. The inspections covered a wide range of engineering, construction, and QA/QC activities and the noncompliances related principally to real or potential construction defects, organizational and procedural aspects of QA/QC, and various other types of problems in program implementation. Seidle, *et al.*, ff. Tr. 9205, at Appendices A and B.

55. Twenty-six inspections of concrete activities and records produced eight of the infractions and one deviation, several of which were addressed specifically in Staff direct testimony. HL&P responded to the noncompliances with appropriate corrective actions, which subsequently were inspected and approved by the Staff. *Id.* at 41-63 and Appendix B. *See also* Findings 293-301 and 327-337, *infra.*

56. Staff inspections and investigations of cadwelding activities produced a total of five noncompliances and one deviation. Some of these also were addressed specifically in Staff direct testimony. The Staff described HL&P's corrective actions for these events and the Staff inspections that approved them and closed out the noncompliances. Seidle, *et al.*, ff. Tr. 9205, at 32-41 and Appendix B. *See also* Findings 338-345 and 360-367, *infra.*

57. All of the noncompliances and deviations prior to I&E Investigation 79-19 may be summarized in five broad categories, as follows:

(a) failure to provide adequate procedures, instructions, specifications, drawings and schedules (six infractions and four deviations);
(b) failure to follow appropriate procedures and specifications (eighteen infractions);
(c) use of improperly qualified personnel (two infractions and two deviations);
(d) failure to maintain adequate records (two infractions, three deficiencies and two deviations); and
(e) inadequate or incomplete inspections (two infractions).

58. A Staff panel testified about the results of several inspections and investigations carried out during this period and summarized its findings and perceptions about those events, their root causes and the responses of HL&P and B&R to deficiencies identified by NRC. Seidle, *et al.*, ff. Tr. 9205. HL&P management was observed to be responsive and

committed to excellence in QA/QC activities. HL&P followed up actively and promptly with corrective actions on identified problems and sought out other problems where they existed. At no time did the company and its contractors make any effort to orchestrate any anti-QA/QC activities. Tr. 9506-07 (Seidle); Tr. 9850-67 (Phillips, Shewmaker, Hayes). Independence of QA from project management was viewed as satisfactory at STP (Tr. 9512-14 (Taylor, Seidle)). Mr. Phillips, the Resident Reactor Inspector for STP, testified that HL&P management was cooperative — probably the most open licensee that he ever dealt with — and seemed to be dedicated to having a model QA program, as reflected, for example, by the presence of the highest number of QA personnel on site that he had observed. His observation was that the allegations in the STP quality history were isolated from each other, but that a flood of them occurred just prior to the NRC decision to initiate Investigation 79-19. Tr. 9516-17 (Phillips).

59. The Staff panel concluded that HL&P was cooperative and diligent in correcting specific problems when cited, but that the same problems eventually resurfaced, evidencing HL&P's inability to control construction activity. Seidle, *et al.*, ff. Tr. 9205, at 64; Shewmaker, *et al.*, ff. Tr. 9576, at 4; Tr. 9506, 9527-28 (Seidle); Tr. 9857 (Phillips). Mr. Taylor did not detect any unwillingness on the part of HL&P to do the proper things, but felt that it lacked the knowledge of how to do them (Tr. 9511-12 (Taylor)). Panel members agreed that inexperience of the licensee was a major factor in occurrences of the problems (Tr. 9506-17). In addition, during the summer of 1978, B&R was understaffed with respect to QC personnel (Tr. 9277 (Seidle)).

60. Mr. Oprea, the HL&P executive with overall responsibility for the STP, testified that during these years, until well into 1979, he had felt that the QA program was working well and that the problems found at STP were isolated events on a large and complex project that was achieving generally satisfactory work. The HL&P audits had indicated that construction was proceeding generally in compliance with NRC and STP design requirements. Allegations concerning confrontations between construction and QC personnel also were viewed as isolated incidents, to which appropriate responses were taken by B&R management. Mr. Oprea had felt that the STP problems were typical of ones being experienced on other nuclear construction projects. The several citations for nonconformances in 1979, increased construction activity and the post-TMI situation caused him some concern about the project to the point of considering an independent audit of QA/QC activities. Oprea, *et al.*, ff. Tr. 1505, at 9-19; Tr. 2221-24, 5095-5100 (Oprea). Mr. Jordan testified that he hadn't thought that there was a QA

problem, but later recognized some deficiencies in the program, especially in its implementation (Tr. 1445-46 (Jordan)). Although not directly related to QA, Mr. Jordan also indicated that during this early period HL&P had problems concerning the adequacy of B&R's engineering performance. HL&P had been led to believe that around 50 percent of B&R's engineering would be completed at the time the construction permits were awarded. In fact, only approximately 8-9 percent of the engineering was complete at that time. Tr. 1228-29 (Jordan).

61. None of the noncompliances and deviations were based on untruthfulness, attempts to mislead NRC personnel, withholding information, lack of cooperation, reluctance to initiate and carry out corrective actions, refusal to acknowledge responsibility, or other such misconduct by HL&P management or personnel. Seidle, *et al.*, ff. Tr. 9205, at 64 and generally, Tr. 9208-9561; Tr. 9850-67.

62. Although HL&P responded properly in correcting specific problems as they arose, it was unable to prevent their recurrence, suggesting inadequate control over construction and the contractor, B&R. Specific items of concern to the Staff in this respect were the recurrent problems in cadwelding, failure to follow specified concrete pour procedures, and various QA/QC management problems. Of special importance, the NRC continued to receive allegations concerning lack of management support for QA/QC inspectors, poor inspector morale, harassment of civil QC inspectors by construction personnel, inadequate QA/QC staffing, and other QA/QC-related complaints, strongly suggesting lack of project control by HL&P. Seidle, *et al.*, ff. Tr. 9205, at 64; Staff Exh. 9 (I&E Rept. 78-13); Tr. 9505-07, 9539-40 (Seidle). The Staff determined that, during this period prior to the 79-19 Investigation, there was an "inordinate amount of friction" between B&R QC inspectors and B&R construction personnel. Indeed, there was more friction than the Staff inspector who authored that language had ever experienced during his more than 30 years' experience in QA/QC matters. Tr. 9369-70 (Taylor); Seidle, *et al.*, ff. Tr. 9205, at 117; *see also* Tr. 9468-69 (Hubacek).

63. Because of those problems, the NRC Staff reviewed past inspection and enforcement reports and concluded that the mid-term QA inspection, scheduled to occur in 1980, should be performed a year earlier. The report for that inspection revealed several QA deficiencies. Staff Exh. 27 (I&E Rept. 79-13); Shewmaker, *et al.*, ff. Tr. 9576, at 4.

64. An additional series of allegations concerning harassment of QC personnel was conveyed to the Resident Reactor Inspector during November 1979, causing the Director of I&E to order an in-depth investigation (79-19) into QA/QC management support, harassment of

personnel, and effectiveness of the STP QA program (Shewmaker, *et al.*, ff. Tr. 9576, at 4-6; Seidle, *et al.*, ff. Tr. 9205, at 64). The nature of these allegations differed from earlier similar allegations in that a group of them was brought to the Staff's attention simultaneously rather than as individual isolated allegations (Tr. 9965 (Phillips)).

(b) Noncompliances Identified in Investigation 79-19

65. Problems identified during the special investigation that resulted in I&E Report 79-19 and resulting enforcement actions taken by NRC were addressed by the Staff's Shewmaker panel, identified in Finding 19. Shewmaker, *et al.*, ff. Tr. 9576.

66. The 79-19 Investigation was conducted because of allegations received on November 2, 1979, by the NRC Resident Reactor Inspector, Mr. Phillips, from workers at the South Texas Project concerning lack of management support, threats, and harassment of civil QC inspectors, similar to other allegations that had been received and investigated earlier. It was undertaken to determine the validity of the recurring allegations and to assess the effectiveness of the QA/QC program at the STP. *Id.* at 6; *see also* Finding 64, *supra.*

67. The investigation was carried out over about 3 months, between November 10, 1979, and February 7, 1980, and consumed 1113 manhours by one investigator and five inspectors, representing four NRC regional offices (I, II, III, and IV). It included observations, document reviews, witnessing of tests and over 100 formal and informal interviews with HL&P and B&R construction and management personnel, engineers, inspectors, Pittsburgh Testing Laboratory employees and other site personnel. Staff Exh. 46, Appendix D, at 1-2, 5-9; Shewmaker, *et al.*, ff. Tr. 9576, at 6-7; Tr. 9590-92 (Hayes, Phillips).

68. Some results of the special investigation were communicated to HL&P in a meeting on December 21, 1979 and in an exit interview on January 24, 1980. The full findings were transmitted to the company on April 30, 1980, as I&E Report 79-19, accompanied by: (a) a Notice of Violation, (b) a Notice of Proposed Imposition of Civil Penalties, and (c) an Order to Show Cause. Staff Exh. 46; *see* Finding 7, *supra.*

69. In the December meeting, Mr. Oprea and Mr. Turner were informed by the Staff that there were serious problems in the QA/QC program with respect to concrete placement and harassment of QC inspectors. In the January meeting, those findings were reiterated and HL&P was notified, in addition, that noncompliances had been identified in welding, nondestructive examination (NDE) and the backfill placement program. The Staff also reported that HL&P and B&R QA organi-

zations were not effectively implementing the auditing program or analyzing trends in nonconformances and that there were inadequacies in document control. Oprea, *et al.*, ff. Tr. 1505, at 20-21.

70. I&E Report 79-19 produced a total of twenty-two noncompliances, which are summarized as follows:

(a) existence of organizational defects in the STP QA/QC program;
(b) failures to provide adequate procedures, instructions, and specifications (three noncompliances);
(c) failures to follow proper procedures (three);
(d) failures to take proper corrective actions when defects were observed (six);
(e) failures to maintain adequate records (three);
(f) inadequate or incomplete inspections or audits (four);
(g) use of improperly qualified personnel (two).

Staff Exh. 46 (I&E Rept. 79-19; Appendix A, Notice of Violation); Shewmaker, *et al.*, ff. Tr. 9576, at 7-34.

71. As a result of the findings of I&E Report 79-19, the company was served with a Notice of Violation citing the noncompliances and requiring that HL&P respond within 25 days, admitting or denying each item of noncompliance, giving reasons for each, corrective steps taken, and the date when full compliance would be achieved. The NRC also proposed a civil penalty of $100,000. Staff Exh. 46, Appendix A, at 19, and Appendix B (Notice of Proposed Imposition of Civil Penalties).[48]

72. HL&P responded to the Notice of Violation on May 23, 1980, stating that each incident probably occurred, admitting to each item of noncompliance, and paying the $100,000 civil penalty. The company identified six "root causes" to which all of the incidents could be traced and indicated that its objective for the next several months would be to attack those causes. Shewmaker, *et al.*, ff. Tr. 9576, at 38; Staff Exh. 47 (HL&P Response to Notice of Violation).

73. Shortcomings identified in the QA/QC program included the following areas:

1. control of concrete placement activities;
2. welding and nondestructive examination activities;
3. control of backfill material placement and testing;
4. identification of recurring problems;
5. effectiveness of corrective actions;

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[48] The Notice of Violation recited that the infractions set forth therein warranted total civil penalties of $372,000 but, as a result of statutory limits, civil penalties of $100,000 were being proposed (Staff Exh. 46, Appendix A, at 19 and Appendix B).

6. independence and authority of civil QC inspectors;
7. auditing program.

Shewmaker, *et al.*, ff. Tr. 9576, at 40-41.

74. Item of Noncompliance No. 1 indicated that the quality control function in the B&R Civil area was not sufficiently independent from construction and had inadequate authority and freedom to identify problems and resolve them adequately. Some of the inspectors were subjected to production pressures, not always supported by B&R QC management, harassed, intimidated and threatened by construction personnel. In this situation, some inspectors felt that it would be easier to approve inadequate procedures and construction than to be confronted by construction or quality-control management. That notwithstanding, results of the investigation did not disclose any instances of any significance in which inspectors failed to do their jobs. The Shewmaker panel emphasized that no irreparable construction deficiencies were found in completed structures. The Staff witnesses concluded that there had been incidents of harassment of inspectors (tension, verbal abuse or friction); but, except possibly in one instance (Tr. 9932 (Hayes)), no intimidation (interference with job performance). Staff Exh. 46, Appendix A, at 1-5; Shewmaker, *et al.*, ff. Tr. 9576, at 7-8, 11-13, 42; Tr. 9237-43 (Taylor, Seidle); Tr. 9632 (Hayes); Tr. 9651 (Phillips); Tr. 9859 (Phillips); Tr. 9930-35 (Phillips, Hayes); Tr. 9958 (Shewmaker, Phillips, Hayes).

75. Allegations of harassment of B&R QC inspectors by construction personnel and friction existing among them have especially important implications for matters to be decided here about corporate competence and character. *See* Findings 62, 64, 66, *supra*. Several such allegations were substantiated in the special investigation, as well as the fact that efforts by HL&P had failed to find the root causes and correct those problems, in spite of having had knowledge about the complaints and allegations for 1 to 2 years. Shewmaker, *et al.*, ff. Tr. 9576, at 7-8, 35-36, 40; Tr. 1378 (Jordan); Tr. 3560-66 (Turner); Oprea, *et al.*, ff. Tr. 1505, at 11-15. These matters are discussed more fully in Findings 381-398, *infra*.

76. Several items of noncompliance dealt with poor practices in concrete placement, inspection and documentation. The QA/QC program was ineffective in preventing recurrence of those problems, which sometimes resulted in voids in structural concrete. The investigation team attributed much of the cause to unclear procedures and qualitative acceptance criteria, personnel with inadequate training, experience and education, and pressures on inspectors through harassment.

Shewmaker, *et al.*, ff. Tr. 9576, at 8, 41, 44; Staff Exh. 46, Appendix A. Some of these matters are further addressed in Findings 327-336, *infra.*

77. Other areas of construction to which noncompliances referred included backfill placement that may not have been sufficiently compacted to meet required densities, improper welding controls and welder qualifications, and inaccurate NDE performance and interpretation. Shewmaker, *et al.*, ff. Tr. 9576, at 8, 44, 45; Staff Exh. 46, Appendix A; Tr. 9921-30 (Hayes, Shewmaker, Phillips). Some of these matters are further discussed in Findings 276-291 and 302-315, *infra.*

78. The HL&P and B&R audit and surveillance programs were not properly implemented and effective trend analyses were not performed, allowing many of the problem areas to become chronic and continue to recur. Shewmaker, *et al.*, ff. Tr. 9576, at 8, 44; Staff Exh. 46, Appendix A; Tr. 2279 (Oprea). *See* Findings 115-116 for more details.

79. An especially important finding of the investigation team was that serious procedural and programmatic inadequacies in the HL&P and B&R QA/QC organizations had resulted in failures to identify quality control problems and to take appropriate steps to correct them and prevent their recurrence. The Staff viewed HL&P management as having been over-reliant on B&R to implement the QA/QC program and negligent in not keeping itself better informed about site activities and problems. Shewmaker, *et al.*, ff. Tr. 9576, at 35-37, 42; Tr. 9859 (Phillips); Tr. 9938 (Hayes); *see also* Finding 116, *infra.*

80. QA/QC organizational problems were not attributed to inadequacies in the written QA/QC program, but to lack of detailed involvement by HL&P in the total scope of construction activities at STP, which hindered HL&P in implementing the requirements and procedures and in maintaining adequate control over its general contractor, B&R. Shewmaker, *et al.*, ff. Tr. 9576, at 7-8, 35-37, 42; Tr. 9602, 9864 (Hayes). Mr. Jordan, in response to a question, indicated his agreement with the NRC findings and stated that in his view the QA organization had been satisfactory, but implementation of the program had been poor (Tr. 1364-65, 1446 (Jordan)). Mr. Amaral agreed. He also testified that there had been too little management involvement and that, in fact, this was the underlying cause among all of the root causes. Tr. 1714-15, 1740-44, 1748, 1917-18, 2061, 2242 (Amaral). Mr. Oprea stated that he did not disagree with Mr. Amaral's diagnosis of the problems and that he now visits the site more often, reads more reports, and talks with many personnel at the site (Tr. 2238, 2241, 2243-45, 2264 (Oprea)). Mr. Frazar distinguished carefully between the written program, which complied with NRC regulations, and its implementation, which some-

times did not (Tr. 1794-1814). Mr. Amaral and Mr. Turner agreed (Tr. 1917-18, 1950, 2065, 3421).

81. Because the findings in I&E Report 79-19 showed widespread noncompliance by both HL&P and B&R, and in view of the past record of inspection and enforcement at STP, the Staff issued an Order to Show Cause why safety-related construction activities on the South Texas Project should not be stopped 90 days from date of the Order and remain stopped until the Licensee completed ten specific items identified in the Order that would permit the Staff to evaluate whether further activities at the STP could be conducted in accordance with Appendix B, 10 C.F.R. Part 50. HL&P responded to this Order on July 28, 1980, providing the basis for permitting continued construction of the STP. Shewmaker, *et al.,* ff. Tr. 9576, at 9-10, 39-48; Staff Exh. 46 (Order to Show Cause at 12-17); Staff Exh. 48 (Licensee's Response to Order to Show Cause).

82. The ten specific items that HL&P was directed to carry out included the following:

1. Contract with an experienced, independent consulting firm, knowledgeable in QA/QC and nuclear construction, to evaluate the STP QA/QC program management.
2. Review safety-related aspects of Category I structural backfill.
3. Review safety-related welding and concrete structures and report on necessary repairs and scheduling.
4. Rescind a B&R brochure on the STP QA program and issue a new one consistent with 10 C.F.R. Part 50, Appendix B.
5. Define personnel authority for stop work more clearly and describe implementation methods.
6. Develop and implement more effective ways for identifying and correcting "root causes" of problems.
7. Develop and implement a more effective program for control of field changes.
8. Develop and implement a more effective system for records control.
9. Develop and implement an improved audit system.
10. Verify or correct, if necessary, the statements in Section 2.5.4 of the FSAR.

Staff Exh. 46 (Order to Show Cause at 12-17); Shewmaker, *et al.,* ff. Tr. 9576, at 9-10.

83. HL&P responses to the findings of Investigation 79-19 began during the period in which the investigation was still under way, before publication of the I&E Report and Show-Cause Order on April 30, 1980.

Shortly after the meeting between the NRC Staff and HL&P management on December 21, 1979, the company voluntarily stopped all complex concrete pours. In January 1980, after several weeks of study, HL&P retained an outside consultant (Bechtel Corporation) to conduct an in-depth audit of the QA/QC program. HL&P also retained outside consultants to review and advise management on other problems that had been identified, including backfill, harassment and welding. More personnel in Houston management were assigned to the STP site and outside consultants were added to the HL&P staff to enhance its capabilities. As a result of all of those activities, the company discovered additional program deficiencies and proceeded to report them to NRC and address them, as well. For example, HL&P voluntarily stopped safety-related welding in April 1980. Tr. 9857-61 (Phillips, Shewmaker); Oprea, *et al.*, ff. Tr. 1505, at 20-23; Tr. 1440-42 (Jordan); Finding 93, *infra*.

84. Not all of the actions taken to improve QA/QC prior to the issuance of the Show-Cause Order were successful. In January 1980, a lecture was given by the project QA manager to construction and QA/QC personnel, and a brochure entitled "Implementation of the Brown & Root Quality Assurance Programs at the South Texas Project Job Site" was issued. In the opinion of the Staff, the lecture and brochure overemphasized the importance of construction pressures at the expense of quality assurance. Shewmaker, *et al.*, ff. Tr. 9576, at 42-43. The Show-Cause Order directed that the brochure be rescinded and replaced (Finding 82, Item 4, *supra*). HL&P agreed to do so and, on July 30, 1980, a new brochure was distributed and discussed at a seminar (Staff Exh. 48, at 4-1 and Exhibit 19; Staff Exh. 64 (I&E Rept. 80-18, at 3); Crossman, *et al.*, ff. Tr. 10,010, at 43-44).

85. The HL&P written response to the Show-Cause Order was submitted in timely fashion on July 28, 1980 and described corrective actions taken and to be taken. These will be discussed in detail subsequently. *See, e.g.*, Findings 122-124, 200-219, 275 *et seq.*; Staff Exh. 48 (Licensee's Response to Order to Show Cause).

86. The Staff indicated that it did not find a total breakdown in the QA/QC program (Tr. 9851 (Phillips)). To the contrary, in many ways the program exceeded NRC requirements and worked well, resulting in most activities being carried out correctly (Tr. 9601 (Hayes); Tr. 9851, 9855 (Phillips)). The HL&P record in reporting construction deficiencies under 10 C.F.R. § 50.55(e) was better than that of most utilities and the company was viewed by the panel as open and honest (Tr. 9855 (Phillips)). No irreparable construction deficiencies were found and, in fact, no noncompliance was issued at the level of a violation, indicating

that the Staff did not feel that the functional integrity of any system had been lost. The Staff did not view any of the noncompliances reported during Investigation 79-19 to be severe enough to indicate that HL&P management was irresponsible or grossly negligent. Shewmaker, *et al.*, ff. Tr. 9576, at 8, 49; Tr. 9829 (Hayes); Tr. 9853-54 (Phillips).

87. The Staff concluded that the record of HL&P compliance with NRC requirements, if viewed without regard to the remedial steps taken by HL&P following 79-19, was not sufficiently poor to conclude that HL&P does not have the managerial competence or character to be granted operating licenses for the South Texas Project. The lack of involvement in the construction activities was the result of inexperience in nuclear construction, rather than irresponsible corporate management. The Staff also concluded that, given the absence of irreparable deficiencies in the construction already completed, if corrective action proposed by HL&P were implemented, the STP would be in compliance with NRC requirements and operating licenses should be granted. Further, it observed that HL&P consistently had shown a willingness to implement corrective action. The Staff's examination of the STP revealed shortcomings in project management during construction that it viewed as relevant to, and probative of, how HL&P would perform under an operating license, but HL&P's prior behavior was not considered by Staff reviewers to be determinative. Shewmaker, *et al.*, ff. Tr. 9576, at 48-50; Tr. 9854, 9861-64, 9935-40 (Phillips, Shewmaker, Hayes); *see* Finding 248, *infra.*

(c) Noncompliances After Investigation 79-19

88. Staff testimony on NRC inspection and enforcement activities after Investigation 79-19 and the Order to Show Cause, as reflected in I&E reports issued since that time, was presented by the Crossman panel (identified earlier in Finding 19). The purposes of this testimony were: (1) to outline noncompliances identified in inspections and investigations conducted since issuance of I&E Report 79-19, (2) to describe the status of HL&P corrective activities for the noncompliances identified in I&E Report 79-19, and (3) to summarize HL&P responses to directives set forth in the Order to Show Cause. Only noncompliances identified since the 79-19 I&E report will be discussed here — the remedial actions will be discussed subsequently in connection with Issue B. Crossman, *et al.*, ff. Tr. 10,010; Staff Exhs. 31, 35, 40, 45, 49-89, 92-100, 113-132. Cross-examination of the panel by parties and examination by the Board is at Tr. 10,011-10,118.

89. After Investigation 79-19 and up through December 31, 1981, the NRC performed seventy-four inspections or investigations at the STP, during which fourteen items of noncompliance were identified, distributed as follows:

(a) failure to develop appropriate procedures (one noncompliance),
(b) failures to follow procedures (six),
(c) failure to assure quality of purchased material (one),
(d) inadequate or incomplete inspection (one),
(e) failures to report construction deficiencies in a timely manner (four), and
(f) failure to respond to B&R audit findings (one).

These noncompliances were described by the panel as being similar to those for which HL&P had been cited in the past. Crossman, *et al.*, ff. Tr. 10,010, at 5 and Appendix A.

90. Two instances of document falsification were substantiated by NRC after the Order to Show Cause. In one instance, a B&R employee admitted that he had initialed and dated a document indicating that a "hold point" inspection had been performed when, in fact, it had not. Shortly after I&E Report 80-14 (Staff Exh. 60) describing this incident was issued, the individual was terminated. Crossman, *et al.*, ff. Tr. 10,010, at 13-15. *See also* Finding 424, *infra.*

91. The other instance involved a B&R foreman who was alleged to have falsified two plant maintenance records concerning inspection of vacuum degasifier pumps and who admitted falsifying one such record and also ordering a subordinate to sign off on maintenance cards for equipment that was inaccessible in a locked building. The findings and conclusions of the Staff investigation of these allegations are reported in I&E Report 80-21 (Staff Exh. 67). The supervisor of the person making these allegations subsequently quit and two other supervisors reported to foster this behavior were transferred off site several months after the investigation. Crossman, *et al.*, ff. Tr. 10,010, at 15-17. *See also* Finding 424, *infra.*

92. Based on all of the inspections and investigations, and taking into account the various noncompliances and corrective actions taken or planned, this panel concluded that there is reasonable assurance that structures in place at the STP as of early 1982 are in conformity with the construction permits and provisions of NRC regulations and that there are no major safety-related problems with the completed structures or physical systems (Crossman, *et al.*, ff. Tr. 10,010, at 52). *See also* findings on Issue E, *infra.*

(d) Evaluation of Root Causes of Noncompliances

93. As pointed out earlier, Mr. Oprea testified that circumstances in late 1979 led him to consider an independent audit of the QA/QC program. *See* Finding 60, *supra.* After several weeks of study and preliminary meetings with NRC personnel concerning the findings of Investigation 79-19, HL&P management recognized the greater breadth and seriousness of the problems and decided to retain the Bechtel Corporation to carry out that study. The scope of the study was defined during two meetings between HL&P and Bechtel in January and February 1980. Later, after the report of Investigation 79-19 had been issued, the scope of the Bechtel study was broadened to include evaluation of alternative types of various QA/QC management organizations, in response to Item 1 of the Show-Cause Order. *See* Finding 83, *supra*; Oprea, *et al.*, ff. Tr. 1505, at 18-19, 22-23, 31, 119; Tr. 1362-64 (Jordan); Tr. 2087-98, 2254-55, 5465-68 (Oprea); Staff Exh. 48 (Licensee's Response to Order to Show Cause), Exhibit 1.

94. The Bechtel report identified six "root causes" of deficiencies in the QA program. HL&P advised the NRC of its intent to concentrate on improvements in the following areas:

1. better clarity in writing specifications and procedures;
2. improved documenting and trending of nonconformances;
3. improving the training of personnel in QA goals, emphasizing STP reliability and safety;
4. improving systems controls to assure that QA activities are initiated, performed, reviewed and documented properly;
5. improving adherence to procedures through audits;
6. increased visibility and participation in QA activities by upper management.

The Bechtel and HL&P studies indicated that problems in the QA program could be traced to one or more deficiencies in those areas. Oprea, *et al.*, ff. Tr. 1505, at 26-27, 119-20; Staff Exh. 48 (Licensee's Response to Order to Show Cause), Exhibit 1.

95. Mr. Amaral, previously identified as Manager of Quality Assurance for Bechtel Power Corporation (Finding 38, *supra*), was in charge of the Bechtel study. He identified the root cause underlying all others as lack of visibility and participation by management. Tr. 2061, 2287, 2294 (Amaral). He indicated that much of management's inadequacy in its knowledgeability about and participation in STP activities could be attributed to communication problems between field personnel and upper management. Tr. 1714-16, 1850-51, 1897-98 (Amaral).

96. Much of the communication problem was caused by long organizational lines, with several administrative layers between field supervisors and executives. This, coupled with inexperience, produced poor communications in the detection and resolution of problems. Further, the remoteness of management weakened authority of onsite inspectors in their dealings with construction personnel. Mr. Amaral suggested that this situation could have contributed to an atmosphere in which QC inspectors could be harassed. In addition, the communication problem was intensified by the fact that audit reports were not issued beyond the level of the organization being audited, virtually eliminating feedback from upper management to correct problems discovered in the audits. Mr. Amaral concluded that HL&P has since taken steps to resolve the communication problem by several actions, especially by its 1980 transfer of Mr. Frazar, the corporate QA manager, to the plant site and having him report directly to Mr. Oprea. Tr. 1714-16, 1739, 1743, 1850-51, 1897-1901, 1934-35 (Amaral).

97. The excessively long chain of command can be illustrated by the fact that, before Investigation 79-19, Mr. Turner, then Vice President, Power Plant Construction and Technical Services, was responsible for both QA and project management and reported to Mr. Oprea. Oprea, *et al.*, ff. Tr. 1505, at 7. Mr. Frazar, the Corporate QA Manager, reported to Mr. Turner. Within the QA Department, the Projects QA Manager reported to Mr. Frazar and supervised the STP Project QA Supervisor, who was located in Houston. The site QA supervisor, Mr. Logan Wilson, was stationed at STP and directed the HL&P QA staff, under supervision of the STP Project QA Supervisor. Information on problems originating at the field inspector level, then, had to move through six management levels at various locations before reaching Mr. Oprea. Indeed, there were four layers of supervision between the site QA organization and Mr. Oprea. Oprea, *et al.*, ff. Tr. 1505, at 7, 42; Tr. 1714-15 (Amaral, Frazar).

98. The intent was that HL&P should serve in an oversight capacity to provide programmatic direction to B&R on implementing the STP QA program. Oprea, *et al.*, ff. Tr. 1505, at 8. Mr. Amaral defined programmatic direction as establishment of the policies and basic procedures by which the program should be implemented. He testified that at the time of the Bechtel audit HL&P did not have adequate experienced staff to provide that direction. Tr. 2228 (Amaral). Mr. Amaral felt that the problems of too little management involvement and poor communications continued because HL&P personnel were inexperienced in constructing and operating nuclear power plants. Tr. 1905-06, 2228-29 (Amaral).

99. Mr. Jordan, HL&P President and Chief Executive Officer, had no prior involvement with nuclear construction or operation before plans were initiated for STP. He knew that construction and operation of a nuclear power plant would be more complex than for a fossil fuel plant, but did not realize just how complex it could be until Investigation 79-19. Tr. 1396-98 (Jordan).

100. Mr. Oprea has been in charge of the project since its inception and is the second-ranking officer of the company. He had an extensive background of engineering experience, but no prior experience with nuclear power plant design or construction before the decision to build STP. Oprea, *et al.*, ff. Tr. 1505, at 3-4.

101. Mr. Turner also had many years of engineering experience, much of it in power plant design and construction. However, he too had no nuclear power plant experience prior to the STP. He expressed the view that lack of corporate experience was an important factor in causing the QA/QC program to be out of compliance with NRC requirements. Tr. 3421 (Turner).

102. Mr. Frazar lacked prior experience in either QA/QC or nuclear construction and readily admitted that his inexperience could have contributed substantially to some of the STP problems (Tr. 3244-46 (Frazar)). Mr. Amaral testified that Mr. Frazar was articulate and bright but lacked the experience required for his position (Tr. 1766-67 (Amaral)). Mr. Jordan expressed confidence in Mr. Frazar's abilities. He conceded, however, that Mr. Frazar needed additional experienced help and that, with the same choice to make today, he (Jordan) would employ somebody with more experience than Mr. Frazar had at the time he was placed in his position. Tr. 1443-45, 1466-68 (Jordan).

103. Mr. Frazar indicated that Mr. Wilson (the site QA supervisor) did not have experience adequate for his position (Tr. 3244 (Frazar)). Mr. Amaral expressed a similar view about Mr. Wilson (Tr. 1935 (Amaral)). Mr. Amaral also testified that some of the managers in the B&R organization also were inexperienced. Specifically, Mr. Thomas G. Warnick, the B&R site QA Manager, seemed overwhelmed by his job and Mr. Amaral thought that the job was beyond him. Tr. 1938-39, 2066 (Amaral). Mr. Amaral had recommended that HL&P and B&R both retain qualified site managers (Tr. 1599 (Amaral)). He also stated that of the approximately twenty to twenty-five supervisory QA/QC positions in HL&P and B&R, about fifteen required changes (Tr. 2069-70 (Amaral)).

104. Mr. Jordan felt that failure to perceive problems stemmed from management's failure to receive the types of information needed for informed decisions and that HL&P's failure to perform adequate audits

and trend analyses contributed to that problem (Tr. 1394-95 (Jordan)). He indicated further that HL&P responded to each individual item of noncompliance adequately, but that the lack of proper trending to examine the situation in depth prevented it from seeing emerging trends in a timely fashion (Tr. 1446-47 (Jordan)).

105. Dr. Broom, who has been identified earlier as Senior Vice President of the B&R Power Group (Finding 39, *supra*), testified that personnel occupying the position of STP General Manager for B&R changed six times between 1977 and 1981, an average of once each 8 months, and the STP site manager was filled by seven persons during that period, giving an average change frequency of once in 7 months (Tr. 4362-63, 4366 (Broom)). Reasons cited by B&R for this high rate of turnover included slipping schedules and rising costs, which forced removal of some managers. Others were changed because they had been placed in the position only for an interim period. Some left in response to more lucrative offers from other companies. Tr. 4366-75 (Broom, Grote).

106. Dr. Broom indicated that a new STP General Manager might require 3-6 months to become fully effective and that it would be preferable for that position to be occupied by the same person for several years (Tr. 4364-66 (Broom)). In spite of potential effects on continuity and employee morale and effectiveness, neither Dr. Broom nor Mr. Grote felt that the unusually large turnover contributed significantly to the STP difficulties (Tr. 4378-84 (Broom, Grote)). Staff witnesses were uncertain about the specific effects of those changes on the STP problems, but one indicated that the poor quality of some of the B&R managers did, in his view, contribute in major fashion to them (Tr. 9522-26 (Seidle, Taylor, Crossman)).

107. Mr. Amaral testified that HL&P management originally did not have a "quality first" philosophy and was not knowledgeable enough about QA/QC. He emphasized the importance of having a commitment to quality throughout the entire organization and indicated that this had not been adequate throughout HL&P and B&R. Tr. 1591, 1752-54, 1850-51 (Amaral). Mr. Oprea stated that the spirit of quality always was there in HL&P, but that there was an implementation problem (Tr. 2295 (Oprea)).

108. Mr. Jordan testified that there long has been a corporate commitment to safety and that his view of QA is that quality should be built into the facility (Jordan, ff. Tr. 1224, at 3-8; Tr. 1266-77 (Jordan)). Soon after becoming President of HL&P, in 1974, he issued a policy statement to that effect (Tr. 1278-83 (Jordan)). His sensitivity to the complexity of nuclear power plants has increased and now he spends more time on STP matters (Tr. 1396-98, 1452 (Jordan)). He had lacked

information needed to recognize some problems, but now spends more time reviewing various aspects of the STP (Tr. 1373, 1394-95 (Jordan)). He did not feel that it would be accurate to say that HL&P had abdicated its responsibility to its contractors or failed to keep itself knowledgeable concerning activities at the STP (Jordan, ff. Tr. 1224, at 8).

109. Mr. Oprea agreed with the diagnosis of the problems as advanced by Mr. Amaral (Tr. 2233-38 (Oprea)). He had felt his knowledge of STP activities was adequate before the Show-Cause Order, but he was dumbfounded when HL&P received the Show-Cause Order (Tr. 2090, 2239-40 (Oprea)). Mr. Amaral testified that Mr. Oprea had not been getting the root causes, but only information on isolated problems that were occurring from time to time (Tr. 2242 (Amaral)). Mr. Oprea agreed with that assessment and indicated that HL&P had been involved in curing problems but not in ascertaining their causes (Tr. 2243, 2235 (Oprea)). Mr. Frazar also stated that HL&P had been treating the symptoms and not the causes of the problems (Tr. 5421-22 (Frazar)). He concluded that there had been inadequate attention to supervision and support of QA personnel by B&R (Tr. 5405 (Frazar)).

110. Mr. Oprea stated that he had always been sensitive to the STP, but that his intensity had increased since Investigation 79-19 (Tr. 2243-44 (Oprea)). Before then, there had not been enough physical visibility of management at the STP site. He had visited the site at intervals of about 4-6 weeks, but subsequently has increased the frequency to about once per week. During those visits, he talks with various workers and others at the site to obtain information and increase management visibility. At the time of his testimony, he indicated that he was spending all of his time on nuclear issues and 90 percent of it on the STP (Tr. 2241-45, 2264, 3395, 3422 (Oprea)).

111. The Staff did not disagree with any of the root causes described by the Applicants' witnesses. The view of the Seidle panel was that the principal root cause was inexperience of HL&P and B&R management with construction and operation of nuclear power plants. Other important factors included the attenuated chain of command and the high rate of turnover in B&R site management personnel. Tr. 9508-14 (Seidle, Crossman, Taylor); Tr. 9517 (Phillips); Tr. 9522-26 (Seidle, Taylor, Crossman); Tr. 9532-34 (Seidle, Taylor). Part of the problem was specifically attributed by the Staff to deficiencies on the part of B&R personnel in management-level positions (Tr. 9522-23 (Taylor)). Mr. Seidle indicated that the inadequate communication and poor feedback from management to field personnel on actions initiated by them led to a perception that there was too little support for QC personnel (Tr. 9519-21 (Seidle)).

112. The Staff's Shewmaker panel also concluded that lack of detailed involvement by HL&P in the construction activities was a major reason behind the problems. The principal failure was not inadequacy of the written QA/QC program but in its implementation. Shewmaker, *et al.*, ff. Tr. 9576, at 7-9, 49. Inexperience in nuclear construction made HL&P rely too much on its contractor in carrying out QA commitments and caused HL&P management to become too involved in attention to details, to the neglect of evaluation of the total operation. Other important factors included production pressures, separation of management from the site operations and turnover in key personnel. Tr. 9864, 9936-40 (Hayes, Shewmaker, Phillips).

*(3) Extent to Which HL&P Abdicated Responsibility*

113. In CLI-80-32, *supra,* the Commission posed the question whether HL&P had abdicated too much responsibility for construction to B&R. According to the Commission, abdication of responsibility could form an independent and sufficient basis for revoking or denying a license on grounds of lack of competence or character of the licensee or applicant. 12 NRC at 291. HL&P's historical record on this question is the subject of Issue A(3) (*see* Finding 13). For convenience, we have evaluated under this issue the extent to which HL&P abdicated responsibility for construction of the STP to B&R, without regard to remedial steps. We are considering the effectiveness of remedial actions taken by HL&P separately in addressing Issue B.

114. Several Staff witnesses addressed Issue A(3). Mr. Seidle testified that the NRC holds HL&P responsible for development and implementation of a viable QA/QC program. HL&P is authorized to delegate the authority for conducting the program to its contractors and subcontractors, but cannot delegate the responsibility. He observed that: "perhaps they abdicated some of the responsibility, not so much at the highest levels of management, but perhaps at the field level" (Tr. 9506 (Seidle)). His contacts with corporate managers convinced him that they were responsive and totally committed to quality assurance and quality control, but that their management controls down to the worker level were not working effectively. He did not see any effort by the licensee or its contractors to orchestrate anything that would be anti-QA/QC. Mr. Taylor, Mr. Crossman and Mr. Phillips attributed much of the problem to lack of experience in constructing nuclear facilities. Tr. 9505-12, 9516-17 (Seidle, Taylor, Crossman, Phillips). There are differences in QA/QC requirements for construction of nuclear and non-

nuclear facilities (*e.g.*, Tr. 9509-10 (Crossman); Tr. 9864 (Hayes); Tr. 9890 (Shewmaker); Tr. 9939-40 (Phillips)).

115. The Staff Shewmaker panel (identified at Finding 19, *supra*) testified that HL&P relied too much on B&R in implementing the QA/QC program and inadequately followed up on surveillance and audit findings relative to that program. It concluded that HL&P did not take responsibility for the QA/QC program at the site and did not assure that the program there was proper. Shewmaker, *et al.*, ff. Tr. 9576, at 35-37. Mr. Hayes stated that part of the reason for program weaknesses was HL&P inexperience. He found the licensee to be responsible in its actions. Tr. 9829 (Hayes). Mr. Phillips elaborated on the basis for that finding in some detail and all other members of the panel agreed with his statements. *See, generally,* Tr. 9847-67.

116. The Staff Crossman panel (*see* Finding 19, *supra*) testified that during Investigation 79-19, HL&P admitted that it had failed to perform semi-annual audits of B&R site organizations and procedures and annual audits of B&R construction site activities, as required by the PSAR and HL&P procedures. It was this and similar findings that led the NRC to conclude that HL&P had abdicated too much responsibility for STP construction to B&R. Crossman, *et al.*, ff. Tr. 10,010, at 30-31.

117. After 79-19, HL&P revised its auditing procedures to require direct observation of the work being performed. I&E Report 80-27 concluded that HL&P had developed a matrix to assure that all procedures would receive proper consideration in planning audits. A 1981 I&E Report (81-07) documented that HL&P actually was performing effective audits at the prescribed frequency. Crossman, *et al.*, ff. Tr. 10,010, at 31; Staff Exh. 71 (I&E Rept. 80-27); Staff Exh. 92 (I&E Rept. 81-07).

118. HL&P President Jordan expressed the view that it would not be fair to state that HL&P abdicated its responsibility for STP to its contractors. He testified that HL&P was fully aware of the necessity for providing guidance and programmatic direction to its contractors. He indicated that HL&P recognized from the outset that nuclear construction required active participation by the owner and that HL&P had assigned highly qualified personnel in large numbers to manage the STP. Jordan, ff. Tr. 1224, at 8-9; Tr. 1389-93 (Jordan). With respect to this point, Staff witness Phillips testified that HL&P employed talented individuals and that the number of QA personnel on the site far exceeded the numbers that he was accustomed to seeing at other sites (Tr. 9516, 9939 (Phillips)). In fact, HL&P became so involved in project details that it was unable to take a broad view of the project and properly exercise its

management overview QA responsibilities. It could not see the forest for the trees. Tr. 9936 (Hayes); Tr. 9937 (Shewmaker).

119. Mr. Amaral, HL&P's expert witness on QA/QC organization, testified that in his judgment HL&P had not abdicated too much authority to Brown & Root (Tr. 1920-21 (Amaral)).

120. Mr. Oprea indicated that HL&P had not at any time abdicated its responsibility for QA at the STP, but had always recognized that responsibility for the QA program rested with HL&P. Over the years, HL&P management had become more involved in the project and more sensitive to the importance of its QA program. This involvement increased steadily over time from the beginning of the project, with HL&P forcing actions to be taken and becoming more involved in decisionmaking. He indicated that this increased involvement resulted in progressively closer supervision of the contractor by HL&P, which he explained was consistent with trends throughout the utility industry in general over the past several years. Oprea, *et al.*, ff. Tr. 1505, at 49-50; Tr. 5457-62 (Oprea). The increasing HL&P involvement from late 1980 on was confirmed by Mr. Goldberg (Tr. 10,488 (Goldberg)).

121. The record contains many examples of the exercise by HL&P of licensee responsibility in managing the STP. One such example is found in its reporting of deficiencies under 10 C.F.R. § 50.55(e). That regulation requires that the holder of a construction permit must notify the NRC of each deficiency found in design and construction which, if uncorrected, could adversely affect safety of operations. The history of HL&P's reporting under that regulation is summarized in Appendix C of testimony by the Crossman panel. Crossman, *et al.*, ff. Tr. 10,010, at 51-52 and Appendix C. Mr. Phillips, the Resident Reactor Inspector, reviewed the HL&P system for reporting those deficiencies and fifty-eight files of reports covering the period April 26, 1977 to July 3, 1980. He found that they contained objective evidence of timely evaluation of the items, and that "[t]heir record of identifying and reporting construction deficiencies, in accordance with 10 C.F.R. 50.55(e) was open and honest, and probably was better than any other utility that I've been at." Staff Exh. 92 (I&E Rept. 81-07, at 10); Tr. 9855, 10,068-69 (Phillips); Tr. 10,067 (Crossman); Staff Exh. 133 (I&E Rept. 81-37, at 6); *see also* Finding 158, *infra*.

122. Another specific example of HL&P exercising responsibility for the QA/QC program arises out of actions taken in response to the problem of low morale among QC inspectors. Mr. Oprea instructed Mr. Frazar to convey HL&P's concern to B&R, which was done through a strong presentation to a B&R QA executive board in January 1978. Mr. Frazar also instructed the HL&P QA staff to perform extra surveillance

during the following few months. That staff reported that B&R subsequently implemented several corrective actions, leading to improvements in the situation. This improvement was reported back to B&R in a subsequent meeting in May 1978. Oprea, *et al.*, ff. Tr. 1505, at 13-14; Tr. 5349-52, Tr. 5417-22 (Frazar, Oprea); App. Exhs. 44, 45.

123. Subsequently, in the summer of 1978, NRC officials from Region IV met with Mr. Turner and HL&P QA staff members to discuss concerns about the morale of B&R QC inspectors. In response, HL&P and B&R took several steps including: direction to the B&R Project QA Manager to spend more time in the field, increase in numbers of construction engineering personnel, changes in project procedures and increased HL&P surveillance of construction activities. Mr. Oprea testified that these changes, combined with those earlier in the year, appeared to improve the morale of QC personnel. Oprea, *et al.*, ff. Tr. 1505, at 15.

124. Yet another example of HL&P assuming responsibility for controlling activities of its contractors is the memorandum sent to B&R in the summer of 1979, expressing dissatisfaction with performance of the B&R site management. It detailed several specific deficiencies in management of STP construction and directed B&R to take corrective actions and report back to HL&P promptly. This stern document was sent after deliberation among HL&P managers and included a threat to consider other alternatives for completing the STP if B&R did not immediately show significant improvement in management, control and execution of its work. CEU Exh. 5; Tr. 5414-16, 5433-37 (Turner). The B&R response, dated 9 days later, itemized corrective actions taken and planned by it (App. Exh. 43).

125. Still another example — not primarily motivated by QA/QC problems but nonetheless strongly representative of HL&P's assumption of responsibility for the project — is HL&P's discharge of B&R and its replacement of that contractor with Bechtel and Ebasco. HL&P recognized that STP might never be completed if B&R remained and took strong action as a result. Goldberg, *et al.*, ff. Tr. 10,403, at 5-7; Tr. 10,413-18, 10,459-60, 10,467-69, 10,485-86, 10,521 (Goldberg). One Staff witness characterized the replacement of B&R as "a testimony to [HL&P's] character" (Tr. 10,082 (Hall)).

*(4) Extent to Which HL&P Failed to Keep Knowledgeable*

126. Another key question posed by the Commission in CLI-80-32 for review in this proceeding is "whether the facts demonstrate an unacceptable failure on the part of Houston to keep itself knowledgeable

*about necessary construction activities.*" The Commission observed that: "abdication of knowledge, whether at the construction or operating phase, could form an independent and sufficient basis for revoking a license or denying a license application on grounds of lack of competence (*i.e.*, technical) or character qualification on the part of the licensee or license applicant." 12 NRC at 291. The Board established Issue A(4) (*see* Finding 13) to examine the extent to which HL&P failed to keep itself knowledgeable. For convenience, we will evaluate that question here and will consider the effects of remedial actions taken by HL&P under Issue B.

127. Mr. Jordan did not feel that it was fair to state that HL&P failed to keep itself knowledgeable about STP activities. He and Mr. Oprea had communicated frequently and there had been nothing to indicate that there was a significant problem in QA/QC. The frequency of those meetings increased, as complexity of the project grew, to become daily occurrences. At the time of Mr. Jordan's testimony in May 1981, there were weekly HL&P management meetings which included updating of company executives on the STP situation by Mr. Oprea and Mr. Goldberg. In addition, Mr. Jordan met at the site at least once per month with the management committee, chief executives of each partner and B&R executives. He attended as many meetings as possible of the STP Management Committee, consisting of all the co-owners, and met periodically with B&R management to review the project status. Jordan, ff. Tr. 1224, at 8-10; Tr. 1261-66, 1372-75 (Jordan).

128. Mr. Jordan expressed the view that individual problems at the STP had been addressed adequately, but that there had been a lack of trending which might have indicated the deeper problems sooner. At the time of his testimony, he received more reports on STP activities and spent more personal time on the subject than ever before. He conveyed his intent to continue to stay in touch with other company officers concerned with the project, especially Mr. Oprea and Mr. Goldberg. In addition, he will receive regular written reports from the contractor and HL&P staffs, continue to attend meetings of the STP Management Committee and with contractor management, and will continue to attend significant hearings and proceedings related to the project. Jordan, ff. Tr. 1224, at 10-11; Tr. 1445-52 (Jordan).

129. Mr. Amaral testified that, at the time of the Bechtel audit, there was too little management involvement in assuring quality construction, and management was not knowledgeable enough about QA/QC (Tr. 1748, 1850 (Amaral)). He attributed this largely to problems with long communication lines and concluded that it subsequently was solved by HL&P changes in organization, especially moving Mr.

Frazar to the site and having him report directly to Mr. Oprea (Tr. 1851, 1897-1901 (Amaral)). *See* Finding 96-98, *supra.* The excessively long communication lines in effect prior to Investigation 79-19, coupled with the lack of experience of HL&P management personnel, resulted in HL&P management's failing to grasp the significance of information provided to it and, as a result, not being adequately knowledgeable about QA/QC problems which were occurring. *See* Tr. 9859 (Phillips); Tr. 9936-37 (Hayes); Tr. 1850-51 (Amaral).

130. Mr. Oprea stated that HL&P had taken appropriate steps to keep informed of day-to-day conditions at the site. HL&P employees monitored construction and QA activities at the B&R engineering offices and on site and participated in many meetings and personal contact with the B&R staff. Knowledge gained by them was communicated to management through committee meetings, correspondence and verbal contacts. Oprea, *et al.,* ff. Tr. 1505, at 47-48.

131. He indicated that there had been a substantial amount of communication and that he had received a large volume of information, including audit reports and I&E reports. Mr. Turner had kept him informed on issues that he considered to be significant and, as late as October 1979, he had felt that he was adequately informed about activities at the project. Two or three months later, Investigation 79-19 disclosed additional information and he discovered that some drastic problems existed at the site, causing him to be "dumbfounded" because of the change in the project in only 2 or 3 months. Tr. 2238-42 (Oprea). Mr. Amaral explained that Mr. Oprea had been receiving reports on individual problems that were occurring, but that he had not been getting the causes of those problems. Mr. Oprea agreed with that assessment. Tr. 2238-43 (Oprea, Amaral); *see also* Findings 109, 137.

132. The Staff Shewmaker panel (*see* Finding 19) testified that HL&P failed to maintain adequate knowledge of site activities to assure that QC problems were being properly identified and effectively corrected. It attributed that situation to inadequate involvement and over-dependence on B&R. Mr. Hayes subsequently explained that by "lack of involvement" the Staff meant that HL&P did not stay informed in enough detail about activities at the site, problems identified there, whether they were being corrected and whether steps were taken to prevent their recurrence. Shewmaker, *et al.,* ff. Tr. 9576, at 42; Tr. 9952 (Hayes).

*(5) Summary of Significant Evidence on Competence and Character of HL&P, as Reflected in Issue A*

133. The concerns expressed by the Commission in CLI-80-32 about the history of the South Texas Project were described as: "relevant to the issue of the basic competence and character of Houston." Questions posed by the Commission (*see* Findings 113, 126) were presented within the context of the greater overall question as to whether HL&P lacked the competence and character to be granted operating licenses. Accordingly, the main thrust of the Commission's direction to this Board was to evaluate the competence and character of HL&P as an applicant for operating licenses. 12 NRC at 291; Finding 11. The several parts of Issue A were explored in detail in Findings 13-132. This group of findings summarizes information and views compiled during testimony on Issue A that in our opinion bear most directly on the competence and character of HL&P.

134. Mr. Jordan, President and Chief Executive Officer of HL&P, indicated that he was acutely aware that this Board had been instructed by the Commission to inquire into the adequacy of HL&P's managerial competence and character to complete and operate the STP and that those questions are extremely serious. He testified that HL&P was fully aware of the absolute necessity for providing guidance and programmatic direction to its contractors. He was deeply disturbed by the findings of Investigation 79-19, but felt that company management appreciates the magnitude of the task at STP and is equal to it. Jordan, ff. Tr. 1224, at 3, 7, 10-11.

135. He stated that there never had been any question as to HL&P's corporate integrity (Jordan, ff. Tr. 1224, at 3). In his view, integrity involved running a business in a straightforward, honest and nondevious manner that allows respect by self and others. He testified that it is fair to measure a corporation by how it obeys laws and how it performs with respect to its business charge. Tr. 1295-96, 1380 (Jordan).

136. As for competence, Mr. Jordan observed that the company's record in providing service to its customers in a rapidly growing region speaks favorably for its competence. He did not feel that results of the NRC investigation indicated or implied that there was a lack of competence in the corporation. Tr. 1291-92, 1447-48 (Jordan). He stated that some competent persons on the project hadn't performed as well as they might have, but that their rapid movement to develop cures upon discovery of problems was a credit to them and to the corporation (Tr. 1448-52 (Jordan)). He concluded that HL&P had had a substantial team all along, but that the present (May 1981) team was much larger and had more technical competence (Tr. 1300-01 (Jordan)).

137. Mr. Oprea testified that he views any violation of NRC regulations as serious and does not respond to them in casual or cavalier fashion (Tr. 5294-96, 5303-04, 5323-27 (Oprea)). His first reaction to the Show-Cause Order was shock because, although he had expected some noncompliances, he did not anticipate that degree of severity. Subsequently, he recognized that there was a need to examine the QA program thoroughly to make certain that the requirements of Appendix B were fully recognized, understood and embraced by all concerned. In retrospect, he felt that the Show-Cause Order had helped HL&P focus on need to improve the QA program. Tr. 5463-64, 5468 (Oprea).

138. Mr. Oprea stated that any malfunction in any part of his organization concerned him, but that an organization with over 3000 persons on site must expect to have isolated incidents. Throughout his years of experience, he has believed that problems should be solved promptly because even small problems may take on unreal proportions if allowed to smolder for a few weeks. Tr. 5471-74 (Oprea).

139. Mr. Oprea stated that, although the Show-Cause Order was very helpful to HL&P in initiating action, before Investigation 79-19 he already had been considering bringing in an outside party to audit the QA program. He testified that Bechtel and Mr. Amaral would have been brought in anyway for that purpose. Accordingly, they might have found the same information that came out of the NRC inspection, but perhaps at a slightly later date. Tr. 5464-68 (Oprea).

140. Mr. Oprea did not believe that there ever had been a breakdown in the QA program, even including Investigation 79-19 (Tr. 2229-30 (Oprea)). Mr. Phillips, Staff Resident Reactor Inspector, expressed a similar view (Tr. 9851-52 (Phillips); *see* Findings 86, *supra,* and 155, *infra*).

141. Mr. Amaral, who led the Bechtel audit, stated that when Bechtel initially evaluated the STP, HL&P had enough or, if anything, more inspectors than would normally be employed for the amount of project activity. He indicated that when an inspection force becomes large one can suspect that problems exist in construction and that the extra inspectors were hired to attempt to solve them. This was the case at the STP. Tr. 1966-67 (Amaral). Mr. Goldberg observed that the excessive QC staffing level was in part the result of B&R's lack of experience. He characterized B&R's efforts as typical of a first-generation nuclear project and opined that B&R's performance would have been better had it been B&R's third or fourth such effort. Tr. 10,476 (Goldberg). *See also* Finding 118, *supra.*

142. Mr. Amaral testified that, before the Show-Cause Order, the QA/QC personnel in HL&P and B&R did not all possess the professional

credentials and experience desired for that type of work. At the time of the Bechtel audit, HL&P did not, in his judgment, have an adequate and experienced enough staff to provide programmatic direction for the project. He had recommended acquisition of qualified quality professionals and, as an interim step, that HL&P should acquire the services of an outside organization with the required expertise. Its personnel could be integrated into the HL&P and B&R organizations until permanent personnel could be obtained. That was done. Finding 212, *infra*; Oprea, *et al.*, ff. Tr. 1505, at 121-22; Tr. 1744-45, 1905, 2228-29 (Amaral).

143. Mr. Amaral stated that the written QA/QC program had met the requirements of Appendix B and would have been satisfactory if properly implemented. The problem was that it wasn't. Mr. Frazar agreed with that assessment. The modified program analyzed during a recent Bechtel audit did meet the requirements, including implementation, and could be classified as about the same as programs of other successfully constructed plants. Mr. Amaral stated, further, that some elements of the program were novel and should be looked at by others. Tr. 1917-20 (Amaral); Tr. 1792-1814 (Frazar).

144. Mr. Oprea did not have differences with the testimony and recommendations of Mr. Amaral concerning personnel. HL&P instituted several changes in accordance with those recommendations, including transfers of some personnel to different positions and additional training of others. Tr. 2236-38 (Oprea); Tr. 1932-40, 1996-2001, 2065-67 (Amaral).

145. Mr. Amaral felt that the attitude of Mr. Oprea and Mr. Turner was positive and could be characterized as a strong desire to overcome the QA problems (Tr. 1966 (Amaral)). His judgment was that Mr. Oprea wanted the best for the project and that his response to the Bechtel input was very satisfying. Most of the Bechtel recommendations had been implemented at the time of the hearing. Tr. 2247-48 (Oprea, Amaral).

146. The Staff's Seidle panel testified that HL&P was cooperative and diligent in correcting problems identified by NRC investigators (Seidle, *et al.*, ff. Tr. 9205, at 64). In response to Board questions, Mr. Seidle stated that in his contacts with HL&P management he could recall no case in which the managers with whom he dealt were not responsive and totally committed to quality assurance and quality control. He did not see any effort by the licensee or its contractors to orchestrate anything that would be anti-QA/QC. Tr. 9506-07 (Seidle). Initially, he had thought that the QA organization was receiving adequate backup from management, but general comments in allegations received by Region IV inspectors raised doubts in his mind. Those allegations led to

a meeting in August 1978, between NRC and HL&P management personnel to discuss the concerns, including alleged problems in the implementation of the site QA/QC civil program, QC inspector morale, and the adequacy of site QA/QC staffing. At the time of that meeting, B&R QA was understaffed by approximately nineteen or twenty persons and HL&P QA was understaffed by two persons. Mr. Seidle did not feel that there was anything contrived in the alleged lack of support, but that poor communication and inadequate feedback caused the perception among QC inspectors that management was not responding to their reports. Seidle, *et al.*, ff. Tr. 9205, at 27-28; Tr. 9277, 9518-21 (Seidle); Staff Exh. 9 (I&E Rept. 78-13).

147. Mr. Taylor's perception was that both HL&P and B&R desired to have a viable QA system, but were limited by lack of experience in nuclear construction. He did not detect any unwillingness on the part of HL&P QA personnel to do the proper things. Tr. 9507-12 (Taylor).

148. Mr. Phillips stated that HL&P management appeared to be dedicated to have not just a routine QA program but a model one. As set forth earlier (Finding 121), he viewed them as cooperative and probably the most open licensee that he had ever dealt with. He agreed with Mr. Taylor that inexperience was the factor that probably contributed most heavily to the problems. Tr. 9517 (Phillips).

149. The Board asked whether anyone on the panel had found anything in HL&P's responses to personnel changes that would reflect unfavorably on its corporate character or competence. Mr. Seidle and Mr. Crossman volunteered negative responses and no panel member disagreed. Tr. 9527.

150. In response to a Board question, Mr. Seidle stated that a review of the testimony and exhibits supporting it had suggested that the HL&P managerial systems might have been breaking down. HL&P's inability to control construction activity had made it appear that there might be a question of HL&P's managerial competence. Tr. 9539-40 (Seidle).

151. Mr. Hayes testified that the record of HL&P as disclosed during the 79-19 Investigation was "poor." Mr. Shewmaker and Mr. Phillips agreed with that evaluation. Tr. 9726-27 (Hayes, Shewmaker, Phillips). They indicated that they were not aware of any specific NRC statement or policy or regulation that defines the specific attributes to use in evaluating whether an applicant has the required "character" to be granted an operating license (Tr. 9740 (Hayes, Shewmaker, Phillips)). Mr. John W. Gilray, QA reviewer from the Office of Nuclear Reactor Regulation (NRR), also perceived HL&P's implementation of its QA program to have been "poor" and not in accord with the requirements of 10 C.F.R. Part 50, Appendix B (Tr. 10,704 (Gilray)).

152. The Shewmaker panel judged that HL&P's record was not sufficiently poor to conclude that it lacked the necessary managerial competence or character to be granted operating licenses. It indicated that lack of involvement by HL&P in construction activities was a reason for the problems observed, and attributed that to inexperience in nuclear construction rather than irresponsible corporate management. Further, no irreparable construction deficiencies were found in any of the construction already completed. Also, HL&P had shown a willingness to implement corrective actions. The panel indicated that although shortcomings in HL&P management of construction are relevant to, and probative of, how it will perform under an operating license, such prior behavior should not be determinative. Shewmaker, *et al.*, ff. Tr. 9576, at 49; *see* Finding 248, *infra.*

153. Those observations led the Staff to the opinion that HL&P's record prior to Investigation 79-19 was insufficient by itself for the Staff to conclude that HL&P did not have the necessary managerial competence and character to be granted an operating license. Further, assuming implementation of the remedial steps ordered by the NRC and proposed by HL&P, the Staff believed that the STP would be in compliance with the NRC requirements for an operating license. Shewmaker, *et al.*, ff..Tr. 9576, at 49-50.

154. In response to questions about the reasons for reaching that conclusion, Mr. Phillips explained that it was the result of a complex evaluation process, on which he subsequently elaborated as summarized in Findings 155-163, *infra* (Tr. 9848-51, 9875 (Phillips)).

155. Mr. Phillips indicated that there had been failures to meet, maybe, eleven of the eighteen criteria in Appendix B, but that there had not been a total breakdown in the QA program. The panel considered all of the nonconformances reported in 79-19 (*see* Finding 70, *supra*) and concluded that none had been at a severity level that constituted being irresponsible and none had been deliberately caused by management. None had resulted in irreparable construction deficiencies. Tr. 9853-54, 9859 (Phillips).

156. Management had not been deceptive in any way during or after the inspection and was not unwilling to correct any deficiencies when pointed out. He concluded that the character of HL&P management was good because they demonstrated responsibility in several ways and tried to obey the Code of Federal Regulations. Tr. 9854 (Phillips).

157. HL&P often exceeded the minimum requirements and commitments that had been made to NRC. Mr. Phillips considered that to be another demonstration of its inexperience, causing it to do things to excess, just as in other instances lack of experience had led HL&P to fail

to perform entirely up to some requirements. His perception, based on dealing with HL&P daily as Resident Inspector, was that it sincerely wanted to build a quality nuclear facility and placed the health and safety of the public first. Tr. 9855 (Phillips).

158. HL&P's attitude was always good in assisting with NRC inspections or investigations. It provided the necessary information, even when detrimental to the company. Its record of identifying and reporting construction deficiencies under 10 C.F.R. § 50.55(e) was open and honest and probably was better than any utility that Mr. Phillips had ever seen. Tr. 9855 (Phillips); *see* Finding 121, *supra.*

159. HL&P had thought that its program was very good, until it learned differently during Investigation 79-19. Then, it took the extraordinary step of asking NRC Region IV for a meeting in December 1979, in which it proposed to take corrective actions for undesirable situations on which noncompliances had not even been formulated. In that meeting, it voluntarily decided to stop complex concrete pours until it could assess the situation, although noncompliances were not issued until April 30, 1980. Tr. 9855-57, 9859 (Phillips).

160. HL&P retained Bechtel in January of 1980 to conduct an in-depth audit of its QA program and Woodward-Clyde Consultants to investigate backfill conditions. It tried on its own, unsuccessfully, to investigate the allegations of harassment of QA inspectors and subsequently engaged a consultant to determine the reasons for recurring problems in that area. Tr. 9857-58 (Phillips).

161. HL&P conducted its own special audit immediately after completion of the onsite NRC investigation and identified additional deficiencies. It stopped welding in April 1980, and engaged a consultant to review that area. HL&P responses to essentially all NRC reports were responsible, good and cooperative, and were followed by corrective actions. Mr. Phillips indicated that all of those actions were voluntary, before issuance of Investigation Report 79-19, and he interpreted them as strong evidence of good character. Tr. 9858-59 (Phillips); *see* Finding 198, *infra.*

162. When it realized that its audit program was not functioning as effectively as it should and that the problem could be attributed to poor communications and separation of management from the site, HL&P moved to improve the situation by shifting key Houston management personnel to the STP site on a full-time basis. Also, Mr. Phillips testified that subsequently it was not unusual for Mr. Oprea to stop by his (Phillips') office and inquire as to whether any problems had been perceived by him. For one of the very top officials of the company to do that was interpreted by Mr. Phillips as positive evidence of corporate

character. Tr. 9860 (Phillips). Mr. Oprea stressed to the Board his willingness and desire to communicate with NRC about project developments (Tr. 1948-49 (Oprea)). *See also* Tr. 5229-30 (Oprea, Frazar). We also regard these expressions of intent by HL&P to represent a positive character trait.

163. Mr. Phillips stated that, for each failure in the program, he could identify many instances in which there were not failures (Tr. 9860 (Phillips)). Mr. Shewmaker followed up on that thought by pointing out that normal inspection reports tend to be negative, not listing everything that is correct although there normally are more compliances than noncompliances in site activities (Tr. 9861 (Shewmaker)).

164. Mr. Shewmaker indicated that, based on his observations, the attitude of high management of HL&P was to resolve problems as rapidly as possible. He viewed HL&P as willing and desiring to meet NRC requirements. HL&P officials frequently phoned him to ensure that their planned actions would meet NRC interpretations and intent of the regulations. Tr. 9861-63 (Shewmaker).

165. As stated earlier with respect to alleged false statements in the FSAR, there was no question of HL&P deliberately trying to confuse anybody or any willfulness on their part to include anything in the application that subsequently was not done in the field (Finding 32).

166. In response to direct questions by the Board, each member of the panel confirmed that he did not disagree with any of the comments of the other members, which have been summarized in Findings 154-165, *supra* (Tr. 9864-67 (Shewmaker, Hayes, Phillips)).

167. All members of the panel felt that their contacts with HL&P management were sufficient and at high enough management levels to be able to reach reliable conclusions about managerial competence and character in the organization (Tr. 9946-48 (Hayes, Shewmaker, Phillips)).

168. The panel indicated that HL&P personnel were very open and candid with them and gave full cooperation. HL&P did not exhibit any reservation about the role and responsibility of NRC in regulating the project. Tr. 9947-48 (Hayes, Shewmaker, Phillips). HL&P has been generally responsive to Staff inquiries and anxious to keep the Staff informed about project developments (Tr. 10,067, 10,075, 10,085-86 (Crossman, Phillips, Tapia)). This view of HL&P's openness and candor was confirmed by the report of NRC's Systematic Assessment of Licensee Performance (SALP) for the period July 1, 1980–June 30, 1981, which judged HL&P's responses to requests for information to be "timely and of good quality" and as demonstrating "achievement of su-

perior safety performance" (Category 1) (Staff Exh. 133 (I&E Rept. 81-37, at 2, 5)).

169. There was no evidence of attempts to cut corners to economize in construction activities at the site. To the contrary, HL&P was carrying out testing and other activities over and beyond the minimum requirements. Tr. 9949-50 (Phillips, Shewmaker).

170. Panel members agreed that the absence of irreparable construction deficiencies was not just a matter of luck. The procedures and QA program and efforts exerted resulted in most of the work being done properly and structures being well-constructed, with deficiencies only in some areas. Tr. 9957-58 (Shewmaker, Phillips, Hayes). Their concern had been that several weaknesses in the program had been identified and had persisted for some time. With more critical work about to be started, they felt that it was necessary for those weaknesses to be corrected. As regulators, they could not sit back and take a chance that the situation would straighten itself out, but had to ensure its correction. Tr. 9958-60 (Hayes, Shewmaker).

171. In response to Board questions, Mr. Hayes stated that failure of HL&P to stay informed in enough detail about activities at the site did not mean that from a technical point of view it was incompetent. He indicated that HL&P had the technical expertise to deal with each of the problems individually but lacked the experience to apply that expertise across the board to other activities as well. Tr. 9952-54 (Hayes).

172. Mr. Shewmaker did not think that one can totally separate experience from competence, but that experience is only one of a number of factors that must be considered in evaluating competence of the company. He cited a different factor as a positive example — the fact that HL&P realized in many instances that the best way to solve a problem was to bring in outside experts with a great deal of competence in the field. He expressed the view that an important management function is to recognize capabilities of the company organization and personnel and get additional resources when appropriate. Tr. 9954-55 (Shewmaker).

173. Mr. Phillips concurred with the views expressed by Mr. Hayes and Mr. Shewmaker. He pointed out that any professional person can do less than an adequate job at times in isolated instances, or can fail to take appropriate action, without being incompetent. That is the way in which he viewed the HL&P management situation — management had failed to maintain good communications, but was still competent. Tr. 9956-57 (Phillips).

*(6) Board Conclusions on HL&P's Character and Competence, as Reflected in Issue A*

174. The task assigned to this Board by the Commission of evaluating the character and competence of HL&P as a license applicant is rendered difficult by the lack of definitive criteria in NRC regulations, policies and decisions identifying the attributes of character and competence that constitute acceptable credentials for a license applicant. Accordingly, it has been clear throughout that we must consider very carefully all evidence compiled during the extensive hearings, including views of witnesses and participants about HL&P character and competence, and their reasons for holding those views, and then exercise our judgment in arriving at a decision. Findings presented above on Issue A distill the testimony to present its essence pertinent to the question before the Board. We now proceed to our conclusions about the character and competence of HL&P, based on our interpretation of those findings.

175. Examination of the findings for Issue A to evaluate character reveals that HL&P was cooperative, truthful, and straightforward in its dealings with the Staff. No instance was reported in which company personnel attempted to deceive or mislead. In no instance did the Staff find that HL&P withheld information because of potential adverse findings.

176. Staff witnesses stated that the company appeared to be dedicated to having a model program and often went beyond NRC requirements in its efforts to do so. It consistently was willing to accept NRC authority and requirements and was not unwilling to correct deficiencies when identified. In fact, during the 79-19 Investigation, HL&P initiated action to correct deficiencies even before 79-19 noncompliances had been issued. Immediately upon completion of Investigation 79-19, HL&P initiated its own further investigations, which disclosed other problems that it reported to NRC. There was no evidence of attempts to economize in construction at the expense of safety or quality.

177. Upon consideration of all of the evidence, the Board finds that the instances of noncompliance set forth in the Notice of Violation and the Order to Show Cause are insufficient to determine that HL&P does not have the necessary character to be granted licenses to operate the STP.

178. Turning to competence, we view that term to include both the qualification and training of project personnel and the experience of those personnel in activities comparable to that under consideration here — *i.e.*, the construction of a large nuclear plant.

179. A review of all of the testimony on Issue A indicates clearly that the STP program before and during Investigation 79-19 possessed deficiencies, as evidenced by inspection reports, noncompliances issued

by NRC and testimony presented to the Board by the many Staff and HL&P witnesses. However, none of the noncompliances was at the level of a "violation," which would have indicated that the function or integrity of a safety system was lost. None resulted in irreparable structural defects. None resulted from deliberate management actions or encouragement to avoid compliance with NRC regulations or to economize on facility cost at the expense of safety or quality of construction.

180. The weight of the evidence convinces us that STP construction to which our attention was directed has been accomplished, for the most part, using sound procedures and has produced good quality structures meeting or exceeding applicable safety requirements. We conclude that successful execution of those complex activities over several years was not the work of a totally incompetent organization. Nor, as pointed out by a Staff panel, can it be attributed to mere luck (Finding 170). The testimony of outside and Staff experts is convincing that deficiencies described in the inspection reports and noncompliances were exceptions rather than the rule on the project (Finding 163). These observations must not be construed as an attempt on our part to minimize either their importance or the vital need for HL&P to avoid similar errors and problems in the future. There is no denying that the errors found on this project have been too many and too serious to allow them to continue without strong regulatory action. That notwithstanding, it is important that we distinguish between errors committed by competent personnel and incompetence itself (Findings 171-173).

181. In spite of detailed probing in extensive cross-examination and by the Board in its questioning, not one witness expressed the view that the past record of HL&P demonstrated that its managerial competence was inadequate to receive operating licenses. To the contrary, those witnesses expressing a view on the subject concluded that, even without considering the remedial actions to be discussed in Issue B, HL&P did have the required competence. We note, however, that with respect to one aspect of competence — experience in constructing large nuclear plants — both HL&P and B&R demonstrated important deficiencies.

182. Based on careful evaluation of all of the testimony, the knowledgeability and demeanor of HL&P witnesses, and conclusions of Staff witnesses who participated in detailed investigations and inspections of HL&P and its contractors, we are convinced that the noncompliances found before and during Investigation 79-19, although reflective of deficiencies in experience (one component of competence), and although indicating that HL&P's competence was questionable in this one respect, do not demonstrate inadequate organizational competence of HL&P.

Instead, we view them as problems arising from various reasons discussed in many of the above findings, but especially lack of experience in nuclear construction, inadequate involvement of management in activities at the STP, and an excessively long chain of command between QA/QC inspectors at the site and upper management. Moreover, where particular personnel proved inadequate to their assigned tasks, they were replaced or transferred to other tasks more suited to their capabilities. To that extent, HL&P took steps to mitigate the prime area of competence in which it was weak.

183. Accordingly, the Board finds that the instances of noncompliance set forth in the Notice of Violation and the Order to Show Cause, although demonstrating a weakness in one aspect of competence, are insufficient to determine that HL&P does not have the necessary managerial competence to be granted licenses to operate the STP.

184. With respect to the question of abdication of authority, the Board finds that in some instances HL&P left too much responsibility in the hands of B&R for certain phases of the STP program. Based on evidence presented in this proceeding, we find that the lapses in project control reported by the Staff were not caused by lack of either technical competence or of a sense of responsibility on the part of HL&P. The principal reasons for those failures were based on lack of experience in management of nuclear construction and poor communications brought about by an excessively long chain of command between field QA/QC personnel and corporate management in Houston (Findings 95-98, 106-112). In other instances, the utility exerted clear and forceful control over its contractor, as illustrated by examples cited in Findings 121-125. As set forth in Finding 96, HL&P recognized its lack of experience and the excessively long chain of command and took steps to remedy those deficiencies. (*See* Issue B, *infra,* for our evaluation of those steps.)

185. The evidence dealing directly with competence and character, summarized in Findings 133-173, included consideration of the question of abdication of authority. Based on that evidence, the Board finds that the instances in which too much authority was left to B&R resulted primarily from HL&P's lack of experience and are insufficient to determine that HL&P does not have the necessary managerial competence or character to be granted licenses to operate the STP.

186. With respect to the failure of HL&P to keep itself knowledgeable about construction activities, the Board finds that, prior to the 79-19 Investigation, there was too little management involvement in the site program and that management was not sufficiently knowledgeable about QA/QC activities (Findings 126-132). Management received much factual information about construction activities but often was unable to

evaluate it properly and in a timely fashion. Based on the record in this proceeding, we conclude that this situation also was caused by inexperience in nuclear construction and poor communications because of an attenuated chain of command between the site and upper echelon management in Houston. Through hiring of new personnel and organizational modification, however, HL&P took steps to alleviate these deficiencies. We will evaluate those steps under Issue B.

187. The evidence dealing directly with competence and character, summarized in Findings 133-173, included consideration of the knowledgeability of HL&P management about construction activities. Based on that evidence, the Board finds that the instances in which Houston management did not keep itself adequately knowledgeable reflect a defect in competence which, if not remedied, would raise serious questions of HL&P's eligibility for operating license; but that, taking into account the fact that corrective actions were taken (but without regard to the effectiveness of those corrective actions), the instances are insufficient in themselves to support a determination that HL&P does not have the necessary managerial competence or character to be granted licenses to operate the STP.

### *Issue B: Adequacy of HL&P's Remedial Actions*

188. Issue B states:

> Has HL&P taken sufficient remedial steps to provide assurance that it now has the managerial competence and character to operate STP safely?

189. As background for Issue B, the number, types and severity of program deficiencies are vitally important in evaluating competence and character of the company and could result in denying an operating license if found to be sufficiently serious or incurable. The Board has studied that aspect of the matter under Issue A and has determined that the deficiencies were insufficient in themselves (considering number, types, and severity) to determine that HL&P does not have the necessary competence and character to be granted licenses to operate the facility, particularly given the willingness and practice of HL&P to adopt corrective actions for perceived deficiencies. *See* Findings 35, 87, 177, 183, *supra.*

190. An equally important consideration in evaluating company competence and character is the adequacy of its responses to observed deficiencies and of steps taken by it to remedy them. HL&P responses and several remedial steps taken by its management have been outlined

in our earlier findings on various subparts of Issue A and in our evaluation of HL&P's competence and character based on Issue A questions. In considering Issue B, we will occasionally refer to those earlier findings and supplement them with information on additional remedial steps, the effectiveness of all such steps, and their significance in evaluating HL&P competence and character.

191. Board examination of the question of alleged false statements in the FSAR (Issue A(1)) has been summarized in Findings 14-34, *supra*. As set forth therein, our finding was that, with one exception which we regard as of little significance (Finding 25), the statements were not false when made; and, further, that there was neither intent by HL&P to deceive the Commission nor disregard for the truth. The remedial actions taken by HL&P, with the concurrence of the NRC Staff, included modifying some parts of the FSAR and changing certain QA procedures to bring field activities and FSAR commitments into agreement with each other. Those changes have been made. They also did not compromise the degree of protection of the public health and safety represented by the earlier FSAR commitments and construction procedures.

192. Noncompliances issued before Investigation 79-19 were summarized in Findings 51-64. Appropriate remedial steps were taken by HL&P in each instance, as documented by subsequent Staff inspections describing the actions and ultimately closing the record on each noncompliance. Staff witnesses testified that HL&P was responsive and followed up promptly and actively on the identified problems and sought other problems where they existed. They were cooperative and open in their dealings with NRC personnel. *See* Findings 58, 61, 168, *supra*.

193. Findings 122-124 describe examples of remedial actions taken by HL&P to correct deficiencies in B&R conduct of construction activities at the site. In each instance positive results were reported to result from those actions. With B&R no longer serving as construction contractor, the specific procedures and other corrective actions adopted by B&R are no longer in effect; but HL&P intends to assure that the root causes underlying the changes it brought about in the B&R program will be adequately considered by the replacement contractors. Tr. 10,458-59 (Goldberg); Finding 125; *see also* Staff Exh. 131 (I&E Rept. 81-36, at 6-7); Tr. 10,090-93 (Phillips, Crossman, Hubacek).

194. Mr. Seidle testified for the Staff that after a meeting between HL&P and NRC, in August 1978, to discuss morale of B&R QC inspectors, management was more responsive to complaints and more accessible to personnel on site. An inspection in the fall of 1979 showed several improvements in access of site personnel to upper management,

feedback from management to field personnel and the overall working relationship between QA/QC and construction personnel. Tr. 9547, 9558-59 (Seidle); Staff Exh. 27 (I&E Rept. 79-13, at 27-28).

195. In a meeting on preliminary findings from Investigation 79-19, held on December 21, 1979, HL&P was informed by NRC personnel that there were serious problems in the QA/QC program and construction practices for concrete placement and that QC inspectors had experienced harassment. On December 28, 1979, HL&P voluntarily committed to stopping placement of safety-related concrete and executing a nine-point program to correct conditions found by the Staff in its inspection. Details about those commitments were outlined by Mr. Frazar. Oprea, *et al.*, ff. Tr. 1505, at 20, 78-80; *see* Finding 159, *supra.*

196. In early January 1980, HL&P assigned additional people from Houston to the STP site to make them more directly and visibly involved in work there and to support the continuing NRC investigation. Oprea, *et al.*, ff. Tr. 1505, at 79-80; App. Exh. 1; Tr. 2226-27 (Oprea). *See* Findings 83, 96, 162, *supra.*

197. In the NRC exit interview for the special investigation on January 24, 1980, HL&P was informed of further details of noncompliances related to welding, nondestructive examination (NDE), backfill activities and QA program deficiencies. Within 2 weeks, in response to this information, HL&P committed to further actions to improve conditions at STP, including: improved trending of nonconformances, revised audit activities, improved training, changes in the welding, NDE and backfill placement programs, and an independent audit of the QA/QC program by an outside consultant, to be conducted at least every 18 months. NRC subsequently was advised of progress on those changes. Oprea, *et al.*, ff. Tr. 1505, at 21-23, 80-83; App. Exh. 3.

198. Upon completion of the NRC onsite investigation, HL&P initiated its own special audit and identified additional deficiencies. A stop-work order on safety-related welding was issued by the B&R Power Group QA Manager on April 11, 1980. Saltarelli, *et al.*, ff. Tr. 7536, at 19; *see* Finding 161, *supra.*

199. After Investigation 79-19, HL&P revised its QA/QC auditing procedures. Reports on subsequent I&E inspections described improvements in this area. *See* Finding 117, *supra.*

200. When the NRC reported its preliminary findings from the special investigation in late 1979, HL&P management recognized the greater breadth and seriousness of the problems and increased movement that had been initiated earlier to retain a consultant to conduct an independent evaluation of the QA/QC program. Oprea, *et al.*, ff. Tr. 1505, at 18-19, 24-25, 119-26; Staff Exh. 48 (Licensee's Response to Order to

Show Cause), Exhibit 1; Tr. 1363-64 (Jordan); Tr. 2084-99, 2104-12, 5462-74 (Oprea).

201. After investigations and interviews of potential candidates, Mr. Oprea, with concurrence of Mr. Jordan, determined that Bechtel Corporation had the required capabilities and experience to undertake an outside independent audit of the HL&P QA program. In January 1980, Bechtel was retained for that purpose, under direction of Mr. Amaral. It was virtually given a "blank check," as described by Mr. Oprea, in conducting the study and making recommendations concerning the program. Mr. Amaral testified that his organization received good cooperation from HL&P and B&R personnel during the audit and that no pressures were placed on them with respect to how the audit should be conducted. Oprea, *et al.*, ff. Tr. 1505, at 23; Tr. 1364 (Jordan); Tr. 1844-47 (Amaral); Tr. 2084-88, 2251-55 (Oprea); *see* Finding 93, *supra.*

202. The Bechtel Report identified six "root causes" for deficiencies in the QA program. The underlying root cause was perceived to be lack of visibility and participation by management, much of which could be attributed to communication problems between field personnel and upper management. HL&P and Staff witnesses agreed with that diagnosis. Much of the HL&P remedial program thereafter was based on correcting the root causes identified in the Bechtel Report. The knowledgeability of management about STP activities and flow of information from field to management and feedback to the field personnel improved thereafter. *See* Findings 94-98, 104, 108-112, 127, 129, 143, *supra.*

203. Upon issuance of I&E Report 79-19, the Notice of Violation and Order to Show Cause on April 30, 1980, Mr. Frazar was assigned responsibility for investigating the twenty-two items of noncompliance and responding to the Notice of Violation. On May 23, 1980, HL&P responded to the Notice of Violation, admitting to the noncompliances and identifying six areas to which problems in the QA program could be traced and in which improvements would be made. Oprea, *et al.*, ff. Tr. 1505, at 25-28, 54-55, 84; Staff Exh. 47 (HL&P Response to Notice of Violation); *see* Findings 71-74, 94, *supra.*

204. Subsequently, Mr. Briskin directed an expanded task force in analyzing and responding to eight of the ten items addressed specifically in the Order to Show Cause. Mr. Frazar and the QA group were assigned responsibility for responding to two items concerning QA activities. Item No. 1 required that HL&P utilize an independent expert consultant to evaluate management of the QA program and consider several specific alternatives for future QA program organization. Bechtel was assigned responsibility for carrying out that analysis. HL&P's response to NRC on those matters is presented in the Licensee's Response to Order to

Show Cause (Staff Exhibit 48) and contained commitments for corrective actions to be taken by HL&P. Additional commitments were made by HL&P during a public meeting in Bay City on August 19, 1980 (Oprea, *et al.*, ff. Tr. 1505, at 73-74).

205. Mr. Briskin's testimony, updated through May 8, 1981 (and introduced into evidence the following week), included a summary of 236 HL&P commitments and their status at that time — 210 had been closed in writing, another 8 closed verbally, 16 were ready for reviews, and 2 were not then ready for review. Procedures followed by the task force, its findings, and actions taken subsequently are discussed in testimony and cross-examination of Messrs. Oprea, Briskin, Frazar, and four panels testifying on backfill, concrete and welding activities. Oprea, *et al.*, ff. Tr. 1505, at 23-31, 36-49, 53-76 and separate "Show Cause Commitments" (dated May 8, 1981), 83-117; Pettersson, *et al.*, ff. Tr. 5796; Wilson/Kirkland, ff. Tr. 2697; Murphy, *et al.*, ff. Tr. 6327; Fraley, *et al.*, ff. Tr. 7241; Saltarelli, *et al.*, ff. Tr. 7536. Cross-examination and Board examination of these panels is at Tr. 1543-2298, 3360-3630, and 5063-5544 (Oprea, *et al.*); Tr. 5919-6137 (Pettersson, *et al.*); Tr. 2710-2896 (Wilson/Kirkland); Tr. 6329-6432 (Murphy, *et al.*); Tr. 7242-7505 (Fraley, *et al.*); and Tr. 7544-7817 (Saltarelli, *et al.*).

206. In response to Item 1 of the Order to Show Cause, Bechtel surveyed five NRC-specified alternatives for future QA/QC management organization and one additional alternative. Its report to HL&P described advantages and disadvantages of each alternative, emphasizing that origins of the problems were in the "root causes" identified in the Bechtel Audit Report and that the organizational structure was of lesser concern. The alternatives and recommendations of Bechtel and from the Management Analysis Corporation (MAC), another consultant retained by HL&P, were evaluated by Mr. Oprea, in consultation with Mr. Frazar. They then were reviewed with Mr. Jordan before reaching the final decision to adopt an alternative that was basically similar to the existing organization. The alternative selected included certain modifications in procedures, a strengthened HL&P role in the QA program at STP, and greater management involvement in STP activities. Some of the specific changes that were made are outlined in Findings 208-214, *infra*; Oprea, *et al.*, ff. Tr. 1505, at 31-33, 120-26; Staff Exh. 48 (Licensee's Response to Order to Show Cause), Exhibit 1 (Bechtel Report). As set forth in our Opinion (note 37, *supra*), because of the transition to Bechtel and Ebasco, the recommended organizational structure is not being followed.

207. Mr. Jordan testified that the causes underlying problems revealed by Investigation Report 79-19 indicated a need for management

improvements, especially expansion and restructuring of the QA/QC organization. He participated actively in discussions with Mr. Oprea and decisions concerning the QA organizational alternatives studied in the Bechtel report and in key changes in assignments of personnel. Jordan, ff. Tr. 1224, at 6-8 (Jordan). Details of those activities are discussed in findings that follow.

208. As part of the HL&P response to the Order to Show Cause, Mr. Jordan assigned Mr. Oprea, Executive Vice President and the company's most senior engineering-oriented executive, to full-time supervision of the company's nuclear activities. That had the effect of having Mr. Oprea devote almost all of his attention to the STP. Jordan, ff. Tr. 1224, at 7; Tr. 1270-73, 1318-19 (Jordan).

209. Mr. Jordan approved creation of a new position — Vice President, Nuclear Engineering and Construction, reporting to Mr. Oprea — and actively participated in recruiting Mr. Jerome H. Goldberg to fill it. Mr. Goldberg was first contacted by HL&P in August 1979, but did not begin serious negotiations (because of his own unavailability) until the spring of 1980. He joined HL&P in October 1980. Mr. Goldberg added significantly to HL&P's capability because of his 26 years' experience in nuclear engineering, including 17 years as a manager. He was a former vice president of Stone and Webster. Jordan, ff. Tr. 1224, at 8; Tr. 1270, 1273-74, 1317, 1319-25 (Jordan); Tr. 859-60, 910-39, 944 (Goldberg); Goldberg/Frazar, ff. Tr. 906, at 3-4; Tr. 9527 (Crossman).

210. Mr. Jordan agreed with Mr. Oprea about the need for stronger QA/QC organizations in both HL&P and B&R and approved moving the head of the corporate QA department to the STP site with full-time responsibility for the Project, reporting directly to Mr. Oprea. This shortened the chain of command by eliminating four layers of QA supervision between the onsite supervisor and Mr. Oprea, giving closer management supervision and greater independence of QA from construction scheduling and costs. Jordan, ff. Tr. 1224, at 7; Tr. 1897-1901 (Amaral); *see* Findings 96-97, *supra.*

211. Mr. Jordan testified that HL&P was actively searching for a Vice President of Nuclear Operations to serve under Mr. Oprea and take over preparations for that phase of the program as soon as feasible. That since has been accomplished through employing Mr. Dewease (*see* Finding 230, *infra*). Brown & Root employed an experienced replacement to head up its QA group. Mr. Jordan testified that HL&P had not attempted to replace its Corporate QA Manager because of confidence in the abilities of Mr. Frazar. Subsequently, a new Corporate QA Manager for STP and a new Project QA Manager have been employed, and

Mr. Frazar was moved into another position in the company. *See* Findings 213, 252, *infra*; Tr. 1444-45, 1454-56 (Jordan); Tr. 1768 (Oprea).

212. About fifteen or sixteen persons from the Management Analysis Corporation (MAC) were brought into the HL&P and B&R organizations to fill QA/QC needs. Those additions allowed 100 percent attainment of Bechtel's recommendations to HL&P concerning the strengthening of the qualifications of QA/QC personnel, as an interim move until permanent qualified and experienced personnel could be recruited for the positions. Mr. Amaral's recommendation to Mr. Oprea to strengthen the Corporate QA Manager position was initially fulfilled by adding a MAC employee, Mr. Zwissler, to serve as backup for Mr. Frazar until a new site QA Manager could be recruited. Oprea, *et al.*, ff. Tr. 1505, at 121-22; Tr. 1766-69, 1786-91, 1901-07, 2069 (Amaral, Frazar). (As set forth in Findings 211 and 213, that recruitment has been accomplished.)

213. Mr. Oprea later identified Mr. James E. Geiger as the new employee who was to head QA for the STP. Mr. Geiger has 22 years of QA experience, including employment by Bechtel as QA Manager for the San Onofre Project. Mr. Geiger was later promoted to Corporate QA Manager for STP and was replaced as Project QA Manager by Mr. Al Walker, who had had 9 years' QA/QC experience in nuclear plant construction. Oprea, *et al.*, ff. Tr. 1505, at 42; Tr. 5063-66, 5378-83 (Oprea); Geiger, *et al.*, ff. Tr. 10,580, at 1-3; Tr. 10,583 (Geiger).

214. The involvement of HL&P and B&R senior executives in STP QA activities was enhanced by increasing their participation in meetings and increased review of reports on STP QA activities. Also, the depth of HL&P reviews of B&R QA activities was increased, and enhanced programmatic direction was provided at all levels. Oprea, *et al.*, ff. Tr. 1505, at 41-47; *see* Findings 108, 110, 127-128.

215. HL&P also took steps to enhance the QA abilities of senior officials who retained responsibility for STP and to reinforce management attitude toward quality. In 1980, Messrs. Jordan and Oprea (as well as a number of B&R officials and employees) attended a seminar on the elements of a good QA program sponsored by Mr. Philip Crosby, a noted QA consultant. Jordan, ff. Tr. 1223, at 10; Tr. 1279-80 (Jordan); Broom/Vurpillat, ff. Tr. 3646, at 47; Tr. 1706-13 (Amaral).

216. Mr. Amaral stated that HL&P had diligently pursued staff upgrading and was aggressively pursuing personnel. He indicated that there were some pretty tough spots to fill and that HL&P was doing a "herculean" job at trying to catch up. Tr. 1761-64 (Amaral). According to him, the most recent Bechtel audit indicated that HL&P did have ade-

quate and competent enough staff to provide programmatic direction at the STP. Tr. 2229 (Amaral).

217. Mr. Frazar testified that there is a corporate commitment to have an annual audit of the QA/QC program by some organization with qualifications similar to Bechtel. Mr. Oprea suggested that at some future time it might be preferable for the auditing organization to be one other than Bechtel to obtain a different viewpoint. Mr. Amaral agreed. Tr. 1942-44 (Frazar, Oprea, Amaral).

218. Mr. Jordan stated that the changes that were made in response to 79-19 were enhancing the role and visibility of HL&P and producing an organization of growing strength. He testified that he was spending more time on the STP and indicated his intention to continue his increased role in oversight of the project through consultation with company executives, participation in appropriate meetings, and close contacts with contractor executives. Jordan, ff. Tr. 1224, at 10-11.

219. Mr. Oprea testified that the corrective actions taken by HL&P will help prevent recurrences of the violations because those actions were directed towards correcting their root causes (Tr. 5365-68 (Oprea)). In addition, the more stringent position of HL&P management and pressure exerted by it in requiring effective actions in construction and inspection activities creates a positive attitude throughout the entire organization (Tr. 5375-76 (Oprea)).

220. Members of the Staff's Crossman panel reviewed the actions of HL&P on each noncompliance described in I&E Report 79-19 and the ultimate disposition of those items. They testified that all items were addressed by HL&P through appropriate changes in procedures, personnel, inspections, tests and formulation of commitments to NRC concerning future actions in management and implementation of the project. Disposition of each noncompliance is discussed more thoroughly in I&E reports cited by the panel, which describe NRC reviews and inspections of corrective actions subsequent to I&E Report 79-19, as well as the bases for decisions by NRC inspectors who concluded ultimately that the noncompliances should be considered closed. In each instance, it was determined that no threat to safety existed as a result of the noncompliance, that NRC regulations had been met through the corrective actions, and that reasonable bases existed for concluding that future performance of HL&P and its contractors could be expected to be consistent with all appropriate requirements. Crossman, *et al.*, ff. Tr. 10,010, at 7-35.

221. The Crossman panel also reviewed actions of HL&P in response to the ten directives listed in the Order to Show Cause. For each of those items, the Staff testimony summarized the nature of the problem

to be addressed, the steps taken by HL&P in response to the NRC directive, and results of the NRC inspections leading to closing of the item. More detailed information on the NRC review of HL&P actions and the bases for decisions to close the items are reported in specific I&E reports cited in the testimony. Crossman, *et al.*, ff. Tr. 10,010, at 35-50.

222. The broad scope of Investigation 79-19 had left thirty unresolved issues and seven open allegations requiring subsequent investigation. The panel reported that all of those matters had been addressed satisfactorily and closed. Crossman, *et al.*, ff. Tr. 10,010, at 51 and Appendix B.

223. The evidence presented before the Board is convincing that the remedial actions taken by HL&P for deficiencies in the STP program were prompt, appropriate and (to the extent carried out as of the closing of the Phase I record) for the most part effective. Testimony of the witnesses and results of inspections and other staff observations indicate that there was no effort by HL&P to avoid responsibility for the problems and no unwillingness to initiate and carry out such actions as were necessary to correct them. To the contrary, HL&P management has energetically taken steps to correct unsatisfactory and undesirable situations and has exhibited an active desire to ensure quality construction and conformance with NRC requirements at the STP. On the other hand, the Staff's SALP evaluation for the period July 1, 1980-June 30, 1981, during which some of B&R's revised procedures were in effect, indicated that B&R was continuing to experience difficulty in effectively correcting deficiencies in a timely fashion (Staff Exh. 133 (I&E Rept. 81-37, at 5-6)) and, in that respect, was only "minimally satisfactory" (*id.* at 3). Further, during the evaluation period, there were continuing allegations of B&R harassment and intimidation of its employees (*id.* at 9). This report raises questions as to the adequacy of B&R's revised procedures.

224. The improvement in managerial competence of HL&P is most forcefully represented by two actions taken by HL&P shortly after Mr. Goldberg joined the company in October 1980. First, based on his extensive prior experience, Mr. Goldberg soon realized that there were serious engineering problems and, in January 1981, HL&P's executive management commissioned the Quadrex Corporation to study B&R's performance in that area (letter, Jack R. Newman to Licensing Board, dated

September 28, 1981; Tr. 2404-06, 10,460 (Goldberg)).[49] Second, he perceived other problems in B&R's performance and, as early as January 1981, recommended that HL&P study alternatives for either upgrading B&R's performance or carrying on the project with other contractors (Tr. 10,518, 10,520 (Goldberg)).

225. To some extent, HL&P's remedial actions included improved procedures under which B&R would carry out various activities, and improved procedures under which it would supervise B&R's QA/QC activities. Although the quality goals underlying the new procedures are to remain in effect, B&R's implementing procedures will not be used by Bechtel and Ebasco. Bechtel and Ebasco procedures for implementing their QA programs, not spelled out in this record, will be substituted. Tr. 10,458-59 (Goldberg); Staff Exh. 131 (I&E Rept. 81-36, at 6-7); Tr. 10,090-93 (Phillips, Crossman, Hubacek); *cf.* Finding 254, *infra.* Given the questions raised by the SALP report (Finding 223), together with the lack of any description of Bechtel and Ebasco implementing procedures (or any evidence as to their effectiveness), we find it necessary that the record be supplemented in Phase II in order to complete the record as to reasonable assurance of the effectiveness of current procedures.

226. The Board finds that, subject to the supplementation of the record in Phase II, HL&P has taken remedial steps which appear sufficient to provide reasonable assurance that it has the managerial competence and character to operate STP safely.

### *Issue C: Character and Competence to Operate the STP*

227. Issue C states:

> In light of (1) HL&P's planned organization for operation of the STP; and (2) the alleged deficiencies in HL&P's management of construction of the STP (including its past actions or lack of action, revised programs for monitoring the activities of its architect-engineer-constructor and those matters set out in Issues A and B), is there reasonable assurance that HL&P will have the competence and commitment to safely operate the STP?

228. HL&P presented its plans for the operation of the STP through a panel consisting of Jerome H. Goldberg, Vice President, Nuclear Engineering and Construction, HL&P, and Jerrold G. Dewease, Vice

[49] The Quadrex Report itself states that the Quadrex review was initiated by HL&P in January 1981 (Quadrex Report (May, 1981), Vol. I, at 1-1). Although the Quadrex Report is not currently a part of the record of this proceeding, we take official notice of the foregoing date. 10 C.F.R. § 2.743(i).

President, Nuclear Plant Operations, HL&P (Goldberg/Dewease, ff. Tr. 10,548). This panel also sponsored into evidence various sections of Chapter 13 of the STP FSAR, as amended through Amendment 25, addressing HL&P's plans for operation of the STP (App. Exh. 56; Tr. 10,553 (Goldberg)).

229. The Applicants' witnesses both indicated that organizational changes were ongoing and that future changes were anticipated (Tr. 10,553-54 (Dewease, Goldberg)). Nonetheless, although operation of the STP is at least 4 years away, HL&P has made considerable progress both in defining the organizational structure that will ultimately be used to manage STP's operation and in filling key operating positions.

230. Mr. Dewease will oversee the nuclear plant operations staff. He will report directly to the Exeoutive Vice President, as will the Manager of QA for Operations and the Director, Nuclear Fuels. Goldberg/Dewease, ff. Tr. 10,548, at 4. Based upon his past job assignments and testimony before this Board, Mr. Dewease appears to have appropriate qualifications to occupy the position of Vice President, Nuclear Plant Operations. He has approximately 23 years of professional experience, including 14 years of nuclear experience with the Tennessee Valley Authority in such positions as instrument engineer, assistant engineering supervisor, quality assurance supervisor and plant superintendent. In his most recent position prior to joining HL&P he was Assistant Director of Nuclear Operations for TVA in which he had responsibilities involving the plant operations staffs of four nuclear plants. He also was responsible for the TVA training center. Goldberg/Dewease, ff. Tr. 10,548, at 2-3.

231. The organization for plant operations is divided into four functional areas: operating, technical, maintenance and training. In addition, two other organizations, the radiation protection group and an administrative group, support the plant. Goldberg/Dewease, ff. Tr. 10,548, at 5-6; App. Exh. 56, FSAR, §§ 13.1, 13.4 and Fig. 13.1-2.

232. The operating section includes licensed operators and auxiliary operators to operate the reactors. It is estimated that this section will eventually consist of seventy-eight persons under the direction of the Operating General Supervisor. The Operating General Supervisor will hold a senior reactor operator (SRO) license for each unit. Six shift supervisor positions are planned for the operating section. Shift supervisors will hold an SRO license for each unit and their functional duties will be established prior to fuel load, emphasizing primary responsibility for safe operation of the plant. Goldberg/Dewease, ff. Tr. 10,548, at 6-7. Shift supervisors, when serving as the senior person on site, will have full authority to order plant shutdown in an emergency (Tr. 10,555

(Dewease)). Unit supervisors, who will also be licensed SROs, will report to shift supervisors and will be responsible for reactor operations command in the control room. Goldberg/Dewease, ff. Tr. 10,548, at 7.

233. As of March 1, 1982, HL&P had one Shift Supervisor, three Unit Supervisors and seventeen support personnel in the operating section. The Shift Supervisor and one of the Unit Supervisors who had been hired were previously licensed SROs on operating commercial nuclear power plants. At that time, the reactor operations personnel retained by HL&P were involved in writing system descriptions and/or operating procedures. Moreover, as systems are turned over to HL&P, these employees were to participate in preoperational testing. Goldberg/Dewease, ff. Tr. 10,548, at 7-8.

234. The technical section is under the direction of the Technical General Supervisor and is made up of four subgroups: reactor engineering, chemical operations, chemical analysis and results engineering (*id.* at 8). The reactor engineering group will consist of a lead reactor engineer and one reactor engineer for each unit. These positions have already been filled, two by persons with extensive nuclear experience and the other by an engineer with appropriate nuclear training. *Id.* at 8-9; Tr. 10,560-62 (Dewease). The reactor engineers are developing the core physics and thermal hydraulic testing programs to monitor core performance. In addition, they are developing the initial start-up test program, the onsite special nuclear materials accountability program and the new fuel receipt, inspection and storage procedures. Goldberg/Dewease, ff. Tr. 10,548, at 9.

235. The chemical operations group is to consist of forty-two persons, including a supervisor, six foremen, fifteen chemical operators and twenty operator trainees and auxiliary operators. As of March 1, 1982, HL&P had hired one chemical operations foreman, three chemical operators and four chemical operator trainees. The chemical operations group will be responsible for the operation of chemical process systems, demineralizer systems, radioactive waste processing systems and nonradioactive waste processing systems. Persons within this group have been utilized to write procedures and develop training materials. *Id.* at 10.

236. The chemical analysis group will ultimately consist of twenty-three people, including a supervisor, two lead technicians, a nuclear plant chemist and nineteen chemical technicians and monitors. As of March 1, 1982, this group consisted of a supervisor, lead technician and six chemical technicians. The chemical analysis group is responsible for plant chemistry and radiochemistry. Personnel within the chemical analysis group have been occupied in writing procedures, developing training

materials, conducting the preoperational environmental sampling program and providing chemical analysis support for hydrostatic tests. *Id.* at 10-11.

237. The results engineering group will consist of a lead results engineer and approximately eleven results engineers. As of March 1, 1982, HL&P had retained the lead results engineer and six of the results engineers. The lead engineer and one of those engineers have had nuclear experience. Two results engineers have completed the 30-week Westinghouse Reactor Operator Training course. *Id.* at 11-12. The results engineers prepare test procedures, perform tests and prepare test reports for initial start-up, maintenance and performance testing of plant systems. Results engineers will also develop solutions to problems and analyze equipment malfunctions in various plant systems. This group has been engaged in developing the programs to implement the various testing activities its personnel will be performing during start-up and eventual plant operation. *Id.* at 12.

238. The maintenance group is divided into four subgroups: electrical, mechanical, instruments and controls, and maintenance support. HL&P has made substantial progress in staffing these various subgroups, and the personnel hired have been performing preventive and corrective maintenance on the reservoir makeup pumping facility and meteorological tower equipment. The maintenance personnel will provide support for various start-up and operations functions. *Id.* at 12-14.

239. The training section is responsible for plant staff training activities and consists of three subgroups: operating training, simulator training and general training. The simulator training group will utilize a plant-specific simulator that, as of the close of the Phase I record, was on order and scheduled to be installed by mid-1983. A substantial number of the instructor positions within the training organization have been filled and those personnel are going to various technical schools and preparing course work. *Id.* at 15-16.

240. The radiation protection group will consist of thirty-three individuals, including one supervisor, two health physicists and thirty radiation protection technicians, monitors and trainees. As of March 1, 1982, HL&P had retained one supervisor and one health physicist. The supervisor has 30 years' experience in applied radiation protection in both the Navy and commercial nuclear power plants. *Id.* at 16.

241. Finally, an administrative group consisting of fifteen to twenty employees is envisioned to provide clerical and administrative support to the plant operations staff (*id.* at 17).

242. With respect to technical support from outside the operations group, HL&P is developing its own capability to perform non-LOCA transient analysis (*id.* at 17). In January 1980, Nuclear Services Corporation completed a study for HL&P to determine (in light of the experience at Three Mile Island) the requisite technical staff HL&P would require to provide in-house technical support during plant operation (*id.*). In this regard, HL&P's goal is to have an onsite staff technically capable of performing the design or design verification for all technical areas, especially those that are uniquely nuclear (*id.* at 18). Mr. Goldberg's engineering and construction organization will also provide technical support, as needed (*id.* at 4). In aid of that goal, HL&P has assigned twenty-six people to Bechtel in order to gain practical experience in the design activity associated with the STP so that HL&P may better maintain the plant after it is completed and is operating. For specialized areas, HL&P anticipates it will continue to employ outside consultant assistants. *Id.* at 18; Tr. 10,558 (Goldberg).

243. Taking into account the stage of construction of the STP, HL&P's overall staffing for the plant operation is well under way. Since 1977, when staffing began, over 100 persons out of approximately 450 for two-unit operation have been hired. Those people hired are performing various preoperational activities. In addition, before fuel is loaded at the STP, HL&P will conduct tests of the plant equipment and systems. A separate HL&P organization, designated as the Startup Group, has been established for this purpose. This group is already writing start-up test procedures. As each plant system nears completion, the HL&P Startup Group, along with HL&P Plant QA, Bechtel QA and Bechtel Engineering, will review the status of the system to determine what must be accomplished before the system will be ready for testing and operation. Goldberg/Dewease, ff. Tr. 10,548, at 20-22.

244. HL&P's plan for its shift organization is similarly well developed. A Shift Supervisor with an SRO license will be on site any time a unit is loaded with fuel. *Id.* at 29. Whenever he or she is the senior person on site, this supervisor will have total authority to shut down the plant (Tr. 10,555 (Dewease)). All personnel on shift are responsible to this individual. Reporting directly to him will be an organization for each reactor unit headed by a Unit Supervisor who has an SRO license and a Chemical Operations Foreman with associated staff. Each unit also will have two operators with RO licenses, a Radiation Protection Technician/Monitor and a Chemical Technician/Monitor. HL&P does not currently contemplate a Shift Technical Advisor (STA) in its shift organization but, rather, plans to provide for the expertise envisioned for an STA through increased training of its Shift Supervisors. If,

however, in the future the NRC requires that a specific Shift Technical Advisor position be established, HL&P has committed to creating such a slot, possibly using an additional licensed operator (probably an SRO) for that purpose. Goldberg/Dewease, ff. Tr. 10,548, at 29-31; Tr. 10,557, 10,565 (Dewease).

245. Procedures are being drafted to limit access to the control room and to govern the turnover in personnel between shifts (Goldberg/Dewease, ff. Tr. 10,548, at 31). A Plant Operations Review Committee (PORC) has been established in accordance with technical specifications to advise the plant superintendent on matters important to safety. Among the activities conducted by the PORC are review of procedures, tests, changes to technical specifications and safety-related systems, technical-specification violations, 24-hour notification items, plant operations and the security and emergency plans. PORC procedures are designed to minimize the possibility of suppression of dissenting opinions regarding safety matters. *Id.* at 33-34. Moreover, there is a corporate level committee known as the Nuclear Safety Review Board (NSRB), with the function of reviewing matters such as proposed changes to procedures, equipment, systems, technical specifications and the operating licenses. The NSRB will further routinely audit various aspects of plant operations. *Id.* at 5, 34-35. Although HL&P has not considered whether a public representative should be included on this Board, and has received no requests for such representation, it indicated that the company has been disposed to allow public participation in other sensitive areas and, at the time of plant operation, would consider that question (Tr. 10,564 (Goldberg)).

246. HL&P's plans for the operation of the STP were addressed by the Staff through a panel consisting of Lawrence P. Crocker and Glen L. Madsen. Crocker/Madsen, ff. Tr. 10,721.[50] Mr. Crocker is the Section Leader, Management Technology Section, Licensee Qualifications Branch of the Division of Human Factors Safety, Office of Nuclear Reactor Regulation (NRR), NRC. He participated in the management and plant staffing review for operation of the STP. Mr. Madsen is the Chief of Reactor Project Branch 1, Region IV, NRC. He is responsible for inspection activities at the STP, including those activities relating to the transition program. *Id.* at 1-2, Professional Qualifications.[51] This panel submitted as evidence the Staff's Partial Safety Evaluation Report

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[50] The testimony of the Crocker/Madsen panel, together with earlier prepared testimony of Mr. Crocker and Mr. Frederick R. Allenspach (NRR) and the Staff's partial SER, were not presented orally but, by agreement of all parties, were stipulated into the record (Tr. 10,718-21).

[51] Mr. Allenspach (*see* note 50, *supra*) is within Mr. Crocker's section and assisted him during the management review.

(PSER) (NUREG-0780, dated April 1981) relating to the adequacy of HL&P's plans for the operation of STP (*id.*, Sections 13 and 17). The evaluation of management was made against the guidelines of NUREG-0731, and HL&P's management was found to be properly organized and prepared for eventual plant operations (*id.* at 13-1). The PSER was issued in response to CLI-80-32 (as well as to the earlier suggestion of this Board, set forth in our Memorandum dated March 10, 1980) and constitutes the Staff's evaluation as of April 1981. Future amendments will be included in the final SER. PSER at 1-1.

247. As a result of its review, the NRC Staff concluded that the Applicants' planned management and operating organizations meet the requirements of current NRC rules and regulations, are in conformance with NRC guidelines, and are acceptable to the NRC Staff. PSER at 1-1.

248. Management of facility construction at STP has been more complex, from an organizational standpoint, than management of plant operation is likely to be. This is so because, unlike construction, almost all operating personnel will be under HL&P's direct control rather than that of contractors. Further, the work force for plant operation is considerably smaller than for plant construction. Tr. 9896-99 (Hayes); Tr. 9903-04 (Shewmaker); Tr. 9906-08 (Phillips).

249. Taking the degree of organizational complexity for operation into account, and based on the testimony of the Goldberg/Dewease and Crocker/Madsen panels, together with cross-examination by the Staff and Board examination of the Goldberg/Dewease panel,[52] we conclude that there is now reasonable assurance that HL&P will have the competence and commitment to operate the STP safely. Because of the preliminary nature of the testimony on this issue, this finding is based on our expectation that the testimony will be updated prior to issuance of any Decision authorizing facility operation.

### *Issue D: Adequacy of Current Construction QA Programs*

250. Issue D, as admitted in the Second Prehearing Conference Order, *supra*, states:

> In light of HL&P's prior performance in the construction of the STP as reflected, in part, in the Notice of Violation and Order to Show Cause dated April 30, 1980, and HL&P's responses thereto (filings of May 23, 1980, and July 28, 1980), and actions

[52] Because of the preliminary status of the information presented, CCANP elected not to cross-examine the Goldberg/Dewease panel (Tr. 10,554).

taken pursuant thereto, do the current HL&P and Brown & Root (B&R) construction QA/QC organizations and practices meet the requirements of 10 C.F.R. Part 50, Appendix B; and is there reasonable assurance that they will be implemented so that construction of STP can be completed in conformance with the construction permits and other applicable requirements?

251. Subsequent to the admission of Issue D, and as a result of Brown and Root's replacement by Bechtel and Ebasco as engineering manager and construction contractor, respectively, the Board advised the parties that it would consider the adequacy of the construction QA/QC program as modified to reflect the new organizational developments. Fourth Prehearing Conference Order, *supra,* at 3-4. Issue D has therefore been modified by replacing the reference to "Brown and Root (B&R)" with "Bechtel/Ebasco."

252. The current QA/QC organization and program were presented by HL&P through a panel made up of James E. Geiger, Donald T. Krisha and Clyde L. Hawn (Geiger, *et al.,* ff. Tr. 10,580). At the time this testimony was offered, Mr. Geiger was the HL&P Project Quality Assurance Manager for the STP. His extensive QA/QC experience includes the position as Project QA Manager for San Onofre Units 1, 2 and 3. *Id.* at 1-3. As of July 1982, Mr. Geiger became the Corporate QA Manager for STP and was replaced as Project Quality Assurance Manager by Mr. Al Walker. Mr. Walker has 9 years' QA/QC experience in nuclear plant construction. Tr. 10,583 (Geiger). Mr. Krisha is the QA Manager for the Houston area office of Bechtel and is currently assigned as the STP Project QA Manager (Geiger, *et al.,* ff. Tr. 10,580, at 1). Mr. Krisha's QA/QC experience includes serving as Bechtel QA Manager/Domestic Projects, where he was responsible for managing the Bechtel QA activities at the Palo Verde, Vogtle and Rancho Seco Nuclear Generating Stations (*id.* at 4). Mr. Hawn is the Quality Program Site Manager for Ebasco Services, Inc. (Ebasco) at the STP (*id.* at 1). Mr. Hawn's QA experience includes holding such positions as Senior QC Supervisor, QA Supervisor, Quality Program Site Manager and QA Manager at WPPSS Nuclear Project Nos. 3 and 5, Laguna Verde, Waterford Unit 3 and the Tokamak Fusion Test Reactor prior to his assignment to the STP (*id.* at 6). This panel also sponsored into evidence App. Exhs. 55 and 55A, which provide a description of the quality assurance program currently being implemented at the STP (Tr. 10,582 (Geiger)).

253. The Staff presented John W. Gilray to testify on the adequacy of HL&P's current QA/QC organization and program for the balance of design and construction (Gilray, ff. Tr. 10,689). Mr. Gilray is the principal quality assurance engineer within the Quality Assurance Branch (QAB) of the Office of Nuclear Reactor Regulation (NRR), Division of

Engineering. Since the Show-Cause Order of April 30, 1980, Mr. Gilray has been the QAB reviewer responsible for the evaluation of changes in HL&P's docketed QA/QC program for design and construction to determine its acceptability. *Id.* at 1. Specifically, Mr. Gilray reviewed HL&P's most recent submittal to the Staff on March 9, 1982, being Revision 3 to its docketed QA program for the remaining design and construction activities at the STP (*id.* at 4-5). In addition, the Staff presented the testimony of Lawrence P. Crocker and Glen L. Madsen concerning the qualifications of Bechtel and Ebasco (Crocker/Madsen, ff. Tr. 10,721). *See* Finding 246, *supra*, for an identification of this panel.

254. HL&P's Revision 3 to its QA program describes three programs: the previously updated and Staff-approved QA program for HL&P, and the QA programs of the two recently assigned principal contractors, Bechtel and Ebasco. The revised HL&P portion of the QA program provides for an improved QA organization with increased authority and responsibilities for surveillance by HL&P personnel during the day-to-day design and construction activities. Gilray, ff. Tr. 10,689, at 5. Bechtel has committed to apply its Staff-approved quality assurance topical report BQ-TOP-1, Revision 3A, as modified in Part B of Revision 3 of HL&P's latest QA program for Bechtel's engineering, procurement, and construction management activities at the STP. Similarly, Ebasco has committed to apply its Staff-approved quality assurance topical report ETR-1001, Revision 10a, as modified in Part C of Revision 3 of HL&P's latest QA program for the quality assurance and quality control of Ebasco's construction services at the STP. *Id.* These topical reports are Bechtel's and Ebasco's descriptions of generic QA/QC programs that satisfy Appendix B criteria. These programs were then modified to conform to the plant-specific needs of STP. Geiger, *et al.*, ff. Tr. 10,580, at 9-11. The program commitments made by HL&P as a result of the Show-Cause Order have been carried into the current QA/QC organization and program, with modification necessary to accommodate the replacement of B&R by Bechtel/Ebasco (*id.* at 13-15).

255. Bechtel's organization for performing its QA function at the STP is under the direction of Bechtel's Los Angeles Power Division. Reporting to the Los Angeles Power Division Manager of QA is a QA Manager for the Houston area office. This manager provides technical and administrative direction to the STP Project QA Manager, who, with the assistance of higher levels of QA management, is responsible for assuring the satisfactory implementation of the Bechtel project quality program at the STP. The Bechtel STP organization consists of three sections reporting to the Project QA Manager: design QA, construction QA, and site QC associated with Bechtel's job site activities. The first

two of these sections are supervised by a project quality assurance engineer (PQAE) and the last section by a project quality control engineer (PQCE). *Id.* at 15-17.

256. The design PQAE is responsible for assuring the implementation of the quality program within the design office through review, surveillance, and audits of engineering and procurement activities. The construction PQAE is responsible for assuring that Ebasco and other contractors' construction activities comply with approved quality program and engineering requirements by surveillance of in-process and completed work, review of documentation, and audits for quality program compliance. *Id.* at 16-17.

257. This QA surveillance over construction is pursuant to Bechtel's construction manager role and represents an additional layer of QA review not present when B&R had both construction and construction manager roles (Tr. 10,619 (Geiger)). Moreover, HL&P will monitor Bechtel's surveillance over Ebasco (Tr. 10,622 (Geiger)). This is all in addition to Ebasco's primary obligation, as constructor, to have a QA/QC program that complies with 10 C.F.R. Part 50, Appendix B. Bechtel's site PQCE is responsible for performing QC inspections associated with Bechtel's job site activities; specifically, receipt, storage and maintenance of permanent plant items. The site PQCE is also responsible for verifying the effectiveness of the contractor's QC program by surveillance and redundant inspections of selected work activities which had previously been accepted by the contractor's QC personnel. The Project QA Manager, the PQAEs and the PQCE all have "stop work" authority over quality-related activities at STP. Geiger, *et al.,* ff. Tr. 10,580, at 17.

258. Bechtel QA is responsible for review and approval of Ebasco's QA/QC procedures and instructions (*id.* at 18). HL&P in turn will monitor Bechtel's approval of Ebasco's implementing procedures (Tr. 10,622 (Geiger)). Bechtel will also audit and monitor the activities and documentation of organizations and individuals involved in the implementation of the constructor's QA/QC program (Geiger, *et al.,* ff. Tr. 10,580, at 18). Bechtel management will be informed of QA/QC activities through audit reports, monthly trend reports, management staff meetings and an annual review meeting that covers the status of the QA/QC programs of the various Bechtel divisions and projects (*id.* at 18-19).

259. Bechtel's QA program is functionally divided into engineering, procurement and construction. Project engineering is responsible for all Bechtel engineering design work performed by and for the project and for checking and reviewing functions performed on the project. Procurement specifications for material and equipment are prepared by engineer-

ing and reviewed by QA for adequacy of specified QA program and documentation requirements. Procurement supplier quality (PSQ) performs a surveillance and inspection function over supplier activities and reviews completed supplier quality verification documents at the supplier's facility. The inspection of items received is performed by Bechtel's QC group at the construction site. QA monitors this process and performs audits and surveillances to assure effective implementation and has the authority to stop supplier work and shipments until required corrective action has been taken and verified. *Id.* at 19-21.

260. The Bechtel construction management organization is responsible for the overall construction program for the STP, including such functions as planning, scheduling, monitoring, and evaluating the Ebasco and contractor construction and QA/QC activities. Each contractor, including Ebasco, is held responsible for performing construction work within the scope of its contract in accordance with approved procedures and a quality program. Contractors are responsible for audits and surveillances of their respective work and QC activities. Bechtel QA is responsible for conducting audits, surveillances, and selected redundant inspections of the Ebasco contractor work and QA/QC activities. *Id.* at 21.

261. Ebasco's STP QA/QC organization consists of three basic groups: QA, QC, and Quality Records. Each of these groups is headed by a site supervisor who reports to the Quality Program Site Manager. The QA group is responsible for performing planned and scheduled audits of Ebasco activities, including the performance of trend analyses of nonconformance reports, and deficiency reports to identify any trends adverse to quality. The QC group is responsible for performing inspections and witnessing or performing examinations and tests of all Ebasco nuclear-safety-related construction activities. The Quality Records group is responsible for assembling documentation packages, verifying the completeness and accuracy of the records, providing adequate safeguards and retrievability of records while under Ebasco control, and for transmitting completed records to HL&P. *Id.* at 22-23.

262. HL&P will conduct a series of reviews of engineering, procurement, construction management, and construction activities to assure proper implementation of its contractors' QA programs. Initially, HL&P has reviewed and approved all aspects of the docketed QA/QC program. HL&P will also conduct a series of audits, surveillances and selective inspections to assure that the procedures of Bechtel, Ebasco, and other constructors not only accurately reflect regulatory requirements but are in fact being implemented. *Id.* at 24-25. In a selective inspection HL&P takes a plant component which has been previously in-

spected and approved by its contractor and performs a reinspection (Tr. 10,620 (Geiger)). In contrast, a surveillance of contractor's activity would involve the situation in which HL&P performs a QC function of ongoing work (Tr. 10,620-21 (Geiger)). HL&P will remain closely involved in the project through daily activities of its QA personnel, weekly meetings with Bechtel and Ebasco QA personnel and receipt of monthly trend reports. An annual independent assessment of the STP QA program will be conducted throughout the life of the project by an organization not involved in the project. Geiger, *et al.*, ff. Tr. 10,580, at 25-26.

263. The Staff review of Bechtel's staffing of key positions within its QA/QC organization indicated that persons with appropriate experience are being assigned (Crocker, *et al.*, ff. Tr. 10,721, at 7). The project QA Manager has 8 years of nuclear QA experience; the design office PQAE has 16 years of QA/QC experience; the construction PQAE has 17 years of nuclear QA/QC experience; and the site PQCE has 15 years of QA/QC experience (Geiger, *et al.*, ff. Tr. 10,580, at 30-31). Similarly, key persons within the Ebasco QA/QC program have had appropriate experience; the site QA supervisor has 11 years' experience in design, construction and QA of power plants and the site QC supervisor has 12 years' experience (*id.* at 32-33).

264. Accordingly, based upon the programs outlined above, it appears that HL&P, Bechtel, and Ebasco QA/QC organizations have the requisite independence from cost and scheduling in order to perform their functions (Tr. 10,632 (Geiger, Krisha)). All organizations report to upper level management off site (Geiger, *et al.*, ff. Tr. 10,580, at 13, 15, 22, and Figures 1, 2, and 3 attached thereto). The Staff performed a detailed review and evaluation of the HL&P program, including Bechtel's and Ebasco's QA programs, and concluded that these programs described the necessary requirements, procedures and controls that, when properly implemented, will comply with the requirements of Appendix B to 10 C.F.R. Part 50. Gilray, ff. Tr. 10,689, at 5-6. The Staff concluded that based upon past experience and association with Ebasco and Bechtel, both corporations are well-qualified in the activities they have been assigned at the STP. The Staff further found that based upon preliminary reviews both organizations are selecting individuals with considerable qualifications and experience to manage their responsibilities at the STP. Crocker/Madsen, ff. Tr. 10,721, at 7.

265. Cross-examination by the Intervenors of the Geiger panel and of Mr. Gilray did not elicit any evidence counter to the direct testimony of these witnesses. The cross-examination of Mr. Gilray served to elucidate some of the changes and improvements to the QA/QC program

which have occurred since issuance of the Show-Cause Order and assumption of responsibilities by Bechtel and Ebasco (Tr. 10,690-10,702 (Gilray)). In particular, we note that proficiency tests for QA/QC inspectors are not to be limited to oral or written tests but rather will also include physical demonstrations that the inspector can perform the inspections to which he will be assigned — a situation which did not always obtain under the B&R QC activities (Gilray, ff. Tr. 10,689, at 3; Tr. 10,697-98 (Gilray); *see also* Tr. 10,711 (Gilray)).

266. In response to Board questions, Mr. Gilray indicated that, based on his past association with Bechtel and Ebasco, the construction QA/QC program would likely be properly implemented. He stated that in most cases QA/QC program implementation by these organizations had been satisfactory. He acknowledged that, in certain cases, Bechtel and Ebasco have had problems, but he opined that they "have learned by their mistakes." Tr. 10,705 (Gilray). He added that, although some problems would likely arise, they would be no worse than what would be expected on a normal large construction project. With respect to Bechtel, he commented that the problems which had arisen were not generic but were specific to particular plants. Tr. 10,717 (Gilray).

267. The three-layer QA/QC program provides an additional layer of quality review to that present under the HL&P–B&R program. Management officials from HL&P and Ebasco expressed the view that the new arrangement on balance provides advantages and does not detract from the ability of the construction contractor to perform a quality job. Tr. 10,508-09 (Goldberg, Crnich). Mr. Gilray expressed a similar view for the Staff (Tr. 10,707-08, 10,713-14).

268. We conclude that the current HL&P, Bechtel, and Ebasco QA/QC organizations and practices meet the requirements of 10 C.F.R. Part 50, Appendix B. We also conclude that there is reasonable assurance that the QA program for the STP will be implemented so that construction of the STP will be completed in conformance with the construction permits and other applicable NRC requirements. Of necessity, however, this latter conclusion is based in large part on the experience of Bechtel and Ebasco at other sites, together with the background and experience of personnel assigned by Bechtel and Ebasco to the STP. For, at the time the Phase I record was closed, Bechtel and Ebasco had not yet undertaken significant safety-related activities at the STP. As pointed out in our opinion on Issue B, however, we are requiring the Staff (and other parties, including the Applicants, if they wish) to supplement the record in this regard by reporting to us during the Phase II hearings concerning the implementation of the QA/QC program for construction. *See* p. 697, *supra*.

269. Although Issue D by its terms is limited to the adequacy of the QA/QC program for continued construction, the replacement of B&R by Bechtel and Ebasco caused us additionally to hear testimony on Bechtel's organizational framework for continued construction, including consideration of plans for design, a review of past problems, project construction and HL&P management involvement. Fourth Prehearing Conference Order, *supra,* at 4. The Applicant addressed these questions through a panel of Jerome H. Goldberg, HL&P's Vice President for Nuclear Engineering and Construction; Burton L. Lex, Bechtel's STP Project Manager; and John Crnich, Construction Manager for Ebasco at STP (Goldberg, *et al.,* ff. Tr. 10,403). The Staff presented the Crocker/Madsen panel (*see* Finding 246, *supra*) (Crocker/Madsen, ff. Tr. 10,721).

270. Bechtel and Ebasco have each had extensive experience in nuclear power plant construction activities. Bechtel is one of the world's largest engineering firms engaged in nuclear power plant design, construction and start-up activities. During the past 8 years it has been involved in the design of fifty nuclear power units (total capacity 51,000 MW) and in the construction of forty-one units (total capacity of 43,000 MW). Ebasco over the last 20 years has served as constructor or construction manager on seventeen nuclear units, as architect-engineer for five nuclear units (at which construction was performed by others), and has constructed one nuclear unit according to another architect-engineer's design. It also has specific experience in the take-over of construction management activities at a non-nuclear facility where work had been started by others. Crocker/Madsen, ff. Tr. 10,721, at 5-7; Goldberg, *et al.,* ff. Tr. 10,403, at 7-9; App. Exhs. 52 and 53.

271. All professional personnel assigned by Bechtel to its transition team (and identified on this record) have had appropriate previous nuclear experience, and assignments to the team appear to have been made to provide for continuity from the transition phase through to project completion. Ebasco also appears to be staffing its organization with persons having considerable nuclear experience. Crocker/Madsen, ff. Tr. 10,721, at 6-7. Moreover, HL&P has taken an active role in both the transition program and in planning for the completion of the design and construction of the STP. It is providing overall project direction to Bechtel's project manager. It has taken special care to assure that the root causes identified in its response to I&E Report 79-19 and the Order to Show Cause are considered by both Bechtel and Ebasco in their transition program and plans for the completion of STP. Goldberg, *et al.,* ff. Tr. 10,403, at 36-44. Significantly, HL&P has expended considerable effort in upgrading the skills and experience of its own managerial and

technical personnel. HL&P replaced many key personnel with individuals who had greater experience and/or training, all in an effort to improve construction performance. Tr. 10,480-84 (Goldberg); *see also* Findings 209-216, *supra.*

272. The foregoing organizational plans and activities and the personnel identified thus far provide reasonable assurance that HL&P, Bechtel and Ebasco have organized themselves in such a manner that the balance of design and construction can be completed in conformity with the construction permits, the Atomic Energy Act, as amended, and the Rules and Regulations of the Commission.

*Issue E: Adequacy of Existing Structures*

273. Issue E states:

> Is there reasonable assurance that the structures now in place at the STP (referred to in Sections V.A(2) and (3) of the Order to Show Cause) are in conformity with the construction permits and the provisions of Commission regulations? If not, has HL&P taken steps to assure that such structures are repaired or replaced as necessary to meet such requirements?

274. The structures referred to in Sections V.A(2) and (3) of the Show-Cause Order are the Units 1 and 2 Reactor Containment Buildings (RCB), Unit 1 Fuel Handling Building and Unit 2 Mechanical-Electrical Auxiliary Building (MEAB) (Staff Exh. 46, Show-Cause Order, at 6). As set forth in the Show-Cause Order, deficiencies with respect to those structures included improper construction practices during the placement of concrete, concrete voids, improper cadwelding practices, improper placement of Category I backfill, a dimensional error in one of the buildings (*see* Findings 318-326 on Intervenor Contention 1.1, *infra*), and inadequate welding controls (*id.* at 1-11). As a result of these deficiencies, the Show-Cause Order directed that a review be made of existing structures to determine whether work in the three areas of soil, concrete, and welding had been properly performed and, if repairs were required, to describe the extent of the repairs and a schedule for completion of work (*id.* at 14-15). This process has now been substantially completed.

275. In HL&P's July 28, 1980, response to the Show-Cause Order (*see* Finding 85), the status of the work in the three areas was reported, verification of the work performed to date was set forth, and a repair program, where appropriate, was outlined (Staff Exh. 48, at 14). We turn to each of the three areas.

*(1) Adequacy of Category I Structural Backfill*

276. The Show-Cause Order directed HL&P to perform five tasks relative to the structural backfill at the STP. HL&P was directed to review information or obtain data to: (1) verify the test fill program that established the soil conditions, lift thickness, compactive effort, and equipment characteristics necessary to develop the requisite in-place densities; (2) perform a comparison of backfill material tested and described in FSAR Section 2.5.4.8.3 (addressing liquefaction) with the backfill used in the field; (3) determine the sequence of construction followed for existing backfill, including the loose-lift thickness and number of passes of the equipment to obtain the required density; (4) determine the adequacy of the density of the existing backfill material, including that under structures founded on backfill; and (5) explain the rationale behind the construction procedure of using 18-inch loose-lifts compacted by eight passes of the equipment to achieve the required densities. Staff Exh. 46, Show-Cause Order, at 14.

277. In addition to the Show-Cause concerns, the Staff reported six items of noncompliance with respect to the STP structural backfill program in Inspection Report 79-19. Specifically, those items of noncompliance found that: (1) Pittsburgh Testing Laboratories' (PTL) procedures did not provide instructions for depth of in-place density testing (Staff Exh. 46, Appendix A, Item of Noncompliance 4); (2) B&R construction procedures failed to set forth an identified and documented basis for the acceptability of the required minimum of eight roller passes (*id.*, Item of Noncompliance 2); (3) PTL did not record the actual number of roller passes or the actual lift thicknesses in the earthwork inspection reports (EIRs) (*id.*, Item of Noncompliance 5); (4) the PTL relative density test apparatus was broken for a period between November 1979 and January 1980, and backfill placement proceeded although the required laboratory test could not be performed (*id.*, Item of Noncompliance 3); (5) Woodward-Clyde Consultants (WCC) used a nonconforming hammer for standard penetration tests of the backfill from January 28, 1980 to February 4, 1980 (*id.*, Item of Noncompliance 16); and (6) WCC used a nonconforming split spoon for its standard penetration testing (*id.*, Item of Noncompliance 17). *See also* Pettersson, *et al.*, ff. Tr. 5796, at 23-24.

278. In January 1980, to respond to initial concerns raised by the Staff inspection team still conducting Inspection 79-19, HL&P and B&R initiated a soil test boring program to assess and verify the adequacy of the in-place Category I structural backfill at the STP. This program was conducted by geotechnical engineers from WCC. Pettersson, *et al.*, ff. Tr. 5796, at 26. The program, completed in April 1980, verified the

overall adequacy of the Category I structural backfill, but recommended further confirmatory investigations in four specific areas to assure engineering adequacy of the backfill. *Id.* and Staff Exh. 48, at 2-2.

279. At the time the Staff Show-Cause Order was issued in April 1980, data obtained during the WCC test boring program were already under analysis. Upon issuance of the Show-Cause Order, HL&P established a special Task Force to respond to the Show-Cause Order, comprised of geotechnical and QA engineers from both HL&P and B&R. The Task Force was to perform a study to verify the acceptability of previously placed backfill, the testing methods used in determining the adequacy of that backfill and the adequacy of the in-place Category I structural backfill. Pettersson, *et al.*, ff. Tr. 5796, at 27. WCC, which was in the process of completing its verification analysis, was assigned by the Task Force to investigate, analyze and conduct further verification studies. *Id.* and Staff Exh. 48, at 2-2 and 2-3. In addition HL&P deemed it desirable that an independent assessment of the Category I structural backfill analysis be performed. Accordingly, in May 1980, the firm of Shannon and Wilson, Inc., was retained as consultant to B&R to establish an independent review committee of geotechnical experts to review the Category I structural backfill construction for the STP and to review the work of the Task Force. Pettersson, *et al.*, ff. Tr. 5796, at 27; Staff Exh. 48, at 2-4 and 2-5.

280. The Applicants presented panels of witnesses from the Task Force and from the Expert Review Committee, respectively, in response to the concerns relative to the backfill expressed in the Show-Cause Order and Board Issue E. The first panel consisted of C. Bernt Pettersson, Assistant Discipline Project Engineer for B&R at the STP; Timothy K. Logan, Project QA Supervisor for HL&P's W.A. Parish Generating Unit and HL&P's QA representative on the STP Soils Task Force; Charles S. Hedges, Project Manager for WCC's work at the STP; and W. Stephen McKay, Corporate Manager for Quality Assurance at PTL. This panel addressed the development of the structural backfill program at the STP and the Task Force's effort in response to the Show-Cause Order. Pettersson, *et al.*, ff. Tr. 5796.

281. The second panel consisted of Stanley D. Wilson, a private consulting engineer and founding partner of Shannon and Wilson, Inc., and Thomas E. Kirkland, senior principal engineer and engineering group leader in Shannon and Wilson's Seattle office. This second panel described the Expert Committee's evaluation of the Task Force's work and its findings on the adequacy of the Category I structural backfill at STP. Wilson/Kirkland, ff. Tr. 2697, at 5. This panel further sponsored

into evidence the Expert Committee's final report concerning the Show-Cause Item #2 structural backfill investigation. App. Exh. 6.

282. The Staff addressed the resolution of the structural backfill issue through its Crossman panel (*see* Finding 19, *supra*) (Crossman, *et al.*, ff. Tr. 10,010).

283. The Applicants' Task Force panel first explained how backfill was placed at the STP. Backfill was placed, compacted, and accepted in individual layers or lifts. The backfill placed at one time in a specific area is called a placement and several placements of backfill are generally required to complete one lift over an entire building foundation area. All placements were compacted before an overlying placement was made. Pettersson, *et al.*, ff. Tr. 5796, at 7.

284. Although no specific code or standard governs placement and the compactive effort of Category I structural backfill for the safety-related structures at the STP, compacted properties of the backfill must be consistent with the structural design criteria for foundations and embedded walls of all Category I structures. *Id.*; Regulatory Guide 1.70. To satisfy this general requirement, specifications were developed in 1974 jointly by B&R and WCC to decide upon material properties of the backfill. Material from the Eagle Lake area (Colorado River Alluvium), approximately 55 miles from the STP site, was determined to be the best source area for the fill material. Pettersson, *et al.*, ff. Tr. 5796, at 8. Upon re-evaluation of this choice in light of the Show-Cause Order, it was again determined that the fill material had all the desired characteristics of an ideal structural backfill (Tr. 2807 (Wilson)).

285. Based on the 1974 laboratory testing of this material, WCC initially recommended that an 80 percent relative density requirement for backfill at STP would provide an ample factor of safety against liquefaction. Pettersson, *et al.*, ff. Tr. 5796, at 8. B&R, with the approval of HL&P, adopted a specification requirement for the STP providing for a minimum relative density of 80 percent and an average relative density of 84 percent (*id.* at 9; Tr. 2736 (Kirkland); Tr. 6091-92 (Hedges)). Construction procedures were developed in an effort to implement these end-process goals in 1976. It was determined that a 10-ton steel drum vibratory roller should be used to compact lifts with a maximum loose-lift thickness of 18 inches. It was further decided that after eight or twelve passes (depending on whether the lift were an underlying or surface lift), it would be appropriate to begin in-place density testing to evaluate the adequacy of compaction. Although not set forth in the construction procedures, the Applicants' witnesses asserted that it was understood by construction that the density tests were end-process tests and that the compaction effort would be continued beyond the minimum

number of passes until proper density was achieved. Pettersson, *et al.*, ff. Tr. 5796, at 11 and 12; Tr. 5949-51 (Pettersson, Hedges); Tr. 5952 (Logan); Tr. 6104-05 (Logan, Hedges).

286. With respect to monitoring this process, PTL inspectors were to provide continuous inspection of the placement of all material (Pettersson, *et al.*, ff. Tr. 5796, at 13). In this context, continuous inspection was interpreted to mean observing the placement process sufficiently to assure that the minimum construction procedures were met and that the final acceptance density was achieved (Tr. 2815 (Wilson)). For example, in the inspectors' earthwork inspection reports (EIRs), a checklist indicated not the actual loose-lift thickness but only that the lift was 18 inches or less. Similarly, inspectors did not check the actual number of roller passes performed to achieve the requisite density but rather only that the minimum number of passes required had occurred. Pettersson, *et al.*, ff. Tr. 5796, at 14. The requirement for a minimum number of passes, which stems from construction procedures, assures a minimum uniformity throughout the entire structural backfill (Tr. 6105, 6118 (Hedges)). If the requisite density is achieved, the number of passes required to achieve that density, beyond the minimum required to achieve uniformity, becomes technically irrelevant (Tr. 6104, 6135 (Logan)).

287. To determine the density of each lift after compaction, PTL inspectors generally performed at least one field density test for every 20,000 square feet of unrestricted backfill. For every fourth field density test, at least one laboratory maximum-minimum test and one gradation test was performed. Pettersson, *et al.*, ff. Tr. 5796, at 10. It was then recorded on the EIR and Density Test Reports whether the required relative density had been achieved (*id.* at 15). In addition, backfill material qualification, placement, inspection, and testing were monitored by HL&P QA personnel (*id.* at 17).

288. All the questions raised in the Show-Cause Order relative to backfill have been adequately answered. Specifically, HL&P found no material difference between the soil properties tested in 1974 and the soil properties found during the 79-19 Inspection. Pettersson, *et al.*, ff. Tr. 5796, at 29; Crossman, *et al.*, ff. Tr. 10,010, at 38; Staff Exh. 120. Construction procedures for Category I structural backfill were developed based upon specification requirements and existing industry practices. Pettersson, *et al.*, ff. Tr. 5796, at 29; Crossman, *et al.*, ff. Tr. 10,010, at 36; *see also* Staff Exh. 40. The original test fill program showed that approximately 80 percent relative density could be obtained by four passes over loose-lifts of between 18 to 24 inches. However, the Expert Committee report found that sixteen to twenty passes or more

are presently needed to consistently meet the desired densities. It further stated that this number of passes is consistent with the number actually performed in the field before the requisite density was met. App. Exh. 6, at 30. Nonetheless, B&R site geotechnical engineers originally recommended that provisions for a minimum of twelve roller passes be initially incorporated into construction procedures. B&R subsequently concluded that the minimum of twelve passes would actually only be necessary on the surface lift. Crossman, *et al.*, ff. Tr. 10,010, at 36. This was so because underlying lifts would receive further densification upon compaction of overlying lifts. Pettersson, *et al.*, ff. Tr. 5796, at 12. Although not set forth in the procedures, HL&P and B&R indicated that it was generally understood the twelve passes represented an appropriate place to begin end-process testing. Pettersson, *et al.*, ff. Tr. 5796, at 11-12.

289. The Staff reviewed the procedures used to perform the test fill program, and the technical reference document entitled, "Test Program for Compaction of Category I Structural Backfill," and the results of the Expert Committee's report (Crossman, *et al.*, ff. Tr. 10,010, at 36; Staff Exhs. 40, 58, and 94). Based upon the Expert Committee's report, the Staff concluded that the Category I structural backfill is adequate at STP (Crossman, *et al.*, ff. Tr. 10,010, at 39; Staff Exh. 94). The Staff concluded that the density of lower lifts is significantly increased by compaction of subsequent lifts and that this multiplying effect demonstrated that a minimum of eight passes of compaction equipment was adequate to begin in-process testing. As a practical matter, it was pointed out by the Staff that if the requisite density was not achieved using the minimum number of passes, additional passes with compaction equipment were made until the required density was achieved prior to continuing the construction effort. Crossman, *et al.*, ff. Tr. 10,010, at 37. The Staff reviewed the findings of both the Task Force and Expert Committee and based upon those findings determined that Item 2 of the Show-Cause Order was satisfied. *See* Crossman, *et al.*, ff. Tr. 10,010, Corrections and Update, at 3; Staff Exh. 94.

290. During cross-examination of the Pettersson panel, CCANP introduced six documents into evidence, all 10 C.F.R. § 50.55(e) reports from HL&P to the Staff concerning backfill. CCANP Exhs. 24, 25, 26, 27, 28 and 30. The first five of these exhibits were interim and final reports concerning the discovery by WCC in January-February 1980, through its boring program, of four areas within the Unit 2 area where backfill densities were below the 80 percent minimum relative density specification referred to in Findings 285-288, *supra*. The final report concludes that the deviations were slight, that the backfill was ad-

equate for the purposes intended, and that no remedial action was required. The Expert Committee concluded that

> The studies established that these zones have adequate factors of safety against liquefaction and that negligible pore pressures which might build up in these zones during the Safe Shutdown Earthquake are not significant with respect to the adequacy and safety of the overall structural backfill.

CCANP Exh. 28. Finally, the Applicants' witnesses testified that the four small areas were insignificant and would not impair the overall adequacy of the existing backfill (Tr. 2824-26, Kirkland, S. Wilson).

291. The other backfill exhibit introduced by CCANP concerned a report of differential soil settlement resulting in a tilting of the Unit 2 Mechanical Electrical Auxiliary Building (MEAB) and a curvature in the basemat under that building. The Applicants explained their corrective actions for this situation and expressed the view that the tilt did not affect the integrity of the MEAB or the piping systems, which had not yet been installed. CCANP Exh. 30; Tr. 6026-30 (Pettersson).

292. CCANP did not submit proposed findings with respect to its Exhibits 24-28 or 30. We see no reason to disagree either with the explanations of the Applicants with respect to the four areas which failed to meet the relative density specification and the differential settlement under the MEAB, or with their evaluation of the safety significance of these questions.

*(2) The Concrete Verification Program*

293. The Show-Cause Order directed HL&P to review safety-related concrete structures, including embedments such as supports and the fuel transfer tube. If, after this review, repairs were required, HL&P was to describe the extent of the repairs necessary and to provide a schedule for completion of that work. Staff Exh. 46, Show-Cause Order, at 15. In addition, among the twenty-two items of noncompliance in Inspection Report 79-19 were citations for failure to implement corrective action relative to concrete placement activities and unqualified Civil QC inspectors (Staff Exh. 46, Appendix A, Items of Noncompliance 7 and 8).

294. At the time the Order to Show Cause was issued, HL&P was already in the midst of an extensive concrete verification program stemming from voids discovered in Lifts 8 and 15 in the RCB. *See* findings on Intervenor Contention 1.2, *infra.* Upon issuance of the Show-Cause Order, HL&P and B&R initiated a Task Force to perform an assessment of safety-related concrete structures at STP. It was determined that embedments such as supports and the fuel transfer tube involve issues of

traceability and the application of Section III of the ASME Code, and that accordingly those items would be addressed by the Welding Task Force in response to Item (3)(a) of the Show-Cause Order. Staff Exh. 48, at 3b-1. *See* Findings 327-337, *infra,* and Staff Exh. 88. The Task Force included over twenty full-time engineers from HL&P and B&R. This team received further assistance from outside consultants due to the same concerns that led to the Expert Committees in the backfill verification program. Staff Exh. 48, at 3b-2.

295. The Applicants presented a panel from the Task Force to testify on the efforts of the Concrete Verification Program. The panel consisted of Gerald R. Murphy, B&R's Assistant Discipline Project Engineer (Civil-Structural Discipline) for the STP; Ralph R. Hernandez, Supervising Engineer for the Civil Nuclear Support Section within the Civil Mechanical Engineering Division of HL&P's Power Plant Engineering Department; and Joseph F. Artuso, President of Construction Engineering Consultants, Inc., an engineering firm providing consulting services, quality control services and materials analysis for construction projects. *See* Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 1-5. Mr. Murphy was the task force leader in response to the concrete verification request in the Show-Cause Order and Mr. Artuso was a member of the consultant panel. The Task Force was charged with determining whether the safety-related concrete work at STP, as of the time of the Show-Cause Order, had been properly performed, and with describing the extent of repairs, if any, that needed to be made in order to correct any deficiencies. *Id.* at 10-11.

296. The Task Force pursued this objective by identifying and examining samples of the safety-related concrete in several structures at STP selected by a conservative, statistically valid method. Its review covered 68 percent of all safety-related concrete placed to April 30, 1980. Omitted were such structures as the containment building shells, the essential cooling water (ECW) intake and discharge structure, and the electrical raceway system. (The containment shells and ECW structure had been subject to other additional reviews.) *Id.* at 11-12, 15; Tr. 6411-12 (Murphy). Once the placements were selected for review, a four-phase verification program was followed, consisting of: (1) a review of all documentation relating to each placement; (2) a comparison of the "as-built" configuration for each placement (as determined by a field survey) with the "as-designed" configuration reflected in the documentation; (3) a visual inspection of each placement to assess the general quality, and to determine potential structural defects as well as to identify areas requiring follow-up testing; and (4) random selection of three sample areas within each selected placement to perform a variety

of specialized tests to investigate the structural properties of the placement. Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 12.

297. The placements were classified into five major generic types and were selected on the basis of accessibility for inspection and testing, and on the amount of information that testing would disclose with respect to the placement. Placements were selected from those determined to be more critical because the complexity of the placement was related to previously identified concerns. *Id.* at 14; Tr. 6364-65 (Murphy, Artuso).

298. After it was determined the documentation was substantially complete (Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 17), the "as-built" configuration was checked against the "as-designed" condition. In the vast majority of cases (over 90 percent) the specified tolerance was met. *Id.* at 19; Tr. 6348 (Murphy). The deviations from tolerance that were identified were minor and in no instance resulted in the rejection of an item because it was out of tolerance to the point that "fit-up" could not be accomplished. The Applicants' witnesses justified the minor deviations from tolerance that occurred by stating that the design tolerances at STP were too restrictive. Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 20; Tr. 6403-06 (Murphy, Hernandez).

299. Next a visual inspection was conducted by the consultant panel. The visual inspection addressed any prior items of noncompliance as well as the known characteristics and accompanying potential problems on each placement. The visual inspection indicated quality workmanship and satisfactory construction. In addition, selected destructive testing was performed. The break samples indicated well-consolidated concrete. In addition, selected cores were compression tested and all met the design requirements. All concrete subjected to a petrographic examination was found to be homogeneous and hard with little or no segregation. Selected Windsor Probe testing indicated that all concrete tested was in excess of design requirements. Ultrasonic testing indicated that the concrete, in addition to having a high strength, had excellent uniformity. Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 21-26.

300. Based on the above verification program, the consulting panel concluded that there was reasonable assurance that the quality of safety-related concrete at STP is adequate and that the concrete structures will perform as designed (*id.* at 27). Accordingly, the panel concluded that based on its review, test, and inspections there is reasonable assurance that the safety-related concrete structures at STP, as constructed or repaired, are substantially in conformance with the construction specifications, and that in the few instances where deviations exist they are insignificant from the point of view of plant safety. This assurance

was reached after examining structures representative of 97 percent of all safety-related concrete at STP. *Id.* at 29-30; Tr. 6412 (Murphy).

301. The Staff concurred with the finding that there are no internal honeycomb or void areas which remain unrepaired in the structures (Staff Exh. 113, at 5). This concurrence was based upon the Applicant's four-phase investigation program, Windsor Probe readings, ultrasonic testing, and petrographic and compressive strength evaluations of drilled core samples. The Staff reviewed all phases of this program prior to its concurrence. *See* Staff Exhs. 82, 85, and 113.

*(3) The Welding Verification Program*

302. The third aspect of construction work addressed in the Show-Cause Order is the adequacy of the welding performed at STP. This subject was addressed by a panel of Applicants' witnesses consisting of Eugene A. Saltarelli, B&R Senior Vice President and Chief Engineer; Matthew D. Muscente, B&R Welding Program Manager for STP; Gordon R. Purdy, B&R QE Manager; Logan D. Wilson, HL&P's Mechanical/NDE Project QA Supervisor; J. Rodolfo Molleda, an HL&P Supervising Engineer; Michael D. Sullivan, a consultant on welding and metallurgy employed by NUTECH; and Dr. Daniel Hauser, a Senior Research Scientist at Battelle Memorial Institute, Columbus, Ohio (Battelle). Saltarelli, *et al.*, ff. Tr. 7536.

303. The Show-Cause Order directed HL&P to review safety-related welding, including civil, structural and piping. If, after this review, repairs were required, HL&P was to describe the extent of the repairs necessary and to provide a schedule for completion of that work. Staff Exh. 46, Show-Cause Order at 15. In addition, seven items of noncompliance were cited in Inspection Report 79-19 relative to the STP welding program. Specifically: (1) the B&R weld filler material specification did not contain the latest documentation change notice (Staff Exh. 46, Appendix A, Item of Noncompliance 9); (2) the STP construction procedures failed to incorporate requirements for welding protection against adverse environmental conditions (*id.*, Item of Noncompliance 10); (3) the quality of numerous radiographs was such that proper interpretation was not possible (*id.*, Item of Noncompliance 11a); (4) the linear indications contained in several radiographs were not recorded on interpretation sheets (*id.*, Item of Noncompliance 11b); (5) the evaluation of certain liquid penetrant indications was not in compliance with the ASME Code (*id.*, Item of Noncompliance 11c); (6) outdated procedures for liquid penetrant examinations were being used (*id.*, Item of

Noncompliance 12); and (7) radiograph evaluation of some welder qualification tests did not comply with the ASME Code in that the penetrameter (radiographic image quality indicator) was placed on the side of the test pipe close to the radiographic film (film side) rather than the preferred radiation source (source side) (*id.*, Item of Noncompliance B).

304. Upon issuance of the Show-Cause Order, HL&P and B&R formed a special Task Force review team to formulate a program to reassess and verify safety-related welding at STP and to determine whether the safety-related welding that was completed as of the date of the order was properly performed. The Task Force was also given the responsibility of identifying any repair work that might be required and to establish a schedule for completion of such work. Staff Exh. 48, at 3a-1. In addition, as was the case in both soils and concrete, early in the review process the Task Force established an Independent Review Committee both to review and to approve the Task Force programs and reports. The Independent Review Committee further was to assure that the Task Force was properly implementing the programs, provide technical and code advice, and to advise the Task Force in making recommendations for corrective action and additional review. *Id.* at 3a-2 and 3a-3.

305. The Task Force defined the scope of its review to encompass examination of randomly selected safety-related ASME piping welds and AWS structural welds made by B&R from the start of construction until the time safety-related welding was stopped on April 11, 1980. All STP welding procedures, specifications and a significant portion of documentation were also examined. The Task Force members developed a plan to evaluate four specific areas of the STP welding program: (1) the safety-related AWS welding program; (2) the ASME welding program including welder qualifications; (3) the nondestructive examination (NDE) program; and (4) code commitments as identified in the engineering specifications and implementing procedures. Saltarelli, *et al.*, ff. Tr. 7536, at 27.

306. With respect to the first of these four commitments, the Task Force visually examined a random sample of seventy-nine safety-related AWS welds selected from all areas of the plant in accordance with accepted sampling procedures. This examination revealed sixty-one welds with nonconformances. *Id.* at 29. The Task Force therefore recommended that all accessible safety-related structural AWS welds be reexamined and that all such welds not in compliance with the AWS Code be repaired and that the adequacy of all inaccessible AWS welds be determined based on the types of nonconformances found in the reexamination of the accessible welds. In addition, it was recommended that all

AWS welders and inspectors be retrained to the requirements of the AWS Code and applicable STP procedures. *Id.* at 30.

307. As a result of the Task Force conclusions with respect to weld deficiencies (both AWS and ASME), B&R and HL&P decided in September 1980, that all accessible safety-related AWS and ASME welds be reexamined and, where required, repaired. This reexamination and repair program encompassed radiography of 100 percent of the accessible ASME welds in the ECW system, requiring that those ECW welds buried under backfill be unearthed. This program was conducted pursuant to a detailed reexamination and repair plan submitted to the Staff on September 10, 1980. *Id.* at 44. In October 1980, the Staff authorized the reexamination and repair of AWS welds as well as limited restart of new AWS welding, based on new management systems and procedures, personnel retraining, the completion of commitments regarding safety-related welding in response to the Show-Cause Order and the completion of all corrective action for previously identified noncompliances related to AWS and ASME welding (*id.* at 45).

308. With respect to the second of the Task Force activities, all radiographs of completed and accepted ASME welds were reviewed by certified NDE Level III examiners in radiography. Twenty-five percent of the radiographed welds that previously had been accepted were considered unacceptable. In addition, the Task Force repeated Code-required visual examination and liquid penetrant testing on a random sample of ASME welds that originally were accepted on the basis of similar examinations. *Id.* at 31. Based upon this reevaluation, the Task Force recommended and HL&P agreed that: (a) all accessible ASME welds with known deficiencies should be repaired; (2) all other accessible ASME welds should be visually reexamined, liquid-penetrant tested and repaired if necessary; and (3) data from the reexamination should be used in the evaluation of the adequacy of the inaccessible ASME welds. The Task Force found that the STP ASME construction procedures and documentation were substantially in compliance with the applicable Code requirements. *Id.* at 32-33.

309. The evaluation of welder performance test records revealed two problems: (1) film-side penetrameter placement for some of the tests; (2) the use of ASME acceptance criteria for both ASME and AWS welder qualifications. The possible effects of the first problem were determined to be insufficient to require further investigation. *Id.* at 33-34. With respect to the second problem, the use of ASME acceptance criteria for AWS welder qualifications was found not to affect previous test results significantly (*id.* at 24).

310. In November 1980, the Staff authorized the reexamination and repair of ASME welds, and limited restart of ASME welding, based on the same factors (outlined in Finding 307, *supra*) which led to similar authorization with respect to AWS welds (*id.* at 45).

311. As of the time the Saltarelli panel testified, approximately half of the accessible AWS welds had been reexamined. Six percent of these welds contained deficiencies directly related to weld strength. All deficiencies found had been repaired, inspected, and accepted. Approximately half of the accessible nonessential cooling water (ECW) ASME welds made prior to the Stop-Work Order had been reexamined and 8 percent contained deficiencies. In addition, 15 percent of the accessible ECW pipe welds had been reexamined and, after finding deficiencies in 83 percent of such welds, these deficiencies were repaired, inspected, and accepted. HL&P committed to radiographing 100 percent of the ECW welds in repairing all deficiencies. *Id.* at 46-47. Finally, the Task Force found AWS construction procedures and weld documentation to be acceptable (*id.* at 30).

312. The Task Force next reviewed the NDE program. It compared the STP NDE procedures for radiography, magnetic particle, liquid penetrant, and visual testing with applicable Code requirements. All procedures were found to be substantially in compliance with the Code. However, the qualification files for NDE inspectors identified various types of irregularities in the qualifications of twenty-one of the seventy personnel, including uncertified personnel performing NDE, an inspector who signed at a higher level, and the expiration of an eye exam certification. In addition, the review determined that documentation regarding nine of the twenty-one inspectors showed insufficient training and/or experience in performing examinations. The Task Force concluded, however, that program improvements implemented since the Stop-Work Order of April 11, 1980 were sufficient to ensure proper control of the NDE inspector certification processes in the future. *Id.* at 34.

313. Finally, the Task Force reviewed the STP engineering specifications and implementing construction/QA procedures in order to determine whether the applicable codes and standards were adequately identified and whether the same commitments had been made in all documents. Although commitments and requirements were found to have been adequately identified in the procedures, it was recommended that procedures be simplified and clarified due to inconsistencies and ambiguities. *Id.* at 35. The recommendation was followed prior to welding restart (*id.* at 36-37).

314. The Staff continuously monitored the activity of the Task Force. *See* Staff Exhs. 72, 82, 88, 112 and 117. The Staff subsequently concluded that virtually all of the commitments made by HL&P relative to its safety-related welding program were completed and therefore closed out Show-Cause Item 3(a) in December 1981. *See* Staff Exh. 131, at 4.

315. Similarly, HL&P resolved all of the Items of Noncompliance relative to safety-related welding set forth in Inspection Report 79-19. Specifically, to assure that the latest document changes were incorporated into both weld filler material specifications and other controlled documents, HL&P revised and updated all control documents and further added an administrative technician to the site HL&P QA staff to be responsible for document control. Crossman, *et al.*, ff. Tr. 10,010, at 22. HL&P further committed to rewriting work procedures to require protection against contamination from rain, snow, and airborne particles during welding operations (*id.* at 23; Staff Exh. 40). These new welding procedures were reviewed and it was verified that adequate requirements had been implemented for maintaining cleanliness during the welding process (Staff Exh. 40, at 7). HL&P further committed to review all radiographic film to identify discrepancies, to revise radiograph film processing procedures to clarify film processing techniques, to retrain and recertify all NDE personnel, and to revise the requirements for recording film conditions (Crossman, *et al.*, ff. Tr. 10,010, at 23; Staff Exh. 82). With respect to inadequate liquid penetrant examinations, all NDE personnel had been retrained in the requirements of inspection procedures with an emphasis upon the importance of adhering to such requirements. Training was followed by a reexamination of all liquid penetrant personnel. Crossman, *et al.*, ff. Tr. 10,010, at 24-25; Staff Exh. 40, at 8.

*(4) Conclusions with Respect to Issue E*

316. We conclude the following with regard to each subpart of CLI-80-32 Issue E:

(1) Adequacy of Category I Structural Backfill

We conclude that there is reasonable assurance that the backfill now in place is in conformity with the construction permits and applicable regulations.

(2) The Concrete Verification Program

We conclude there is reasonable assurance that the concrete work now in place at the STP is in conformity with the construction permits and applicable NRC regulations or that such work will be repaired or replaced as necessary to meet such requirements.

(3) The Welding Verification Program

We find that HL&P is conducting a thorough reevaluation of the STP welding program. This evaluation has resulted in the discovery of significant defects in existing welds and significant improvements in the welding program to prevent recurrence of those defects. The welding verification and repair program indicates that there is reasonable assurance that the welding work now in place at the STP is either in conformity with the construction permits and applicable NRC regulations or that such welded components or structures will be repaired or replaced as necessary to meet such requirements.

## C. Findings on Intervenor Contentions

317. Contention 1 alleges:

There is no reasonable assurance that the activities authorized by the operating license[s] for the South Texas Nuclear Project can be conducted without endangering the health and safety of the public in that:

1. There has been a surveying error which has resulted in the eastern edge of the Unit 2 Mechanical-Electrical Auxiliary Building being constructed one (1) foot short (in the east-west direction) from its design location. This error violates 10 C.F.R. Part 50, Appendix B, Sections X and XI.
2. There has been a field construction error and as a result, extensive voids exist in the concrete wall enclosing the containment building, in violation of 10 C.F.R. Part 50, Appendix B, Sections IX and X.
3. In violation of Quality Assurance and Quality Control requirements applicable to the South Texas Nuclear Project with regard to document control (10 C.F.R. Part 50, Appendix B, Sections VI and XVII), a field document relating to cadweld inspections has been lost.
4. There are membrane seals in the containment structure which are damaged, indicating a violation of 10 C.F.R. Part 50, Appendix B, Sections X, XV and XVI.
5. There are steel reinforcement bars which are missing from the concrete around the equipment doors in the containment and such bars are missing from the containment structure as well, indicating violations of 10 C.F.R. Part 50, Appendix B, Sections X, XV and XVI.
6. There are cadwelds which have been integrated into parts of the plant structure which are not capable of being verified with regard to compliance with 10 C.F.R. Part 50, Appendix B, in violation of Sections IX and X of Appendix B.
7. Quality Control as per the requirements of 10 C.F.R. Part 50, Appendix B, in particular Sections III and IX, has not been complied with, because:
   a. Efforts by quality control inspectors to verify that design changes were executed in accordance with the purposes of the original design were repeatedly and systematically thwarted.

b. There were personnel other than the original designer approving design changes with no first-hand knowledge of the purpose of the original design.
c. There were design changes approved by personnel unqualified in the type of design where the change was made.
d. There were numerous pour cards that were supposed to record the correct execution of concrete pours which were falsified by numerous persons.
e. There has been and continues to be assaults on the Applicant's quality control inspectors, continual threats of bodily harm to those inspectors, firing of inspectors, and other acts constituting a pattern of behavior designed to intimidate the inspectors. As a result of the intimidations, certain inspections were never done because the inspectors decided to play cards over a period of four months rather than risk their safety on the plant grounds.

8.a. As evidenced by the investigative results in Allegation 1 of I&E Report 81-28, Houston Lighting and Power management failed to assure prompt corrective action by Brown and Root in the area of access engineering in violation of Criterion XVI of 10 C.F.R. Part 50, Appendix B.
b. As evidenced by the investigative results in Allegation 1 of I&E Report 81-28, Houston Lighting and Power management does not have a consistent policy on the issuance of stop work orders in violation of Criterion I of 10 C.F.R. Part 50, Appendix B.
c. As evidenced by the investigative results in Allegation 2 of I&E Report 81-28, Houston Lighting and Power management personnel are not committed to respecting the mandates of NRC regulations, especially Criteria I and II of 10 C.F.R. Part 50, Appendix B.
d. As evidenced by the investigative results in Allegation 4 of I&E Report 81-28, HL&P management failed to effectively implement a quality assurance program in violation of Criterion I of 10 C.F.R. Part 50, Appendix B.

As a result of the foregoing, the Commission cannot make the findings required by 10 C.F.R. § 50.57(a)(1) and (2) necessary for issuance of [operating licenses] for the South Texas Nuclear Project.

### *Contention 1.1*

318. With respect to this contention, the Applicants presented the testimony of Richard W. Peverley, the Assistant Engineering Project Manager-Special Services for B&R (Peverley (Contention 1.1), ff. Tr. 7826). The Staff presented testimony by the Seidle panel, identified in Finding 40 (Seidle, *et al.*, ff. Tr. 9205); and by the Crossman panel, identified in Finding 19 (Crossman, *et al.*, ff. Tr. 10,010).

319. There was a surveying error that resulted in the eastern edge of the Unit 2 Mechanical-Electrical Auxiliary Building (MEAB) basemat being 1 foot short. The error was discovered in September 1978 by B&R field engineers and was reported to NRC as a 10 C.F.R. § 50.55(e) item on October 4, 1978. Peverley (Contention 1.1), ff. Tr. 7826, at 3, 7; Seidle, *et al.*, ff. Tr. 9205, at 35; Crossman, *et al.*, ff. Tr. 10,010, Appendix C, Item 8; Staff Exh. 113 (I&E Rept. 81-16, at 2).

320. The error was apparently caused by the surveyors using the wrong reference line to lay out the MEAB (Peverley (Contention 1.1), ff. Tr. 7826, at 7). There was no formal QA/QC procedure to detect surveying errors (*id.* at 8; Tr. 7967-68 (Peverley)). The Staff attributed the error to the failure of the Field Engineering department to properly check survey calculations (Seidle, *et al.*, ff. Tr. 9205, at 36; Staff Exh. 113 (I&E Rept. 81-16, at 2)). Layout points should have been, but were not, traversed back to the original building location monuments, and this failure constituted a poor surveying practice (Tr. 7891-92 (Peverley)).

321. To correct for the surveying error, the equipment layout in the MEAB was redesigned (Peverley (Contention 1.1), ff. Tr. 7826, at 4-5). The redesign affected only the west one-fourth of the building and eliminated excess floor space (*id.* at 5 (Peverley); Seidle, *et al.*, ff. Tr. 9205, at 36; Staff Exh. 113 (I&E Rept. 81-16, at 2)).

322. The redesign was verified against the applicable provisions of the FSAR and the Security Plan (Peverley (Contention 1.1), ff. Tr. 7826, at 5-7). The Staff reviewed the Applicants' engineering evaluation of the redesign against the safety criteria in the FSAR for the layout of systems and components and concluded that the error was resolved and the 50.55(e) item should be closed (Staff Exh. 113 (I&E Rept. 81-16, at 2)).

323. The redesign does not result in any increased safety hazard (Tr. 7975 (Peverley)). Nor does it create any difficulty with operation, inspection, maintenance, or replacement of the equipment (Tr. 7973 (Peverley)).

324. The lack of a formal QA/QC procedure either to detect surveying errors or to assure that errors would not occur violated 10 C.F.R. Part 50, Appendix B, Criterion X (as well as Criterion II). Since surveying is not a test activity, there was no violation (as alleged) of Criterion XI (Peverley (Contention 1.1), ff. Tr. 7826, at 9).

325. To prevent surveying errors in the future, organizational and procedural changes were implemented (Peverley (Contention 1.1), ff. Tr. 7826, at 9). (They would be applicable, however, only during the period when B&R served as construction contractor.) In particular, all major layouts were to be double-checked, and all building layout points traversed back to the original survey points so that closure occurs. A different surveying crew was to survey back to the original building location monuments to assure that the original survey was correct. Peverley (Contention 1.1), ff. Tr. 7826, at 9 (Peverley); Tr. 7891 (Peverley). These changes resulted in a program which complied with 10 C.F.R. Part 50, Appendix B, Criteria II and X.

326. Based on the foregoing, uncontroverted findings, the Board concludes that the surveying error does not prevent the findings required by 10 C.F.R. § 50.57.

### *Contention 1.2*

327. The Applicants addressed this contention by a panel of witnesses called to testify on concrete-related contentions, comprised of Gerald R. Murphy, Assistant Discipline Project Engineer (Civil Structural Discipline) for STP, B&R; Gerald L. Fisher, B&R Discipline Project Engineer for the STP Civil Structural Group; Charles M. Singleton, B&R's Civil Discipline QC Superintendent for STP; Joseph F. Artuso, President of Construction Engineering Consultants, Inc.; Ralph R. Hernandez, Supervising Engineer, Power Plant Engineering Dept. (Civil Nuclear Support Section), HL&P; and David G. Long, Senior Engineer and former Lead Engineer, QA, HL&P (Murphy, *et al.* (Contentions), ff. Tr. 6522). The Staff addressed this contention through the Seidle panel (*see* Finding 40, *supra*) (Seidle, *et al.*, ff. Tr. 9205).

328. The walls of each reactor containment building (RCB) are constructed in circumferential portions called "lifts." A lift is generally 10 feet high. The concrete forming a lift may be the result of one, or of more than one, pouring. Each such pouring and the portion of the lift it forms is called a placement. On the interior of the RCB walls is a 3/8-inch carbon steel liner that provides a leak-tight membrane for the containment. Construction of the corresponding portion of the liner precedes each placement. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 8-9; Tr. 6536-43 (Murphy, Long, Hernandez).

329. There were voids in the shell walls of the RCBs. Voids were first detected in Lift 15 of the Unit 1 RCB, and then were detected in Lift 8 of that same RCB. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 10-11, 13; Seidle, *et al.*, ff. Tr. 9205, at 36-37; Staff Exh. 113 (I&E Rept. 81-16, at 4); Staff Exh. 118 (I&E Rept. 81-22, at 4). Subsequently, an investigation of Lifts 1-17 of the Unit 1 RCB and Lifts 1-6 of the Unit 2 RCB (80 percent of the shell walls) identified eighty-nine void areas in Unit 1 and sixteen void areas in Unit 2 (Murphy, *et al.* (Contentions), ff. Tr. 6522, at 14, 18).

330. The investigation of the RCB lifts consisted of visual inspection of external surfaces, soundings (the systematic tapping of the containment liner to identify potential voids), mapping of hollow-sounding areas, and drilling (at all points at which soundings identified a potential void and at more than 160 additional test points). Where voids were

discovered, their size and shape were determined by use of a fiberscope. *Id.* at 11-15; Seidle, *et al.*, ff. Tr. 9205, at 37.

331. The investigation demonstrated that voids occurred only in areas of high rebar (reinforcing steel bar) concentration beneath penetrations and beneath the 8-inch channel which served as a plate stiffener (Murphy, *et al.* (Contentions), ff. Tr. 6522, at 12, 14; Staff Exh. 118 (I&E Rept. 81-22, at 4)). The main cause of the voids was the complex structural arrangement in those areas, where the existence of additional rebar and horizontal liner stiffeners or bracket anchorages interfered with concrete flow. In addition, access and visibility limitations, insufficient vibration of the poured concrete, and equipment malfunctions and associated delays were contributing factors. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 12-13. Furthermore, procedures for stopping work when problems were encountered during an ongoing pour were not properly exercised by B&R construction or quality control personnel (Seidle, *et al.*, ff. Tr. 9205, at 36).

332. To particularize with respect to the Lift 15 voids, the particular pour began at 8:00 or 9:00 a.m. in the morning and lasted until around 6:00 a.m. the following morning — a duration of almost 20 hours. The extended duration of the pour was caused in part by repeated failures (as many as five) of pumps. By the end of the pour, the engineering and QC personnel involved had become quite fatigued. Moreover, there apparently was inadequate lighting to perform the pour after dark. The pour in the area around the crane — where voids were found — occurred between 3:00-6:00 a.m. of the second day. The voids were not detected by QC personnel but rather by construction personnel who were cleaning up after the pour. Tr. 7086, 7131, 7133-34, 7151 (Singleton); Tr. 7080-81 (Hernandez).

333. When confronted with repeated pump failures and an extended duration of the pour, QC personnel should have realized the potential for voids and should have advised construction personnel to take steps to prevent them. Two QC inspectors were disciplined for failing to grasp the seriousness of the situation and for reporting an absence of problems with the pour. Tr. 7087, 7129 (Singleton). Greater experience might have assisted the QC inspectors in detecting a situation where voids were likely and in taking steps in anticipation of a work stoppage (Tr. 7153-55 (Artuso); Tr. 7162-63 (Singleton)).

334. Permitting concrete pouring to have taken place under inappropriate circumstance such as attended the Lift 15 pour constituted a violation of 10 C.F.R. Part 50, Appendix B, Criterion IX (as well as Criterion II). The failure of QC personnel to have discovered the Lift 15 voids constituted a violation of Criterion X.

335. All voids were completely filled with grout which has a strength greater than or equal to the surrounding concrete (Murphy, *et al.* (Contentions), ff. Tr. 6522, at 15-17; Tr. 6723-24 (Murphy)). The Staff has reviewed these repairs and found them adequate (Staff Exh. 113 (I&E Rept. 81-16, at 4-5); Staff Exh. 118 (I&E Rept. 81-22, at 4-5)). In addition, prior to operation of STP, the RCB walls will be subject to further tests, including pressurizing the containment in excess of design basis events (Murphy, *et al.* (Contentions), ff. Tr. 6522, at 20; Tr. 6888-89, 7197-98 (Hernandez)).

336. While B&R remained as construction contractor, construction and QA/QC procedures were improved to prevent further voids. Visibility and access were improved by relocating the construction joint so that the 8-inch stiffeners are now near the top of the placement; this relocation makes it easier to vibrate and inspect the concrete during placement. The horizontal rebar was also repositioned in order to improve access to the placement for inspectors and vibrator operators. Furthermore, instead of a normal concrete mix, a fine aggregate concrete (grout) mix was to be used beneath penetrations and in congested areas. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 19. Finally, post-placement meetings were established to identify and resolve any problems experienced during the placement (*id.* at 20).

337. Based on the foregoing findings, the Board finds that the voids in the RCB walls, now fully repaired, do not prevent the findings required by 10 C.F.R. § 50.57.

### *Contention 1.3*

338. The Applicants addressed this contention through the Murphy panel of witnesses identified in Finding 327, *supra* (Murphy, *et al.* (Contentions), ff. Tr. 6522). The Staff addressed this contention through the panels of Seidle, *et al.*, and Crossman, *et al.* (*see* Findings 40 and 19, respectively)) (Seidle, *et al.*, ff. Tr. 9205; Crossman, *et al.*, ff. Tr. 10,010).

339. A cadweld is a connector used to join two pieces of reinforcing steel bar or to connect a piece of reinforcing steel to a structural member. *See* Murphy, *et al.* (Contentions), ff. Tr. 6522, at 24-26.

340. The Intervenors never identified which field document they claim was lost. However, on September 9, 1978, the NRC Staff received allegations that there were cadwelding irregularities at the STP; one of the allegations was that field sketch No. FSQ-030 had been lost (Staff Exh. 13 (I&E Rept. 78-15, at 2, 6-7); Murphy, *et al.* (Contentions), ff. Tr. 6522, at 34-35).

341. FSQ-030, which should have recorded the precise location of Cadwelds 28H31 through 28H44, was in fact never prepared (Staff Exh. 14 (I&E Rept. 78-18, at 2); Murphy, *et al.* (Contentions), ff. Tr. 6522, at 34).

342. The approximate location in the reactor containment building of Cadwelds 28H31 through 28H44 can be determined by a design drawing (Staff Exh. 14 (I&E Rept. 78-18, at 2)). If necessary, the cadwelds could be found, though with some difficulty (Tr. 6807-08 (Murphy); Tr. 6812 (Singleton)).

343. Knowing the precise location of cadwelds is only necessary if test splices indicate that a batch of cadwelds might be defective (Murphy, *et al.* (Contentions), ff. Tr. 6522, at 39; Tr. 6811-12 (Singleton)). There was no evidence, however, of test splice failure for the batch of cadwelds in question. Moreover, the cadweld inspection book showed that Cadwelds 28H31 through 28H44 had been inspected and approved. The Staff considered the matter as resolved. Staff Exh. 14 (I&E Rept. 78-18, at 2); Murphy, *et al.* (Contentions), ff. Tr. 6522, at 35.

344. The failure to prepare field sketch FSQ-030 violated 10 C.F.R. Part 50, Appendix B, Criterion VI. The failure to have a document like FSQ-30 among the project QA records violated Criterion XVII. For reasons previously stated, there was no safety significance to these violations.

345. Based on the foregoing findings, the Board finds that the absence of the field sketch and the resulting difficulty in determining the exact location of Cadwelds 28H31 through 28H44 in the RCB does not prevent the findings required by 10 C.F.R. § 50.57.

### *Contention 1.4*

346. With respect to this contention, the Applicants presented the Murphy panel (*see* Finding 327) (Murphy, *et al.* (Contentions), ff. Tr. 6522). This contention was addressed for the Staff by the Seidle panel (*see* Finding 40) (Seidle, *et al.*, ff. Tr. 9205).

347. A waterproofing membrane is placed around the STP containment buildings to cover all exterior vertical and horizontal surfaces below grade. The membrane is a laminated sheet material consisting of rubberized asphalt bonded to a polyethylene sheet. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 40.

348. During a site inspection, an NRC inspector received allegations that the membrane seals in the Unit 1 RCB had been installed at night,

without proper QC inspection prior to the placement of backfill, and apparently had been damaged. Seidle, *et al.*, ff. Tr. 9205, at 52-53.

349. An onsite investigation of these allegations was conducted, the results of which were detailed in I&E Rept. 79-14 (Staff Exh. 32). The investigation did not substantiate the allegations (Seidle, *et al.*, ff. Tr. 9205, at 53; Staff Exh. 32 (I&E Rept. 79-14, at 3)).

350. There were instances where the membrane seals were damaged during construction. Such damage was identified by the QA/QC program prior to backfilling and documented in nonconformance reports (NCRs). The damage was then repaired and the NCR closed out. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 43.

351. In one instance, review of NCRs indicated that backfill had been placed over a membrane prior to inspection. However, the backfill was subsequently removed, and the membrane inspected. *Id.* at 92.

352. The membrane seal is a secondary means of protecting against groundwater seepage. It is not required by any code or standard applicable to STP. Primary waterproofing is provided by (1) the continuous steel liner and (2) the reinforced concrete containment structure (basemat and walls). *Id.* at 39-40; 63-64.

353. No violation of 10 C.F.R. Part 50, Appendix B, Criteria X, XV and XVII has been demonstrated with respect to the membrane seals.

354. Based on the foregoing, uncontroverted findings, the Board finds that any damage to the membranes surrounding the containment buildings' basemat and walls below grade has been repaired, and therefore, does not prevent the findings required by 10 C.F.R. § 50.57.

### *Contention 1.5*

355. The Applicants addressed this contention through various members of the Murphy panel (*see* Finding 327) (Murphy, *et al.* (Contentions), ff. Tr. 6522). The Staff presented testimony through the Seidle panel (*see* Finding 40) (Seidle, *et al.*, ff. Tr. 9205).

356. In June 1979, the NRC Staff investigated allegations that rebar was missing from parts of the Unit 1 containment structure. The investigation revealed no irregularities. Seidle, *et al.*, ff. Tr. 9205, at 38.

357. A subsequent investigation was conducted by the NRC Staff in January and February 1981, in response to an article in the Houston Post and to CCANP's contention (Staff Exh. 54 (I&E Rept. 80-08, at 3)). This investigation revealed "no documented evidence that reinforcing bars were missing" (*id.* at 10).

358. HL&P also investigated the allegations of "missing rebar," including review of documents concerning the concrete pours identified

by CCANP in response to interrogatories. HL&P found no rebar to have been improperly omitted from the containment. Although rebar was omitted in instances (*e.g.*, where it could not be erected in accordance with design drawings), the omissions were documented through a nonconformance report (NCR) or field request for engineering action (FREA) and were subject to a corresponding design change and engineering review. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 51-52, 68-72.

359. Based on the foregoing, uncontroverted findings, the Board finds that any omitted rebar was documented and subject to the appropriate design change; accordingly, no violation of Appendix B, Criteria X, XV and XVI has been demonstrated with respect to missing rebar in the containment structure, and the omissions do not prevent our findings pursuant to 10 C.F.R. § 50.57.

### *Contention 1.6*

360. The Applicants addressed this contention through the Murphy panel (*see* Finding 327) (Murphy, *et al.* (Contentions), ff. Tr. 6522). The Staff covered this contention through the Seidle and Crossman panels (*see* Findings 40 and 19, respectively) (Seidle, *et al.*, ff. Tr. 9205; Crossman, *et al.*, ff. Tr. 10,010).

361. In May 1978, the NRC Staff investigated allegations of irregularities in cadwelding procedures — in particular, claims that cadweld documents had been falsified. The investigation revealed no evidence of falsification. Staff Exh. 7 (I&E Rept. 78-09); Seidle, *et al.*, ff. Tr. 9205, at 21-23.

362. Subsequent investigations during 1978 and early 1979 revealed problems in cadwelding procedures and quality control. The problems were resolved by the institution of a reinspection program, by retraining cadwelders and inspectors, and by revising quality control procedures. The investigations did not reveal any falsified cadweld records. Staff Exhs. 13, 14, 15, 16, 17, 18 and 19; Seidle, *et al.*, ff. Tr. 9205, at 32-33.

363. In June and October of 1979, reports of two further NRC investigations substantiated discrepancies and omissions in cadweld documentation. The investigative reports indicated, however, that HL&P and B&R had already identified the problem and were actively pursuing its resolution. Staff Exh. 26 (I&E Rept. 79-09, at 3); Staff Exh. 32 (I&E Rept. 79-14, at 4). *See also* Murphy, *et al.* (Contentions), ff. Tr. 6522, at 87-88. Allegations that cadweld inspection reports had been falsified were not substantiated (Staff Exh. 32 (I&E Rept. 79-14, at 4)).

364. HL&P and B&R established a cadweld documentation task force to review all cadweld records. The task force reviewed the records for over 36,000 cadwelds. It determined that 190 cadwelds (about 0.5 percent) lacked in-process and visual inspection records. However, inspection of 150 of the undocumented cadwelds was verified by pour cards for the placements in which those cadwelds were located; therefore, only 40 cadwelds remain undocumented. Murphy, *et al.* (Contentions), ff. Tr. 6522, at 87-88.

365. The visual-inspection rejection rate for cadwelds is approximately 1 percent, and even a cadweld that has been rejected upon visual inspection rarely fails to meet tensile strength requirements. *Id.* at 30, 38. Therefore, the Board concludes that it is unlikely that even one of the forty unverified cadwelds fails to meet tensile strength requirements. *See id.* at 31. Furthermore, the STP design is sufficiently conservative to compensate for instances of cadwelds that were below strength. *Id.* at 61-62.

366. The various document deficiencies, including the absence of documentation for forty cadwelds, even though insignificant from a safety standpoint, constitute at least technical violations of 10 C.F.R. Part 50, Appendix B, Criteria IX and X (and Criterion VI as well).

367. Based on the foregoing, uncontroverted findings, the Board finds that cadweld documentation deficiencies do not prevent the findings required by 10 C.F.R. § 50.57.

*Contention 1.7(a)*

368. The Applicants addressed this contention through Mr. Peverley (identified in Finding 318) (Peverley (Contention 1.7), ff. Tr. 7835). The Staff presented its testimony through the Seidle panel (*see* Finding 40) (Seidle, *et al.*, ff. Tr. 9205).

369. During B&R's tenure at STP, changes to the requirements of a design document, or clarifications of those requirements, were effected through the use of Field Requests for Engineering Action (FREAs) (after the Show-Cause Order, known as Field Change Requests (FCRs)). These requests were transmitted to the Engineering Department, which had the authority to approve or disapprove the request. Any changes pursuant to a FREA or FCR were subject to the same review as was the original design document. In addition, all FREAs and FCRs written against safety-related or seismic Category I design documents required formal design verification. Peverley (Contention 1.7), ff. Tr. 7835, at 5-6; *see also* Crossman, *et al.*, ff. Tr. 10,010, at 47; Tr. 7895, 7898, 7903, 7908 (Peverley).

370. QC inspectors did not have the responsibility for verifying that design changes are executed in accordance with the purpose of the original design. Their responsibility is to provide documented verification that construction is performed in accordance with appropriate procedures and in conformance with the appropriate design documents, as amended. This function was accomplished using pre-planned checklists provided by Quality Engineering. Peverley (Contention 1.7), ff. Tr. 7835, at 4.

371. The Intervenors never specifically pointed to any example of a QC inspector who was thwarted in an effort to verify that design changes were executed in accordance with the purposes of the original design. The record establishes that the premise of this contention — *i.e.,* that QC inspectors had a role in assuring that design changes were properly executed — is without foundation. Therefore, Contention 1.7(a) is without merit. No violation of 10 C.F.R. Part 50, Appendix B, Criteria III and IX has been demonstrated.

*Contentions 1.7(b) and 1.7(c)*

372. The Applicants presented Mr. Peverley (*see* Finding 318) with respect to these contentions (Peverley (Contention 1.7), ff. Tr. 7835). The Staff addressed them through the Seidle panel (*see* Finding 40) (Seidle, *et al.,* ff. Tr. 9205).

373. The Intervenors provided no specific examples of design changes being approved by persons not qualified to do so. The Applicants believed that the concerns raised by Contentions 1.7(b) and 1.7(c) stemmed from the period around May 1978, when, at the direction of HL&P, B&R modified its design change procedures to permit the onsite processing of FREAs, when feasible. To accomplish the onsite processing of FREAs, it was necessary to assign Design Engineers to the site. During the transition to the new system, Mr. Douglas Robertson, a civil engineer already stationed at the site to assist in geotechnical activities, was given the authority to authorize civil/structural FREAs. However, prior to authorizing a civil/structural FREA, Mr. Robertson was required to familiarize himself with the situation and contact by telephone the responsible Design Engineer. If the responsible Design Engineer decided the request was significant and required calculational activities, the FREA would be sent to the Design Engineer for authorization; if the Design Engineer decided that the request was straightforward and did not involve calculational activities, he would authorize onsite approval. Onsite approval, however, did not obviate subsequent review by Design Engineering; all FREAs were forwarded to the

appropriate Design Engineer and were processed without regard to the advance onsite approval of the design change. This process did not violate any rules of the Commission and is consistent with the requirements of Regulatory Guide 1.64. Peverley (Contention 1.7), ff. Tr. 7835, at 6-11; Tr. 7893 (Peverley).

374. An NRC investigation of an allegation concerning the inability of construction engineers to assure that construction was performed in accordance with drawings and procedures did not uncover any evidence that unqualified persons were approving design changes. Seidle, *et al.*, ff. Tr. 9205, at 26.

375. As the Staff observes, although it may be correct to state that Mr. Robertson may have approved design changes with no firsthand knowledge of the purpose of the original design, it cannot be said that the verification against the original design never occurred (Staff FOF at 176). Furthermore, Mr. Robertson was an experienced civil engineer, with previous experience in earthwork construction, surveying, soils and concrete testing, construction project management where piping, steel erection and concrete structures were involved, foundation investigations and design analysis, and airport construction. Peverley (Contention 1.7), ff. Tr. 7835, at 10; Tr. 7901, 7904 (Peverley). For these reasons, to the extent Contentions 1.7(b) and (c) include allegations of safety significance, they are without merit. In addition, no violation of Appendix B, Criteria III and IX, has been demonstrated.

### *Contention 1.7(d)*

376. With respect to Contentions 1.7(d) and 1.7(e), which include several related allegations, the Applicants presented the testimony of Dr. Knox M. Broom, Senior Vice President of the B&R Power Group (*see* Finding 39) (Broom/Vurpillat, *et al.*, ff. Tr. 3646); Charles M. Singleton, Civil Discipline QC Superintendent, B&R (Warnick, *et al.*, ff. Tr. 8032); and Stephen H. Grote, Senior Vice President of Operations, B&R Power Group (at Tr. 4341 *et seq.*). At the Board's request, the Applicants also recalled John B. Duke, a B&R QC Inspector at STP from February 1976 to June 1977, who had been presented by the Applicants on another subject (Buckalew/Duke, ff. Tr. 6265; Duke, recalled at Tr. 6463). The Staff presented the Seidle panel on these contentions (*see* Finding 40) (Seidle, *et al.*, ff. Tr. 9205).

377. Although the NRC Staff on several occasions investigated allegations that construction documents had been falsified (Staff Exhs. 1, 2, 3, 7, 60, and 67; CCANP Exh. 10), the investigations produced no evidence of pour-card falsification, the subject of this contention.

378. In late 1979, the NRC received an allegation that QC inspectors were signing inspection documents (presumably including pour cards) without having conducted the inspection. The QC inspectors allegedly played cards while they were supposed to have been performing inspections. In response to discovery requests, CCANP indicated that the alleged falsifications of pour cards were the result of the alleged card games. An NRC investigation did not substantiate the allegation. Staff. Exh. 32. (*See* Findings 389-391, *infra,* for a more detailed discussion of the alleged card games.)

379. Following the Show-Cause Order and I&E Report 79-19 (Staff Exh. 46, Appendix D), the Applicants established a Special Task Force on Concrete. The Task Force reviewed the accuracy of construction field documents relating to concrete, including pour cards, and found the documentation substantially complete (with minor exceptions) and of good quality. Murphy, *et al.* (Concrete Verification), ff. Tr. 6327, at 10, 16-18.

380. No violation of Appendix B, Criteria III and IX, arising from pour-card falsification has been demonstrated.

*Contention 1.7(e)*

381. During B&R's tenure at STP, there were repeated instances of friction between construction personnel and QC inspectors. The Notice of Violation issued by NRC on April 30, 1980 charged, *inter alia,* that some QC inspectors were harassed, intimidated and threatened. Several instances were cited. Staff Exh. 46, Appendix A, ¶ A.1. In their response to these charges, the Applicants stated that they could not affirm or deny particular statements but that "our own review suggests that such instances probably did occur" (Staff Exh. 47, Attachment, at 1). We here review the considerable testimony offered on this subject.

382. On July 1, 1977, the NRC received a telephone call, during which it was alleged that a B&R construction foreman had assaulted and injured a B&R Civil QC inspector, that the incident was just one of a series of threats and harassments against B&R QC inspectors, and that a B&R construction superintendent had advised his workers that any B&R Civil QC inspector who reported unacceptable items found during concrete placement inspections would be liable for a beating.

The NRC investigated these allegations and found an inordinate amount of friction existing between B&R Civil QC inspectors and B&R construction personnel, and the existence of various minor harassments. The investigation also substantiated two specific incidents: (1) an argument and threats between a B&R Civil QC inspector and a B&R construction foreman on the morning of June 30, 1977; and (2) a physical altercation between a B&R construction foreman and a B&R Civil QC inspector on the afternoon of June 30, 1977. The QC inspector also stated that he was threatened by an unknown laborer as he was leaving the plant site after the incident.

The investigation, however, found no directed program of harassment and intimidation, and no evidence that a B&R construction superintendent had advised his workers that any B&R QC inspector who found and reported unacceptable items during concrete placement would be liable for a beating. None of the ten inspectors interviewed during the investigation stated that any harassment led to the overlooking of unacceptable items. I&E Rept. 77-08 (Staff Exh. 4); *see also* CCANP Exhs. 16 and 20; Staff Exh. 112.

383. HL&P project personnel had discussions with the B&R construction Project Manager and a policy aimed at minimizing friction and altercations was developed (Oprea, *et al.*, ff. Tr. 1505, at 12; *see* Warnick, *et al.*, ff. Tr. 8032, at 31-32). The policy was communicated to all B&R QA/QC personnel by memorandum dated July 27, 1977. It emphasized that confrontations were not tolerated and that disputes were to be referred to supervisors. CCANP Exh. 16; Tr. 3834-35 (Broom). The foreman involved in the physical altercation was subsequently discharged by B&R. Warnick, *et al.*, ff. Tr. 8032, at 13; Tr. 3821 (Broom).

384. On July 19, 1978, the NRC received a telephone call, wherein it was alleged, *inter alia*, that there was undue pressure from B&R construction personnel on B&R QC inspectors. The NRC investigated and concluded that this allegation "could be valid" because of inadequate in-process inspection practices (inadequate pre-pour inspections by craft foreman and field engineers); however, no items of noncompliance or deviations were identified. The investigation also indicated apparent low morale of some B&R Civil QA/QC inspectors. I&E Rept. 78-12 (Staff Exh. 8); Tr. 9269-71 (Seidle). These problems were discussed in a meeting between the NRC and HL&P on August 15, 1978 (I&E Rept. 78-13 (Staff Exh. 9)). *See* Staff Exh. 10 for Applicants' response.

385. On August 22-25, 1978, the NRC investigated the firing of a B&R QC inspector who had allegedly solicited a bribe from a construction person. The NRC also investigated the allegation that other QC inspectors would be intimidated by the firing. The investigation did not substantiate either that a bribe had been solicited or that QC inspectors would be intimidated by the firing. I&E Rept. 78-14 (Staff Exh. 12). After the B&R QC inspector had been discharged, HL&P increased its surveillance program of concrete placements for several months.

Through this surveillance, HL&P concluded that the firing had no adverse impact on the job performance of the remaining QC inspectors. Warnick, *et al.*, ff. Tr. 8032, at 37-38.

386. On August 24, 1978, B&R issued a memorandum to all B&R QA/QC personnel. The memorandum required those personnel to report any abuse to their supervisors, but not to retaliate. CCANP Exh. 54; Warnick, *et al.*, ff. Tr. 8032, at 38.

387. In March 1979, the NRC investigated an altercation between a QC inspector and a construction engineer (I&E Rept. 79-04, at 7 (Staff Exh. 20)). The Applicants testified that a QC inspector and a construction engineer had a dispute over pour cleanliness on a concrete pour, that the QC inspector made remarks which the construction engineer interpreted as calling him a liar, and that the construction engineer then swung at the QC inspector. The QC inspector was reprimanded for his unprofessional conduct and the construction engineer was transferred to another project. The inspector's supervisor, who was present throughout the incident, was given a three-day suspension for allowing the situation to deteriorate to a physical confrontation. Warnick, *et al.*, ff. Tr. 8032, at 13-14, 33-34; Tr. 8070-80 (Warnick).

The NRC investigation determined that the management response to this matter had been timely and effective in indicating support for the site QA program (Staff Exh. 20 (I&E Rept. 79-04, at 7)).

388. During a site inspection in August 1979, the NRC received allegations of intimidation of QC inspectors by B&R construction personnel (I&E Rept. 79-13, at 28-29 (Staff Exh. 27)). In September 1979, the NRC investigated these allegations, specifically that two B&R QC inspectors were intimidated by five B&R construction personnel. The intimidation could be neither substantiated nor refuted, owing to conflicting statements. I&E Rept. 79-14, at 3 (Staff Exh. 32).

389. Among the allegations investigated in the September 1979 inspection were charges, which initially had surfaced in March 1979, by a former B&R QC inspector (the one who had been discharged for allegedly accepting a bribe, *see* Finding 385, *supra*) to the effect that QC inspectors were involved in continuous card games during a several-month period in 1977 and while so involved, signed inspection forms without having performed the inspection. Staff Exh. 32 (I&E Rept. 79-14, at 3); Broom/Vurpillat, ff. Tr. 3646, at 31-33; Seidle, *et al.*, ff. Tr. 9205, at 54-55. NRC investigators interviewed nine QC inspectors, none of whom were aware of card games in 1977. Some of the inspectors admitted that there had been some card games in 1976, but stated that the games were not of the scope alleged and had no adverse impact on QC inspections. The investigation did not, therefore, substantiate the

allegations. Staff Exh. 32 (I&E Rept. 79-14, at 9-10); Seidle, *et al.*, ff. Tr. 9205, at 54-55.

390. B&R interviewed inspectors allegedly involved in the card games. The inspectors stated that there were some card games during lunch and during periods of low construction activity, but that the games did not interfere with any inspections. B&R also reviewed inspection records, found no decrease in deficiency reports during the period in question, and inferred therefrom that the inspections were being performed. Broom/Vurpillat, ff. Tr. 3646, at 32-33.

391. Two inspectors allegedly involved in the card games testified before the Board. One testified that there had been card playing during lunch and during periods of low construction activity from December 1976 to January 1977. The other inspector testified that there had been card games during lunch from the summer of 1976 to the winter of 1976. Both inspectors stated that the games did not interfere with inspections. Warnick, *et al.*, ff. Tr. 8032, at 26-27; Tr. 6461-62 (Duke).

392. On November 2, 1979, the NRC was contacted by a B&R QC inspector who alleged that Civil QC inspectors were being harassed and intimidated by B&R construction personnel and QA/QC management. As a result of these allegations and past allegations of a similar nature (described above), the NRC initiated a special investigation, conducted from November 10, 1979 to February 7, 1980. The investigation substantiated eight of ten allegations of harassment, intimidation, and lack of support for QC inspectors, with several others remaining unresolved as of that time.

The substantiated allegations included, *inter alia:* (1) that the site QA manager told QA/QC inspectors that he would know if they talked to the NRC (insinuating that action or trouble would follow) and that the NRC was tired of hearing complaints; (2) that on two occasions a general foreman (the same foreman on each occasion) threatened a QC inspector (not the same inspector on each occasion); (3) that Civil QC inspectors had lost the support of their supervisors when they were confronted by construction personnel; (4) that a construction superintendent threatened a QC inspector with bodily harm; (5) that a QC supervisor told inspectors that QC inspectors who talked to the NRC would be "hitting the gate" (*i.e.*, discharged); and (6) that QC inspectors are taught not to expect support from their supervisors. I&E Rept. 79-19 (Staff Exh. 46, Appendix D); *see* Findings 64, 66 and 74-75, *supra*.

393. In response to I&E Report 79-19, HL&P increased its involvement in the QA/QC program. Assessments by B&R and HL&P were conducted to identify sources of friction. QC supervisory personnel were upgraded to provide them stature equal to that of their construction

counterparts, QC salaries were revised to attract additional qualified personnel and to reduce attrition, and an extensive recruiting program was instituted.

In addition, written procedures and policies were revised to stress the proper means of dispute resolution and to ensure that any instances of harassment or intimidation were immediately reported. A formal written dispute resolution procedure (STP-PGM-02) was adopted on January 7, 1980 (prior to the formal release of I&E Report 79-19 but subsequent to preliminary notification by NRC to HL&P of some of the findings of that report). QA/QC and construction supervisors received additional training in employee motivation, human relations, and supervisory skills. QA/QC and construction personnel received refresher courses in project procedures. B&R QA management and HL&P site surveillance personnel increased their visits to the site, and access by QA/QC personnel to top level management was facilitated.

Finally, two B&R construction supervisory personnel against whom allegations of intimidation and harassment had been made were removed from the project.

Staff Exh. 47 (Applicants' Response to the Show-Cause Order). *See also* Broom/Vurpillat, ff. Tr. 3646, at 45-50; Warnick, *et al.,* ff. Tr. 8032, at 43.

394. In September 1980, the NRC held a routine announced inspection, including follow-up relative to I&E Report 79-19 items. The NRC inspectors conducted interviews with B&R QC inspectors and addressed the particular I&E Report 79-19 findings of harassment, intimidation, and lack of support. The interviews revealed only a few isolated negative comments, including what was termed an "idle threat" made to a QC inspector. *See* Tr. 8769 (Singleton); Tr. 8930, 8975 (Wilson). The NRC inspectors found the overall interview results to be very positive and considered the previously identified conditions, which had caused the noncompliances, to be corrected. I&E Rept. 80-25 (Staff Exh. 45).

395. At the hearing, the Applicants described further steps which HL&P might take if further instances of harassment, intimidation or threats against QC inspectors were to occur. Specified persons employed by HL&P would be assigned to determine the facts, what immediate corrective action was necessary, and what long-term corrective actions would be necessary to preclude recurrence. HL&P's Vice President, Nuclear Engineering and Construction, stated that he hoped to be able to organize a small team of people that would have a complement of skills (mechanical, civil, electrical) who could become a focal point for HL&P's reviews of such occurrences. Tr. 2537-39, 2740-41 (Goldberg).

396. The investigative results described above indicate excessive friction between QC inspectors and construction personnel, and poor management. This friction continued as late as the spring of 1980. It did not, however, reflect a pattern of intimidation. *See* Tr. 9370 (Seidle); Warnick, *et al.*, ff. Tr. 8032, at 17; Tr. 8129-30 (Warnick); Tr. 8421 (Singleton). Nor, though the question was posed in each investigation, did the investigations reveal QC inspectors failing to perform assigned inspections or overlooking nonconforming conditions as a result of attempted intimidation or harassment (except possibly in one instance). Warnick, *et al.*, ff. Tr. 8032, at 27, 44; *see* Finding 74 with respect to the one instance of possible intimidation).

397. As discussed in greater detail earlier (*e.g.*, Findings 96, 111, *supra*, relative to Issue A), the likely root cause of the widespread and continuing instances of harassment and threats was inexperienced management and an unusually long chain of command from the site to upper management (which resulted in the masking of critical information). Tr. 1739 (Amaral). HL&P has taken steps to alleviate these problems. The Staff judged the current written QA program to be excellent and the grievance procedure better than at most other sites. Tr. 9516, 9548, 10,098 (Phillips). Nonetheless, as of the close of the Phase I record, questions raised by incidents of harassment and intimidation were not entirely resolved (Finding 223, *supra*).

398. The proven incidents of harassment and intimidation do not constitute violations of 10 C.F.R. Part 50, Appendix B, Criterion III (Design Control) or IX (Control of Special Processes). They do constitute violations of the implementation requirements of Criterion II (Quality Assurance Program). Notwithstanding these violations, the Applicants have instituted measures to preclude future incidents of this type. The performance of the Applicants and their contractors in this regard is to be included in the Phase II review which we are directing (*see* p. 697, *supra*).

399. Based on the foregoing findings, the Board finds that the specific allegations of Contentions 1.7(a) through (d) have been rebutted; the allegations of Contention 1.7(e) have been addressed by HL&P and adequately remedied, subject to the review we are directing for Phase II. Therefore, these allegations do not at this time prevent our making findings pursuant to 10 C.F.R. § 50.57.

### *Contentions 1.8(a)-(d) Introduction*

400. Contentions 1.8(a), (b), (c), and (d) each arose out of the Staff inspection described in I&E Report 81-28 (Staff Exh. 124). Each conten-

tion addresses activities which assertedly constitute QA/QC deficiencies and violate various criteria of 10 C.F.R. Part 50, Appendix B. The Applicants addressed all of these contentions through a panel consisting of R.A. Frazar, the STP QA Manager during the pertinent time period; J.L. Blau, the acting Project Engineering Manager at that time; and H.G. Overstreet, supervisor, of the QA group which had responsibility in the area covered by the inspection report (Frazar, *et al.*, ff. Tr. 10,123). The Staff testimony was presented through I&E Report 81-28 (Staff Exh. 124) and by H.S. Phillips, Resident Reactor Inspector for STP and R.K. Herr, Senior Investigator, Region IV (Tr. 10,011).

*Contentions 1.8(a) and 1.8(b)*

401. In November 1980, HL&P QA wrote an NCR. Two of the issues in the NCR were: (1) B&R did not appear to have procedures in place to implement access design, and (2) the PSI/ISI Manual containing the criteria for access design had not been updated every 6 months as required by its terms. In June 1981, a revised NCR was written by HL&P QA; it contained the two issues above and a third issue, that access design was being conducted using a draft Technical Reference Document. Frazar, *et al.*, ff. Tr. 10,123, at 3-5.

402. At the same time as the revised NCR was issued, HL&P QA drafted a stop-work letter. The purpose of the proposed stop-work letter was not to terminate an activity which might cause irreparable construction deficiencies but rather to obtain B&R's immediate attention and action to resolve the issued raised by the NCR. Staff Exh. 124 (I&E Rept. 81-28, at 4); Frazar, *et al.*, ff. Tr. 10,123, at 5-6; Tr. 10,201 (Overstreet).

403. Prior to issuing the stop-work letter, HL&P QA informed HL&P management of its intent. HL&P management requested an opportunity to discuss the issues with B&R in order to resolve them expeditiously. Because HL&P's QA motivation in drafting the stop-work notice was the prompt resolution of the issues, HL&P QA acceded to the request. Staff Exh. 124 (I&E Rept. 81-28, at 4-5); Frazar, *et al.*, ff. Tr. 10,123, at 6. The issues were subsequently and promptly resolved. Frazar, *et al.*, ff. Tr. 10,123, at 9.

404. In July-August 1981, the NRC Staff investigated an allegation, based on the aforementioned events, that HL&P management, in determining that a stop-work notice should not be issued, had failed to support HL&P QA. Staff Exh. 124 (I&E Rept. 81-28, at 4). The investigation revealed no instance where HL&P failed to meet an NRC requirement. Staff Exh. 124.

405. HL&P has written qualitative standards governing the issuance of stop-work notices. Tr. 10,213-15 (Frazar). HL&P QA's stop-work procedure contains no mechanically applied test to determine when a stop-work notice should be issued; rather, an exercise of judgment is required. Frazar, *et al.*, ff. Tr. 10,123, at 6; Tr. 10,153-54 (Frazar).

*Contention 1.8(c)*

406. In July-August 1981, the NRC Staff also investigated an allegation that a member of HL&P QA management had told HL&P audit personnel not to write up NCRs on nonconformances with the FSAR or with the new QA Program Description (QAPD) given to the NRC, because these documents were licensing documents, not regulatory items. Staff Exh. 124 (I&E Rept. 81-28, at 6).

407. Mr. R.A. Frazar, HL&P's corporate QA Manager from April 1977 to June 1980, and from June 1981 to February 1, 1982, and the STP Project QA Manager from 1975 to April 1977 and from June 1980 to June 1981, identified himself as the individual who made the statement (Frazar, *et al.*, ff. Tr. 10,123, at 2, 11).

408. Mr. Frazar's statement was an attempt to resolve a problem that had arisen, that HL&P QA personnel were using documents in their field audits of the B&R site QA program that were different from the documents used by the B&R field personnel. HL&P QA personnel were using upper tier documents, *i.e.*, the FSAR and QAPD, which described the QA program in general terms. B&R field personnel, on the other hand, were using specific procedures that had been written to implement the upper tier documents. Frazar, *et al.*, ff. Tr. 10,123, at 10-11.

409. Mr. Frazar clarified his statement in a letter dated August 24, 1981 (*id.* at 12).

410. The NRC investigation concluded that the letter resolved the misunderstanding (Staff Exh. 124 (I&E Rept. 81-28, at 3)). The investigation found no instance where HL&P failed to meet an NRC requirement (Staff Exh. 124).

*Contention 1.8(d)*

411. In July-August 1981, the NRC also investigated an allegation that two individuals in the HL&P QA Procurement Program were "screwing up everything" because they lacked experience. It was also alleged that they were the only ones who could write up NCRs; and when

other HL&P QA personnel requested that an NCR be written, these individuals would refer the requester to B&R. Staff Exh. 124 (I&E Rept. 81-28, at 8).

412. The NRC investigation concluded that HL&P QA procurement personnel were instructed properly by HL&P QA management in regard to initiating NCRs (*id.* (I&E Rept. 81-28, at 2)). The investigation determined that the individuals had sufficient experience (*id.* at 8-9). Although it was substantiated that on one occasion HL&P QA personnel had been referred to B&R by their supervisor when they sought an NCR, it was determined that the referral was not in violation of HL&P procedures (Staff Exh. 124).

### *Contentions 1.8(a)-(d) Summary*

413. Based on the foregoing, uncontroverted findings, the Board finds that Allegations 1, 2, and 4 in I&E Rept. 81-28 (and Contentions 1.8(a)-(d)) have been satisfactorily rebutted and that no violation of Appendix B criteria has been demonstrated. Therefore, these allegations do not prevent our findings pursuant to 10 C.F.R. § 50.57.

### *Contention 2*

414. Contention 2 alleges:

> NRC inspection records (Inspection and Enforcement Reports #77-03, 2/77; #77-03, 4/77, and #78-08, 5/78) indicate that South Texas Project construction records have been falsified by employees of Houston Lighting and Power Company and Brown and Root, in violation of 10 C.F.R. Part 50, Appendix B, Sections VI and XVII.
>
> As a result, the Commission cannot make the findings required by 10 C.F.R. §§ 50.57(a)(1) and (2).

415. With respect to the alleged record falsifications covered by this contention, the Applicants presented testimony by W. Stephen McKay, Corporate Manager for QA, Pittsburgh Testing Laboratory (PTL) and Timothy K. Logan, Senior QA Engineer and (since 1977) Lead Engineer for HL&P. (These witnesses comprised a portion of the Pettersson panel, ff. Tr. 5796.) The Applicants also addressed this contention through Richard Buckalew, a systems engineering technician for B&R, and John B. Duke, a former STP QC Inspector for B&R (Buckalew/Duke, ff. Tr. 6265). The Staff presented the Seidle and Crossman panels with respect to this contention (*see* Findings 40 and

19, respectively) (Seidle, *et al.*, ff. Tr. 9205; Crossman, *et al.*, ff. Tr. 10,010).

416. On February 1, 1977, HL&P reported to the NRC that an employee of PTL, a subcontractor, had falsified records to indicate that he had tested concrete sand for gradation, when in fact he had not. Staff Exh. 1 (I&E Rept. 77-03, at 2-3); Seidle, *et al.*, ff. Tr. 9205, at 11-12; Pettersson, *et al.*, ff. Tr. 6227, at 5.

417. In February 1977, the NRC investigated the reported falsification and substantiated it (Staff Exh. 1 (I&E Rept. 77-03)). The PTL employee admitted not performing the tests, and his employment was terminated (*id.* at 4; Pettersson, *et al.*, ff. Tr. 6227, at 5). The Staff investigation established that neither the Applicants' management nor their contractors knew of the falsification prior to its being reported by a PTL employee (Seidle, *et al.*, ff. Tr. 9205, at 12, 14). The Staff inspectors attributed the record falsification to production pressure (Staff Exh. 1 (I&E Rept. 77-03, at 6)).

418. Other tests had been performed that assured the adequacy of the concrete constituents (Staff Exh. 1 (I&E Rept. 77-03, at 2); Pettersson, *et al.*, ff. Tr. 6227, at 10-11; Seidle, *et al.*, ff. Tr. 9205, at 12-14). PTL also conducted a review to detect anomalies in its test results (I&E Rept. 77-05, at 1-1 (Staff Exh. 2); Seidle, *et al.*, ff. Tr. 9205, at 13-14). A follow-up investigation concluded that corrective action had been performed and that no further action was necessary (Staff Exh. 2 (I&E Rept. 77-05, at 4, 6)).

419. The falsification of construction records (Finding 417) reflects a violation of 10 C.F.R. Part 50, Appendix B, Criteria VI. There was no culpability of HL&P or B&R management, although B&R might have taken steps to mitigate production pressures. As a result of the tests and investigations, it does not appear that the falsification in fact had safety significance.

420. I&E Report 78-08 (Staff Exh. 3), also cited in Contention 2, did not address any instances or allegations of falsified construction documents. I&E inspectors did review records related to placements of concrete, but no items of noncompliance or deviations were identified (*id.*).

421. An inspection in April 1978, I&E Report 78-07 (CCANP Exh. 10), found that a QC inspector had marked a record print to indicate that an inspection was complete, when in fact it was not; a bolted joint of four structural beams had only been partially inspected (and the joint was so marked), but the record print indicated that the inspection was complete. *Id.*, Appendix A (Notice of Violation), at 3-4; I&E Rept. 78-07, at 8.

422. The investigation determined that B&R QC Procedures did not contain a definition differentiating a connection from a joint (which might comprise several connections). As a result, inspectors used different methods for marking bolt inspections on record prints; some inspectors marked each connection of a joint, while others marked the whole joint. This definitional deficiency contributed to the QC inspector's misinterpretation of the inspection requirements. *Id.* at 8-9; Pettersson, *et al.*, ff. Tr. 6227, at 11-13. The investigation did not find that there had been a deliberate falsification (CCANP Exh. 10). The NRC closed out the incident in I&E Rept. 78-11 following a procedure revision by B&R (Pettersson, *et al.*, ff. Tr. 6227, at 12).

423. Although not explicitly covered by the allegations of Contention 2, the Staff presented evidence concerning instances of alleged falsification of documents. In May 1978, the NRC investigated alleged falsification of cadweld records. The investigation did not substantiate the allegations, and identified no items of noncompliance. Staff Exh. 7 (I&E Rept. 78-09); *see* Findings 56, 338-345, 360-367, *supra.*

424. Two other instances of substantiated document falsification appear in I&E Reports 80-14 and 80-21 (two documents, one admittedly falsified) (Staff Exhs. 60 and 67). The first, involving a falsified fabrication checklist, was done by a field inspector without the knowledge of either HL&P or B&R management. Shortly after the issuance of I&E Report 80-14, the individual was terminated. Crossman, *et al.*, ff. Tr. 10,010, at 15. The second, concerning falsification of two maintenance records, also did not involve management. The individuals involved in this incident were removed from safety-related work and in one instance terminated. *Id.* at 15-17; Tr. 3781, 4946 (Broom); Tr. 4946 (Grote); Tr. 4159 (Vurpillat). *See also* Findings 90-91, *supra.*

425. Based on the foregoing findings, the Board finds that the inspection reports to which CCANP alludes (and the allegations and conclusions therein) do not prevent our findings pursuant to 10 C.F.R. § 50.57.

## II. CONCLUSIONS OF LAW

Based upon the foregoing findings of fact and upon consideration of the entire evidentiary record in this proceeding, the Board makes the following conclusions of law, recognizing that such conclusions may be subject to change based on the record in the forthcoming phases of this hearing:

(1) There is no basis for concluding that at this time HL&P lacks managerial competence or character, as those terms are used in the Atomic Energy Act, as amended, and the Rules and Regulations of the

Commission, sufficient to preclude an eventual award of operating licenses for STP.

(2) There is reasonable assurance that safety-related construction work thus far completed at STP is adequate to perform its intended purpose or that such work will be repaired or replaced as necessary to make such construction work adequate to perform its intended purpose, in conformity with the construction permits, the Atomic Energy Act, as amended, and the Rules and Regulations of the Commission.

(3) No construction deficiencies have been identified which would preclude this Board from making the findings required by 10 C.F.R. § 50.57(a)(1) and (2).

(4) HL&P is presently managing, planning, and implementing its program for the balance of design and construction of STP, including its QA program, in a manner which provides reasonable assurance that future design and construction work at STP will be in conformity with the construction permits, the Atomic Energy Act, as amended, and the Rules and Regulations of the Commission.

(5) Based on HL&P's performance in the management of design, construction and planning and preparation for operation of the STP, there is now reasonable assurance that HL&P will have the necessary managerial competence and character (including commitment to safety) to operate the STP safely and in compliance with all applicable NRC requirements.

## Order

On the basis of the foregoing Findings of Fact, Conclusions of Law and Opinion, and the entire record, it is, this 14th day of March 1984,

ORDERED

1. CLI-80-32 Issues A-E, and Intervenors' Contentions 1 and 2, are *resolved* as set forth in this Decision and subject to the terms and conditions set forth herein.

2. The NRC Staff is *directed,* and other parties are *permitted,* to provide the Board during the Phase II evidentiary hearings with the report set forth under Issue B, p. 697, *supra.*

3. CCANP's August 8, 1983, motion to reopen the Phase I record is denied.

4. In accordance with 10 C.F.R. §§ 2.760, 2.762, 2.764, 2.785, and 2.786, this Partial Initial Decision shall become effective immediately and will constitute, with respect to the matters resolved herein, the final

decision of the Commission thirty (30) days after issuance hereof, subject to any review pursuant to the above-cited Rules of Practice. Any party may take an appeal from this decision by filing a Notice of Appeal within ten (10) days after service of this Partial Initial Decision. Each appellant must file a brief supporting its position on appeal within thirty (30) days after filing its Notice of Appeal (forty (40) days if the Staff is the appellant). Within thirty (30) days after the period has expired for the filing and service of the briefs of all appellants (forty (40) days in the case of the Staff), a party who is not an appellant may file a brief in support of, or in opposition to, any such appeal(s). A responding party shall file a single, responsive brief *only,* regardless of the number of appellants' briefs filed. [*See,* in particular, 10 C.F.R. § 2.762, as amended effective December 19, 1983, 48 Fed. Reg. 52,282, 52,283 (November 17, 1983).]

THE ATOMIC SAFETY AND
LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Dr. James C. Lamb
ADMINISTRATIVE JUDGE

Ernest E. Hill
ADMINISTRATIVE JUDGE

[Appendices A, B, C, and D have been omitted from this publication, but may be found in the NRC Public Document Room, 1717 H Street. NW, Washington, DC 20555.]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Dr. Robert M. Lazo**, Chairman
**Dr. Cadet H. Hand, Jr.**
**Dr. Peter A. Morris**

**In the Matter of** | **Docket No. 50-309-OLA (ASLBP No. 80-437-02-LA)**

**MAINE YANKEE ATOMIC POWER COMPANY**
**(Maine Yankee Atomic Power Station)**

**March 9, 1984**

Upon review of an Agreement reached among the parties, the Licensing Board grants intervenors' motions to withdraw their contentions and requests for hearing, and authorizes the issuance of a license amendment.

# ORDER

## GRANTING MOTIONS TO WITHDRAW CONTENTIONS, GRANTING MOTION TO WITHDRAW A PORTION OF APPLICATION, AND AUTHORIZING ISSUANCE OF AMENDMENT TO OPERATING LICENSE

On October 24, 1979, the Nuclear Regulatory Commission published in the *Federal Register* a notice of proposed issuance of amendment to facility operating license, 44 Fed. Reg. 61,273, in connection with the application of Maine Yankee Atomic Power Company (Licensee) to expand the storage capacity of the spent fuel pool at its nuclear electric power generation facility in Wiscasset, Maine. This increase in storage capacity was to be accomplished through a modified spent fuel pin storage concept involving the disassembly of spent fuel assemblies and reassembly into consolidated fuel bundles within the existing fuel racks. Thereafter, the Licensee supplemented its application to seek permission to increase the number of storage locations by "reracking" the spent fuel pool and to utilize the cask laydown area for short-term storage when necessary. In light of that supplementation, the Nuclear Regulatory Commission, on January 28, 1981, published a Supplemental Notice of Proposed Issuance of Amendment to Facility Operating License. 46 Fed. Reg. 9315.

As a result of the filing of appropriate petitions and contentions, a hearing was noticed, Sensible Maine Power ("SMP") and the State of Maine ("Maine") were admitted as intervenors and full parties to the proceeding, and certain contentions were admitted into litigation in the proceeding. Subsequently, the Licensee, the Staff and the intervenors reached agreement among themselves as to resolution of the proceeding as a result of which certain motions have been filed with the Board.

Two of the motions are filed by SMP and Maine, based on an Agreement with Maine Yankee dated January 30, 1984, and are conditional motions for leave to withdraw their requests for a hearing and all of their contentions already admitted into litigation in this proceeding. These motions are *granted* and the contentions of SMP and Maine are hereby deemed withdrawn.

Also before the Board is a conditional motion by the Licensee for leave to withdraw a portion of its application. As it currently stands, the Licensee's application as amended to and including Supplement 3 filed July 21, 1982 for an operating license amendment seeks an amendment which would permit the following spent fuel storage measures:

(a) Replacement of the existing spent fuel racks with new spent fuel racks in which the storage location center to center spacing is 10.25 inches. Reracking will increase the number of spent fuel permanent storage locations from the 953 now existing to 1476.

(b) Storage of spent fuel in the new racks described in (a) above in the form of consolidated fuel assemblies. This concept is also referred to as pin storage. Application of this concept at Maine Yankee will allow either up to the equivalent of 2038 spent fuel assemblies to be stored in consolidated form while at the same time reserve 217 permanent storage spaces for full core reserve, or up to the equivalent of 2390 spent fuel assemblies in consolidated form by completely filling all 1476 locations with stored pins.

(c) *Temporary* storage of up to 121 spent fuel assemblies in their as discharged form in an emergency rack located in the spent fuel cask laydown area. The temporary storage option reduces the number of permanent rack locations which should prudently be held in reserve for removal of the entire core for inspection of the vessel or other short term operations.

By its motion, the Licensee seeks to withdraw so much of its application as would allow it to consolidate more than twenty standard fuel assemblies.

This motion is *granted.*

Finally, there is before the Board a joint motion of all parties for entry of an order authorizing issuance of an amendment to the operating license in conformity with the application as modified by the above-described withdrawal. This motion reflects the Agreement reached by the parties. The Board having independently considered this matter, and having found nothing in the proposed order which would in any way compromise the public health and safety and in light of the policy expressed in 10 C.F.R. § 2.759 encouraging settlements of controversies, it is hereby, in accordance with the Atomic Energy Act, as amended, and the regulations of the Nuclear Regulatory Commission, ORDERED:

That the Director of Nuclear Reactor Regulation is authorized to make appropriate findings in accordance with the Commission's regulations and to issue an appropriate license amendment authorizing the applicant to rerack as requested to store 1476 standard fuel assemblies in the spent fuel pool, to permit no more than 20 standard fuel assemblies to be consolidated, and to temporarily store up to 121 standard spent fuel assemblies in the emergency rack in the cask laydown area.

Dr. Peter A. Morris and Dr. Cadet H. Hand, Jr., concur in this memorandum and order.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Robert M. Lazo, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 9th day of March 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**James L. Kelley,** Chairman
**Dr. James H. Carpenter**
**Glenn O. Bright**

**In the Matter of**

**Docket Nos. 50-400**
**50-401**
**(ASLBP No. 82-468-01-OL)**

**CAROLINA POWER & LIGHT COMPANY and**
**NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY**
**(Shearon Harris Nuclear Plant, Units 1 and 2)**

**March 15, 1984**

On requests for reconsideration, the Licensing Board rejects certain health effects contentions relating to estimates of genetic damage and cancer caused by radiation because a previously expected Board witness had become unavailable and because it appeared that the Intervenors' proposed witnesses could not shed any additional light on the contentions. The Board also rules on several other contentions and procedural questions.

# MEMORANDUM AND ORDER

**(Ruling on Responses to the Memorandum and Order of January 27, 1984 Concerning Health Effects and Certain Other Matters)**

## INTRODUCTION

The Board's Memorandum and Order of January 27, 1984 (LBP-84-7, 19 NRC 432) has evoked a range of responses from the parties. As we stated in the telephone conference call of March 8, 1984, Dr. John Gofman will not be available as a voluntary Board witness in the late spring-early summer time frame we had envisioned. As we shall explain, this development renders the bulk of the pleadings before us moot.

## THE APPLICANTS' MOTIONS FOR RECONSIDERATION AND CLARIFICATION

The Applicants ask us to reconsider and grant their motions for summary disposition of Joint Contention II(a) and (c). Given Dr. Gofman's unavailability, we agree that the motion as to Contention II(a) should be granted, but not entirely for the reasons cited by the Applicants.

First of all, we reaffirm the rationale of our January 27 Order. Contrary to the Applicants' argument (Motion at 2) we do not think that we have "run afoul" of any NRC rules or decisions. It is apparent to us, however, that the Applicants did not fully understand our rulings on those contentions. *See,* in particular, Motion at 9, last paragraph. We restate our rationale in its essentials, as follows:

(1) There are material issues of fact concerning Dr. Gofman's cancer and genetic risk estimates which the Applicants and Staff have failed to negate.

(2) However, on basic health effects issues which challenge the BEIR estimates, the *Black Fox* decision (*Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), CLI-80-31, 12 NRC 264 (1980)) requires as a precondition to a hearing not only a material issue of fact, but also a showing by the Intervenor that he can make a substantial contribution to its resolution. Non-expert cross-examination is not sufficient.

(3) Here, the Intervenors' proposed case consisted of cross-examination and the testimony of Drs. Ernest Sternglass and Carl Johnson. We held that such a case does not meet the *Black Fox* substantiality test.

(4) It appeared, however, that Dr. Gofman would be available to testify, and his testimony would have satisfied the substantiality test. As long as that was the case, we indicated that Drs. Sternglass and Johnson could also testify on the admitted issues, subject to *voir dire* challenge.

Now that Dr. Gofman is no longer in the picture, the justification for a hearing on Joint Contention II(a) has been removed. As we stated in the January 27 Order:

> If Dr. Sternglass were the only person available as an opposing expert witness, we would grant the motions for summary disposition, notwithstanding the existence to some disputes over material facts.

19 NRC at 438. We also found that the proposed testimony of Dr. Johnson would not justify convening a hearing. Accordingly, the motion for summary disposition of Joint Contention II(a) is granted, notwithstanding the existence of disputes over genuine issues of fact. We recognize, of course, that our ruling represents a departure from a general principle of summary disposition law — that the remedy is not available where material issues of fact remain. But we think this departure is required by the *Black Fox* ruling. If we are wrong, it means that Boards and parties may be obliged to go to time-consuming and expensive hearings on generic issues having no particularized relationship to the facility in order to listen to a witness like Dr. Ernest Sternglass. We do not believe the Commission in writing the *Black Fox* decision could have had that in mind.

During the telephone conference call of March 8, Mr. Wells Eddleman suggested that he might seek to subpoena Dr. Gofman. We have considered whether the prospect of bringing Dr. Gofman to the witness stand involuntarily should be sufficient to satisfy the *Black Fox* substantiality test. We think not. Dr. Gofman made it clear to us that he has conflicting commitments and does not wish to appear. Our experience has shown that an unwilling witness is generally less helpful than a willing one. More importantly, a witness can be compelled to attend, but he cannot be compelled to prepare, much less to write advance testimony. Particularly in an area as arcane as cancer risk estimates, a witness would need to spend substantial time on preparation to be effective. It would also be important, in fairness to the opposing parties, for a witness like Dr. Gofman to prepare fairly detailed written testimony in advance.

The remaining points raised by the Applicants no longer require a response, other than consideration of Contention II(d) (see below).

## THE NRC STAFF'S RESPONSE

This response is rendered moot by Dr. Gofman's unavailability and our grant of summary judgment on Contention II(a).

## WELLS EDDLEMAN'S REQUESTS FOR CLARIFICATION AND OBJECTIONS

Mr. Eddleman asks whether, if health effects are shown to be "far larger" than the Staff's estimates, a new contention needs to be filed on pain and suffering? The answer would have been affirmative if Contention II(a) had remained in the case. Since that contention has now been excluded, there will be no occasion to consider pain and suffering in this case.

The discussion at pages 2-3 concerning Dr. Johnson and his qualifications is confusing. Mr. Eddleman complains that we granted summary judgment on 8F(2) "without reference to the planned testimony of Dr. Johnson." That is correct. We did not consider the question whether Dr. Johnson's testimony would meet the *Black Fox* test of substantiality on that contention. That question is not reached until one first concludes that a genuine issue of material fact exists. We answered that first question in the negative as to Contention 8F(2).

In that connection, we note our view that the *Black Fox* "substantiality" test is relatively narrow and probably applies only to estimates of radiation-induced cancers and genetic effects. These estimates are technically very complex and rest on rather sparse data and incompletely developed theory. Thus the possibility of an intervenor's making a contribution on such questions, unassisted by a highly qualified expert, is remote. To put it a different way, the *Black Fox* test should not apply broadly to all "health effects" issues, some of which can be effectively addressed by an intervenor without necessarily producing expert witnesses. Thus we are not applying a substantiality test to the remaining health effects issues in this case — II(c), II(e) (concerning fly ash), and 8F(1).

In Mr. Eddleman's February 6, 1984 filing, he states that

> I believe the Board is simply wrong to say that because the nuclides from the fuel cycle are dispersed over larger areas, they are therefore of less concern. Under the linear hypothesis used by BEIR, the number of cancers would be the same whether the nuclides deliver their dose to a few people or to many, as long as the dose is the same in total.

This response by Mr. Eddleman comes from the Board statement that "the concern we expressed there about the possibility of aggregate doses to people living near the [Harris] facility does not apply to fuel cycle effluents, which are dispersed over many different geographical areas." LBP-84-7, 19 NRC at 462.

Certainly dispersion over larger areas results in exposure of more people *but,* because the concentration of fuel cycle effluent is drastically reduced during the process of dispersion, the dose to individuals and individual risk are greatly reduced. However, in a statistical sense, one can still calculate a cancer risk estimate. As an example, the fuel cycle radon emissions were considered by the Appeal Board in *Philadelphia Electric Co.* (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-701, 16 NRC 1517 (1982) with the conclusion that the fuel cycle radon emissions from the operation of any particular reactor (nominal 1000 MW(e)) would increase the risk of cancer induction by radon exposure by one part in 100,000 above the unavoidable risk that the U.S. population experiences from natural sources. The Appeal Board considered this *incremental* health risk to the population to be "vanishingly small." This Board agrees that such effects are *de minimus* and this is the basis for our statement concerning the fuel cycle effluents.

Mr. Eddleman, in his request for clarification on Contention 8F(2), states that "I don't understand" why time periods corresponding to the half-lives of the longest-lived nuclides are not considered in health effects estimates. The Board, in our brief comments on this matter, stated that such an estimation would be a speculative exercise. We attempt to clarify our views by noting that to consider doses over millions of years would require some delineation of the following aspects:

1. The location of the materials would be extremely uncertain as a result of substantial geomorphic changes from erosional and depositional processes on such a time scale; *i.e.,* some of the materials could be transported to the deep-sea sediments through erosion and some of the material might be buried due to depositional processes that are important even on archaeological time scales.
2. The location on the surface of the earth of what is now called the State of North Carolina would be problematical due to the drift of the continents. If the present rate of drift (Durham is 0.5 inch further from London each year) were to persist for millions of years, Durham might be located where Fairbanks, Alaska is currently positioned or some other position with large climatic differences.

3. The human species, *Homo sapiens,* is thought to have existed for roughly one million years in the past. Even the existence of humans on a time scale of millions of years into the future is a matter of uncertain guesswork.

These are only a few of the considerations that lead us to conclude that estimates of radiation doses into a future of millions of years would be gross speculation. The Board cannot see that there is any chance that the witness proffered by Mr. Eddleman would be able to shed any light on such intractable questions.

Mr. Eddleman seeks clarification of footnote 1 on page 44 of the January 27 Order (19 NRC at 460 n.1). We find Mr. Eddleman's arguments unclear. We think footnote 1 is clear and we reaffirm it. There need not be, in our view, any explicit NRC rule authorizing an offset of coal particulates from plants to be displaced by Harris against the particulates postulated in Table S-3. Absent a rule barring such an offset (there is none) this approach is realistic and therefore appropriate.

## THE JOINT INTERVENORS' RESPONSE

The Joint Intervenors ask whether they will be required to employ health effects experts on the admitted contentions. As we indicated previously, the admitted contentions — II(c), II(e) (fly ash), and 8F(1) are not so complex that expert assistance is essential, as witnesses or cross-examiners. Thus the question is answered in the negative. However, the Joint Intervenors are certainly encouraged to obtain the services of health effects experts.

## EDDLEMAN MOTION TO DECLARE APPLICANTS' SUMMARY DISPOSITION MOTION ON CONTENTION 15-AA UNTIMELY

Mr. Eddleman moves to declare this summary disposition motion untimely or, in the alternative, for a 20-day extension of time to respond to it. We agree with the Applicants that the previously established schedule calling for summary disposition motions by September 1, 1983 did not contemplate a contention like 15-AA, which was admitted in late August. Under all the circumstances, including particularly the postponement of the January 1984 hearing, it is certainly appropriate that we entertain a summary disposition motion on that contention now. Much of Mr. Eddleman's argument concerning what the Applicants may have proposed in the past on scheduling questions is not relevant. The motion for declaration of untimeliness is denied.

Mr. Eddleman's alternative motion for a 20-day extension is cast in general terms and normally would not justify the full extension requested. However, the Board is aware that Mr. Eddleman has been quite ill of late and this has undoubtedly created a backlog of work. Under the circumstances, including the need to move the case forward, Mr. Eddleman is granted a 15-day extension, until April 13, 1984, to answer the Applicants' summary disposition motion on Contention 15-AA.

## JOINT CONTENTION II(d)

This subpart of Contention II states:

> (d) Substantial increases in cancer mortality rates have been observed in the vicinity of nuclear facilities. Sternglass, "Cancer Mortality Changes Around Nuclear Facilities in Connecticut," February 1978.

The Board inadvertently failed to rule on this contention in our previous order. We do so now.

This contention rests solely on an unpublished manuscript of Ernest J. Sternglass. The manuscript has received substantial public attention and was reviewed, therefore, by representatives of the U.S. Department of Health, Education and Welfare and the U.S. Environmental Protection Agency (*see* Attachments A and B, *infra*). These reviewers found the document to be fallacious.

As a result of Sternglass' repeated allegations, misrepresentations of facts and distorted scientific perspective, the president-elect and all living past presidents of the Health Physics Society (including Karl Morgan) unanimously signed and issued in July 1971 the following statement at the 16th Annual Meeting of the Health Physics Society, thereby publicly rejecting Sternglass' allegations and criticizing his past papers:

> On the third such occasion since 1968, Dr. Ernest Sternglass has, at an annual meeting of the Health Physics Society, presented a paper in which he associates an increase in infant mortality with low levels of radiation exposure [from discharges from nuclear facilities]. The material contained in Dr. Sternglass' paper [Sternglass, 1971] has also been presented publicly at other occasions in various parts of the country. His allegations, made in several forums, have in each instance been analyzed by scientists, physicians and bio-statisticians in the Federal government, in individual States that have been involved in his reports, and by qualified scientists in other countries. Without exception, these agencies and scientists have concluded that Dr. Sternglass' arguments are not substantiated by the data he presents. The

United States Public Health Service, the Environmental Protection Agency, the States of New York, Pennsylvania, Michigan and Illinois have issued formal reports in rebuttal of Dr. Sternglass' arguments. We, the President and Past Presidents of the Health Physics Society, do not agree with the claim of Dr. Sternglass that he has shown that radiation exposure from nuclear power operations has resulted in an increase in infant mortality.

With specific reference to the Sternglass manuscript cited in Contention II(d), the Applicants' affiant, Dr. Jacob Fabrikant, stated that:

These studies are as good examples as any of the long standing problems with Sternglass' approach in his population studies. First, his Millstone conclusions are based on aggregations of vital studies crudely compiled by region. As I have indicated, this is an inappropriate procedure, even if done in good faith.

Second, his conclusions are wrong and his procedure is unprofessional. In his Millstone population study (1978), Sternglass used polemics to push a particular point of view, selected only facts suggesting that view, was illogically inconsistent and failed to consider alternative explanations, including the possibility of random occurrences. Further, the 1978 Millstone report, referenced by Joint Intervenors, is based on an earlier Sternglass report concerning elevated levels of strontium-90 and cesium-137 around the Millstone Plant. This earlier report selectively picked data points and drew erroneous scientific conclusions from his misinterpretation of official, documented records of environmental strontium-90 and cesium-137 levels in the atmosphere and in milk.

(For documentation of these rebuttals and criticisms, see letters and report critiques from Environmental Protection Agency Administrator, Douglas M. Costle, August 9, 1978; Nuclear Regulatory Commission Chairman Joseph M. Hendrie, January 18, 1978, incorporating the review of Professor Marvin Goldman, Director of Radiobiology Laboratory, University of California, Davis, May 31, 1978.)

Dr. Fabrikant's sworn statements, in light of his expert credentials and Dr. Sternglass' lack of same, effectively discredit the Sternglass manuscript. The intervenors filed nothing in opposition to the Fabrikant affidavit. Contention II(d) raises no issue of material fact and, therefore, summary disposition of it is granted.

## CONTENTIONS ON THE FINAL ENVIRONMENTAL STATEMENT

In a pleading dated December 16, 1983, Mr. Eddleman addressed his previously deferred Contentions 8F-3 and 85 in light of the FES. Our rulings follow.

Contention 8F-3 states that "the DEIS does not give sufficient information about how the NRC calculates the doses from Table S-3 radioactive effluents to enable these calculations to be verified as accurate." Mr. Eddleman's argument in support of the contention consists of two parts. The first part discusses cancer deaths and risk of cancer, which the Board finds to be outside the scope of this contention. This material does not contribute to our evaluation of whether or not the contention should be admitted.

The other part of Mr. Eddleman's response leads us to his statement that "I can't figure out their modeling of dose commitments from Appendix C." The Board notes that the NRC Staff took notice of this contention in preparing the FES. In contrast to the DEIS that only referenced Volume 3 of NUREG-0002 for the estimation of the dose commitments for radon-222, the FES references GESMO (NUREG-0002), pages IV JA-20, -21 and -22 and several other parts of NUREG-0002 as being the basis for the dose calculations. Intervenor Eddleman fails to allege any deficiencies in the documents referenced in the FES.

The Board finds that adequate, detailed references for the dose computations are provided in the FES. There is no basis for this contention and admission is denied.

## EDDLEMAN CONTENTION 85

In the original May 14, 1982 filing by Mr. Eddleman, Contention 85 was stated as "CPL has failed to take appropriate measures to prevent or minimize fish kills from the causes discussed below. Contention 86 is incorporated herein by reference for its information."

Contention 86 stated that

> the ES's consideration of fish kills due to hot water discharge into SHNPP reservoir (lake) is inadequate as
> (1) the upper lethal temperature limits at which significant fish mortality occurs have not been established for important fish species (ER 5.1.3-2, amendment 1).
> (2) the time for which any of the ER 2.2.0 important fish species (or others found in the Harris reservoir (lake)) can tolerate the high temperatures in the discharge mixing zone is not established on the basis of experimental data, nor have periods such species of fish (individuals of those species) are likely to spend in the mixing zone been established by actual study or experiment of those same species in lakes like the SHNPP reservoir.
> (3) average, not peak, mixing zone temperatures have been used in the analysis of fish sensitivity to temperature in the mixing zone. Actually, the peak temperature that can be expected during the time and period established by research per (2) above should be used in projecting the probability and numerical occurrences of fish kills from Harris cooling tower blowdown by temperature. Further, the *addition* of heat by cooling tower blowdown above the maximum temperatures of over 31° recorded in the area (ER Ref. 5.1.3-5) or the actual highest temperature recorded in the reservoir without cooling tower blowdown, needs to be the basis in establishing the temperatures in the mixing

zone and the areas over which these maximum temperatures can be expected to extend for the time and periods as calculated above in this (3) and (2).

(4) the additional toxic synergistic effects of the presence of chlorine, hydrazine, ammonia, and other chemicals in addition to elevated temperatures per (3) above in cooling tower blowdown on fish kills in the mixing zone must be considered on a conservative basis. The sensitivity of important fish species per ER 2.2.0 and other fish species found in the reservoir must be established accurately to these combinations of chemical and temperature conditions to be expected.

(5) the original FES did not consider the above accurately because then SHNPP was going to be once-through cooling with no cooling towers, thus cooling tower blowdown chemicals and the mixing zone were very different from the current plan, and plant setup. These changes must be addressed (as set forth above, e.g.) in the striking of the environmental balance under NEPA for SHNPP operating license issuance.

The Board deferred ruling on Contentions 85 and 86 (LBP-82-119A, 16 NRC 2069, 2104 (1982)), pending the availability of the environmental statement. The DES was served on May 11, 1983 and Mr. Eddleman responded on June 20, 1983 with the principal point being that "the DEIS fails entirely to document or lay out Staff's analysis" with respect to fish kills.

The Board issued a Memorandum and Order on August 18, 1983 (unpublished) that further deferred our ruling on this contention until after the issuance of the FES. In that order (at 14), a typographical error in which the contention was numbered 85B was missed by the Board. We acknowledge that oversight and require that all parties use the correct designation of "85" or "85/86."

In our August 18, 1983 Order, the Board took the position that "the Staff's sole function is to factor the impacts previously determined by the EPA (and the State of North Carolina) into the NRC's cost/benefit analysis." However, we further took the position that the Staff analysis should be understandable and demonstrably accurate.

The FES for the Harris plant was issued in October 1983 and Mr. Eddleman responded on December 16, 1983. He continues to raise a number of issues that may or may not be important in the cost/benefit analyses and states that he can't quite figure out how the Staff did the analysis.

The Board cannot resolve these issues with the materials before us and, therefore, the contention is admitted. However, we speculate that these issues can be resolved in an off-the-record meeting between the Applicants' technical staff and Mr. Eddleman. We also suggest that a working hypothesis that might be pursued is that:

1. Construction of cooling towers is a major thermal impact mitigation measure that denies part of this contention, *i.e.*, that CPL has failed to take appropriate measures.
2. The exact area that will be warmer than the North Carolina regulatory standards is not known for certain but both Staff's and Applicants' analyses indicate that it will be smaller than the 200 acres specified in the NPDES permit.
3. While the NRC Staff has yet to make an explicit statement, it is not unreasonable to consider that the Staff view is that 200 acres in a 4000-acre reservoir is 5% of the surface area and the "off-standard" waters will be found approximately in the upper 5 feet of the reservoir, so that the volume in question is 1 to 2% of the total volume. Such numbers may be the basis for the Staff conclusion at 6-2 of the DES that damage to aquatic resources will be "small."
4. The existence of a limited area of "off-standard" waters would have the impact of denying a small part of the potential habitat to the fish populations of the reservoir. Consideration of the above magnitude of such an effect may suggest that an undiscernable effect on the fish production of the reservoir may be anticipated.
5. Based on observations in other reservoirs in North Carolina that receive heated water discharges, it may be demonstrable that the effect of the limited "off-standard" waters will be avoidance of such areas by fishes, rather than fish kills as postulated in this contention.

Following the specified conference, the Board would expect that if specific, focused issues remain, such issues would be identified and the present broad statements would be replaced with litigable issues. The above suggestions by the Board are made in conformance with 10 C.F.R. § 2.759.

## RULINGS ON SER CONTENTIONS

In the telephone conference of March 8, 1984, the Board ruled on Mr. Eddleman's contentions on the SER on the bases of his January 17, 1984 filing and the Applicants' and Staff's responses to it. At that point, we had forgotten that we had granted Mr. Eddleman extensions of time to reply to the responses. We received Mr. Eddleman's reply shortly after the conference call. We have reconsidered each of our rulings on the SER proposed contentions in light of Mr. Eddleman's reply. While we agree with some of the points Mr. Eddleman makes — *e.g.*, that his

TDI diesel generator contentions are not untimely — Mr. Eddleman's reply does not cause us to change any of our rulings, and those rulings are reaffirmed.

THE ATOMIC SAFETY AND
LICENSING BOARD

Glenn O. Bright
ADMINISTRATIVE JUDGE

Dr. James H. Carpenter
ADMINISTRATIVE JUDGE

James L. Kelley, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
March 15, 1984

## APPENDIX A

The Honorable James C. Cleveland
House of Representatives
Washington, D.C. 20515

Dear Mr. Cleveland:

At Dr. Upton's request I have reviewed the manuscript by Dr. Ernest J. Sternglass, entitled "Cancer mortality changes around nuclear facilities in Connecticut," and presented at a Congressional Seminar on Low-Level Radiation, February 10, 1978. In my judgment, this paper is of no value as a guide to the possible carcinogenic risks from radioactive isotopes emitted by nuclear power plants. The paper is logically incoherent and lacking in the balance and scrupulous consideration of alternative explanations that are required of a serious scientific work. The paper is heavily laden with polemics in which selected facts and analogies have been presented in a way designed to push a particular point of view, namely, that increased cancer mortality has been caused by radioactive emissions from nuclear power plants in Connecticut and elsewhere.

Cancer mortality data are subject to a number of influences, e.g., changes in the age and racial makeup of populations, differences in socio-economic class, urbanization, and industrialization which may increase or decrease rates. Random variation is an even more important factor, particularly when small populations are involved. By ignoring these other important factors, it is not difficult to select rates to show an increasing cancer trend associated with almost any environmental factor. Dr. Sternglass's presentation, and his past work on similar subjects, indicate that the necessary care to control for these other factors has not been taken.

Another of the logical inconsistencies in this paper concerns the types of cancer investigated. In the first few pages, the discussion centers around levels of strontium-90, a bone-seeker, and the estimated radiation dose to the bone marrow for children drinking milk from certain dairies. Reference is made to studies linking childhood leukemia with fetal x-ray, and childhood and adult leukemia with the radiation exposures received by the Japanese A-bomb survivors. It is curious, then, that the discussion of death rates does not mention childhood cancer, nor leukemia at any age, but is confined to mortality at ages 50 or older from cancers of the lung, female breast, and pancreas. That is, the case for there being unusually heavy exposures to sensitive tissues is made in such a way as to suggest an increased hazard in terms of childhood leukemia, and perhaps other childhood cancers and adult leukemias, but apparently there is no evidence of such increased risk. Instead, we are told that other radiogenic cancers, whose causal relationship to the discussed exposures seems tenuous at best, have increased due to these exposures. In fact, adult mortality from cancers of the lung, breast, and pancreas has been increasing steadily for a number of years; smoking, dietary factors, drug use, and changing patterns of diagnosis have all had something to do with this.

One of the difficulties in reviewing this paper is that the violations of good scientific practice in it are so many and so varied that it would be a vast undertaking to explicate each one. I have highlighted what I consider to be a few of the major problems, but there are numerous others also.

I am a statistician, professionally concerned with the logic of scientific inference. For the past 5 years or so I have worked principally on epidemiologic investigations of the relationships between radiation dose and cancer incidence and mortality in populations exposed to ionizing radiation, mainly the Japanese A-bomb survivors but also other irradiated populations. I am deeply concerned about radiation-induced cancer and other hazards of radiation exposures, and feel that the use of nuclear and radiological technology should be based on a careful assessment of risks. Papers such as the reviewed one by Sternglass contribute only confusion to this process, and in fact, impede it by deflecting investigative resources from the work at hand. We trust this information will be helpful in your response to Ms. Juliette Zivic.

Yours sincerely,

Charles E. Land, Ph.D.
Environmental Epidemiology Branch
3C07 Landow Building
Bethesda, MD 20014

## APPENDIX B

Honorable James C. Cleveland
House of Representatives
Washington, D.C. 20515

Dear Mr. Cleveland:

This is in response to your letter of July 11, 1978. The Office of Radiation Programs has informally reviewed the report by Dr. E.J. Sternglass entitled "Mortality Changes Around Nuclear Facilities in Connecticut." It is unfortunate that a report of this kind, which was presented to a lay audience without any scientific review, has received the widespread discussion in newspaper articles to which you referred.

Dr. Sternglass has been presenting similar reports for the last 10 years which, on careful analyses, have been shown by a number of reputable scientists to be based on a highly selective and very biased use of mortality data. In every case we have found that Dr. Sternglass only uses data which support his pronounced views which are usually directed against nuclear power.

We believe the public health questions surrounding nuclear power and other sources of population exposure to radiation are too important to be treated irresponsibly. Because of this importance, we asked the National Academy of Sciences to review all recent findings concerning radiation health effects. Their report, which is due this fall, will include a discussion of Dr. Sternglass' reports. While I have no advance knowledge of Academy findings, I would be surprised if they placed much credence in his allegations. Certainly, to date, no reputable scientists have published any reports verifying his analyses.

Sincerely yours,

W.D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs (AW-458)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Helen F. Hoyt, Chairperson**
**Ernest E. Hill**
**David R. Schink**

**In the Matter of**

**ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE**

**(TRIGA-Type Research Reactor)**

**Docket No. 50-170**
**(ASLBP No. 81-451-01-LA)**
**(Renewal of Facility License No. R-84)**

**(Cobalt-60 Storage Facility)**

**Docket No. 30-6931**
**(ASLBP No. 82-469-01-SP)**
**(Renewal of Byproducts Material License No. 19-08330-03)**

**March 15, 1984**

In this Order, the Licensing Board grants the joint motions of Licensee, NRC Staff and Intervenor resolving all remaining issues and dismisses the proceeding.

## ORDER
### (Dismissing Proceeding)

1. By a *Joint Motion for Approval of Stipulation and Withdrawal of Intervenor* served by counsel for Licensee on February 23, 1984 and

signed by counsel for Citizens for Nuclear Reactor Safety (Intervenor), Armed Forces Radiobiology Research Institute (Licensee) and the Nuclear Regulatory Commission Staff, approval by this Board is urged for withdrawal by the Intervenor of all of its contentions. The motion further asserts that upon approval there are no remaining issues in controversy with respect to the renewal of Facility License No. R-84 in Docket No. 50-170 or renewal of Byproducts Material License No. 19-08330-03 in Docket No. 30-6931.

2. In *Joint Stipulation Resolving Intervention by the Citizens for Nuclear Reactor Safety* signed by counsel for each of the parties to these proceedings, the parties assert that based on the Memorandum of Agreement[1] entered into between the Licensee and the Montgomery County (Maryland) Government, the matters raised by the Intervenor are ready for disposition. In addition, the Intervenor stipulates it will not submit any reply to Motions for Summary Disposition filed by Licensee and NRC Staff with this Board on February 25, 1983 and upon entry of the Memorandum of Agreement into force, the intervention on all contentions (including emergency planning) may be deemed terminated.

3. The Board accepts the Joint Stipulation Resolving Intervention by Citizens for Nuclear Reactor Safety with its Exhibit A, grants the Joint Motion for Approval of Stipulation and Withdrawal of Intervenor and grants the Licensee and Staff Motions for Summary Disposition. The Board finds that there are no undisputed facts before it and that all issues have been resolved. Accordingly, there is no need for further proceedings before the Board and these adjudicatory proceedings are dismissed.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Helen F. Hoyt, Chairperson
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 15th day of March 1984.

[1] Copy attached to this Order with transmittal letter signed by Montgomery County Council President.

## ATTACHMENT TO LBP-84-15A

### Exhibit A

December 2, 1983

Col. Bobby R. Adcock, MS, USA
Director, AFRRI
Building 42, NMC, NCR
Bethesda, Maryland 20814

Dear Col. Adcock:

The Montgomery County Council and the County Executive have endorsed and signed the Memorandum of Agreement on the research nuclear reactor at the Armed Forces Radiobiology Research Institute. I am enclosing one copy of the signed agreement for your files.

As stated in the Memorandum of Agreement, the County Government is now in a position to support the application by the Defense Nuclear Agency to the Nuclear Regulatory Commission for the relicensing of this facility and is forwarding copies of this letter and the Memorandum of Agreement to the Nuclear Regulatory Commission to record this support.

I am delighted that this negotiation between the Defense Nuclear Agency, the County Government, and the Citizens for Nuclear Reactor Safety has come to such an agreeable solution and look forward to continued working relationships between DNA and the County Government through the procedures set out in the Memorandum of Agreement.

Sincerely,

David L. Scull
Council President

SMcK/jm

Attachment

cc: Nuclear Regulatory Commission

# MEMORANDUM OF AGREEMENT

The Armed Forces Radiobiology Research Institute (AFRRI) and the Montgomery County Government (MCG), recognizing that concerns have been expressed regarding the relicensing and operations of the TRIGA research reactor and the cobalt-60 facility at AFRRI, hereby agree that the following actions shall be taken by the respective parties:

1. AFRRI shall provide the results of its quarterly environmental monitoring for both air and water to the Montgomery County Government by U.S. mail to:

   Montgomery County Department of
   Environmental Protection
   Executive Office Building, 6th Floor
   101 Monroe Street
   Rockville, MD 20850

   Furthermore, AFRRI shall cooperate with and provide technical advice and information to the Montgomery County Government if it decides to engage in independent radionuclide monitoring of the environment near AFRRI.
2. AFRRI shall, through Headquarters, Defense Nuclear Agency (DNA), immediately notify the Montgomery County Government, through its Department of Environmental Protection (at 251-2400 or number to be designated for after hours), of any Class 1 or higher reactor emergency condition or of any NRC reportable event involving the cobalt-60 facility.
3. AFRRI shall permit Mr. John Menke, Director of MCDEP, to be present and observe at the meetings of the AFRRI Reactor and Radiation Facility Safety Committee (RRFSC). If the Montgomery County Government desires to replace Mr. Menke, the replacement shall be a senior technical professional with some knowledge of reactor physics, employed full-time by MCG. The person shall be approved by both AFRRI and MCG. Notification of meetings shall be at least 5 working days in advance. This is to take effect upon the completion of USNRC licensing renewal action. The MCG observer will be able to ask questions and to enter into conversations relevant to the AFRRI facility at the meetings of the RRFSC.
4. The MCG shall provide AFRRI with copies of the results of any air and water environmental sampling for radionuclides taken by MCG with respect to AFRRI. These shall be sent to:

Director, AFRRI
Building 42, NMC, NCR
ATTN: SAF
Bethesda, MD 20814

5. Upon completion of this agreement the MCG shall promptly inform the Nuclear Regulatory Commission Atomic Safety and Licensing Board that its concerns regarding the relicensing of AFRRI have been satisfied, and that MCG supports renewal of the license and completion of the licensing action.
6. This Memorandum of Agreement shall remain in effect during the term of the license granted by the USNRC in the currently pending actions.

BOBBY R. ADCOCK — 18 Oct. 83
Colonel, MS, USA
Director, AFRRI

CHARLES W. GILCHRIST — 12/6/83
County Executive
Montgomery County Government

DAVID L. SCULL — 12/2/83
Council President
Montgomery County Government

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Lawrence Brenner,** Chairman
**Dr. Richard F. Cole**
**Dr. Peter A. Morris**

| | |
|---|---|
| **In the Matter of** | **Docket Nos. 50-352-OL** |
| | **50-353-OL** |
| **PHILADELPHIA ELECTRIC COMPANY** | |
| **(Limerick Generating Station,** | **March 16, 1984** |
| **Units 1 and 2)** | ***As corrected:* Tr. 8511-13** |

In a written confirmation of an oral ruling, the Board, exercising jurisdiction over a proposed Part 70 license, denies a motion to admit contentions, a motion to stay receipt of new fuel at the Limerick site, and a petition to intervene and request for hearing addressed to the Director of Nuclear Material Safety and Safeguards.

## LICENSING BOARDS: JURISDICTION OVER PART 70 LICENSES

Licensing boards established to conduct hearings on operating licenses also have jurisdiction over issues arising under applications for Part 70 licenses to receive and store unirradiated fuel at the nuclear power plant. This jurisdiction can be asserted on the grounds of 10 C.F.R. § 2.717(b), which grants the presiding officer in an operating license proceeding the power to modify "as appropriate for the purpose of the proceeding" any Staff order "related to the subject matter of the pending proceeding." *Cincinnati Gas and Electric Co.* (William H. Zimmer Nuclear Station), LBP-79-24, 10 NRC 226 (1979). In affirming the *Diablo Canyon* Licensing Board's assertion of jurisdiction over a materials license proceeding, the Commission said, "that license is integral to the Diablo Canyon project .... Given that Board's familiarity with the Diablo Canyon project, it made good practical sense for it to hear and decide the related issues raised by the Part 70 materials license application." *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-76-1, 3 NRC 73, 74 n.1 (1976).

## LICENSING BOARDS: JURISDICTION OVER STAFF ORDERS

Section 2.717(b), which grants the presiding officer in an operating license proceeding the power to modify "as appropriate for the purpose of the proceeding" any Staff order "related to the subject matter of the pending proceeding," does not postpone the board's jurisdiction over the related order until the Staff has actually issued the order. The purpose of Section 2.717(b) clearly is to permit integration of an operating license proceeding with Staff orders on matters related to that proceeding. Common sense says that this integration can take place, indeed is often more efficient if it takes place, before the Staff issues an order on a related matter. *See Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), LBP-83-38, 18 NRC 61, 63 (1983).

## LICENSING BOARDS: JURISDICTION OVER PART 70 LICENSES

Though it is unusual for a judicial body to exercise jurisdiction where it is not sought by the petitioner, a board's exercise of jurisdiction over a petition addressed to the Director of Nuclear Material Safety and Safeguards to intervene on a proposed Part 70 license is not an act of Constitutional dimensions. It makes sense for the board to rule on the petition, for it knows the parties and the circumstances of the case. If the board were to decline jurisdiction now and let the petition follow the path the intervenor intended it to, it would, given past practice, likely be the licensing board delegated the responsibility of conducting a hearing on the subject of the petition.

## RULES OF PRACTICE: ADMISSIBILITY OF LATE-FILED CONTENTIONS

The admissibility of the Intervenors' Part 70 motions, though filed several months after the Applicant filed for a Part 70 license, and years after the start of the operating license hearings, is not to be measured by the criteria for late-filed contentions in 10 C.F.R. § 2.714(a)(1) and *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983), for the Applicant did not comply with a standing order in this proceeding to serve all relevant papers on the Board and parties. An intervenor should be expected to foresee that an Applicant would have to receive unirradiated fuel before low-power testing and that such fuel would have to be outside at the site for a finite time, but not that the Applicant would request that a fuel license be issued before a low-power operating license, or that the fuel might be stored outside for months, or that there would have to be a security plan tailored to such storage because the normal facility security plan would not be implemented as a prerequisite.

## RULES OF PRACTICE: ADMISSIBILITY OF LATE-FILED CONTENTIONS

Despite a standing Board order to serve on the Board and parties papers related to the operating license hearing, the Applicant did not serve its new fuel license application and amendments thereto, thus delaying the Intervenors' responses to the application. The delay has enabled the Applicant to argue that the Intervenors' responses were late-filed. Had the Applicant's argument been accepted, the Applicant, by merely delaying the service of relevant information, would in effect have tightened the standards for admitting contentions. Thus the circumstance here is an exception to the Commission's general belief that manipulation of the availability of licensing documents (here the device of limited service contrary to expectations) was unlikely to occur. *See Catawba, supra,* 17 NRC at 1047.

## ADJUDICATORY HEARINGS: STAFF OBLIGATION TO INFORM BOARD AND PARTIES OF STAFF ACTION

Staff counsel did not learn of the Applicant's application for a Part 70 license until an amended application was filed months later. Staff counsel then informed the Board and the Intervenors of the amended application, thus giving the Intervenors their first information about the original application, but by then the Applicant was already in a position to argue that the Intervenors' filings in response to the original application were late. It may sometimes be difficult for Staff counsel to be relevantly informed. However, the Staff appears before us in these proceedings as one body. Counsel should be informed when its client is considering a Part 70 application. Indeed, the Staff should assure that the Board and all parties in a nuclear facility proceeding, as well as its own counsel, are given prompt notice that a Part 70 license related to the facility is being considered.

### RULES OF PRACTICE: STAY OF AGENCY ACTION

Section 50.91(a)(4), which makes the issuance of an operating license amendment effective before any required hearing only if no significant hazards considerations are involved, does not imply that an intervenor's petition for a hearing on a proposed amendment to a new fuel license could, by virtue of its being filed, stay the effectiveness of any Staff issuance of the amendment.

### RULES OF PRACTICE: APPELLATE REVIEW; JURISDICTION OF APPEAL BOARD

Final orders on motions related to Part 70 licenses to receive and store unirradiated fuel issued during an operating license hearing are appealable upon issuance. *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-76-1, 3 NRC 73, 74 (1976). Appeals should be directed to the Commission, unless the Commission specifically delegates appellate jurisdiction to the Appeal Board. *Id.* at 74 n.1; 10 C.F.R. § 2.785.

### TECHNICAL ISSUES DISCUSSED

New Fuel Stored Outside:
- Criticality Accidents
- Criticality Monitoring
- Non-Criticality Accidents
- Security Plan.

## MEMORANDUM AND ORDER
## ON FOE'S CONTENTIONS AND LEA'S PETITION BASED ON A PART 70 APPLICATION TO STORE NEW FUEL

### INTRODUCTION

Before us are two pleadings from intervenors in this operating license proceeding, both related to Philadelphia Electric's application for a Part

70 license to receive and store unirradiated new fuel outside at the Limerick site.[1] The first of the pleadings, dated February 23, 1984, is styled an "Application by Anthony/FOE to File a Contention Based on New Matter, *i.e.,* PECo's Application Part 70 to Store Fuel at the Limerick Plant, Served 2/21/84." The "Application Part 70" to which the title of the pleading refers is an amended application the Applicant submitted on January 24, 1984. The "Application by Anthony/FOE" is a filing of contentions alleging certain dangers from the presence of new fuel at the site. On February 28, 1984, Mr. Anthony filed "additions" to his "Application." The second pleading, dated February 28, 1984, is Limerick Ecology Action's (LEA) "Petition to Intervene and Request for Hearing Pursuant to the Atomic Energy Act, as Amended January 4, 1983, P.L. 97-415, Section 12(a)." The Petition is in response to the Applicant's February 6, 1984 revision of the amended application it had submitted on January 24. The Petition does not present contentions; it claims that the Applicant's February 6 revision "fails to comply with applicable statutes and Commission rules and regulations" as to general categories which LEA recites. LEA filed its Petition not with this Board, but with the Commission, through the Director of the Office of Nuclear Material Safety and Safeguards. In September 1983, the Staff granted the Applicant a license to receive milligram quantities of sealed source material for instrumentation, but it has not yet granted the Applicant a license to receive and store fuel.

In an order dated February 27, 1984, we set a schedule for the Staff's and the Applicant's responses to FOE Part 70 contentions, and we heard argument on those contentions and on LEA's Petition during prehearing conferences and hearings the week of March 5. On March 6, we made an oral ruling not admitting FOE's contentions and denying LEA's Petition. Tr. 7909-39. We are now confirming in writing that oral ruling.

We made our ruling subject to certain affidavits we ordered the Staff and the Applicant to submit. The requirements for those affidavits and the schedule related to them are discussed below at pp. 870-71. We expect those affidavits, and the responses we have invited FOE and LEA to make to them, to confirm our ruling, and if they do, we shall issue an order making final what we ruled on March 6. The grounds of our ruling were substantive: Most of the concerns[2] raised by the contentions and the Petition are about the health and safety of the public

[1] The application says that storage outside would be for an interim period of a few months. Sec. 1.2.1. The Applicant is now predicting a shorter period of time. Tr. 7869. Apparently, storage outside is made expedient by a combination of economic and scheduling factors. *Id.*

[2] An exception, which we discuss later, is LEA's concern about security plans for the new fuel.

due to accidents damaging the fuel, but the presence of unirradiated fuel at Limerick does not threaten the health and safety of the public.

However, before we can set out here in some detail the substantive basis for our ruling, we must deal with certain procedural questions made somewhat complicated by the Commission's rules and the procedural implementation of them by the technical Staff, and by the way in which the parties, especially the Applicant, have dealt with this Part 70 matter.

The Applicant filed back on June 1, 1983, its original application to receive and store fuel at the site (outside for an interim period of some months), but, despite a long-standing Board order on service of papers related to the operating license proceeding, the Applicant informed neither the Board nor the Intervenors of this original application, or the January 24 amendment, or the February 6 revised amendment. Indeed, not even Staff counsel were aware of the June application until they learned of the January 24 amendment. The Board and the other parties learned of the application amendments through Staff counsel by letters dated February 13 and 15, 1984. Now we must consider whether filings essentially responding to a license application made last June, and concerning an event arguably foreseeable when this hearing began years ago, are not late-filed. We find that they are not. But we must also consider whether this Board, established to conduct a hearing on an application for an operating license, has jurisdiction over filings responding to a not-yet-granted application for a fuel license — and more, whether we have jurisdiction over one such filing even when it is not directed to us. We find that we do. We consider the question of jurisdiction first.

## THE BOARD'S JURISDICTION OVER PART 70 MATTERS

In our February 27, 1984 order setting a schedule for answers to FOE's Part 70 contentions, we directed the parties' attention to two cases which argued the legality and good sense of giving licensing boards established to conduct operating license hearings jurisdiction over proceedings for nuclear fuel materials licenses for the same plant: *Cincinnati Gas and Electric Co.* (William H. Zimmer Nuclear Station), LBP-79-24, 10 NRC 226 (1979), and *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-76-1, 3 NRC 73, 74 n.1 (1976). This jurisdiction can be asserted on the grounds of 10 C.F.R. § 2.717(b), which grants the presiding officer in an operating license proceeding the power to modify "as appropriate for the purpose of the proceeding" any Staff order "related to the subject matter of the pending proceeding." Behind our application of 2.717(b) to FOE's contentions lies the practicality spelled out by the Commission in affirming the *Diablo Canyon* Licensing Board's assertion of jurisdiction over a materials license proceeding: "[T]hat license is integral to the Diablo Canyon project .... Given that Board's familiarity with the Diablo Canyon project, it made good practical sense for it to hear and decide the related issues raised by the Part 70 materials license application." 3 NRC at 74 n.1.

Despite *Zimmer* — in which PECo's law firm and counsel took part on behalf of the Zimmer applicant — and *Diablo Canyon,* the Applicant argues that our jurisdiction does not cover FOE's contentions. Applicant's March 1, 1984 Answer at 2 n.4; Tr. 7803. The Applicant claims that there is no "reasonable argument that the hazards this Board is looking at in the [operating] license proceeding" are related to the materials license application. Tr. 7804. But recognizing that precedent does not support its claim, the Applicant goes on to argue that since in *Zimmer* and *Diablo Canyon,* the materials licenses in question had been issued before the Boards took jurisdiction over them, we cannot take jurisdiction over Part 70 matters until the Staff has actually issued the fuel license. The Staff agrees with this reading of *Zimmer* and *Diablo Canyon,* and cites a case in which a licensing board declined jurisdiction over a pending application for a fuel license, *Pennsylvania Power and Light Co.* (Susquehanna Steam Electric Station, Units 1 and 2), Docket Nos. 50-387 and 50-388, slip op. at 29 (May 29, 1981) (unpublished).

Our jurisdiction over Part 70 matters is not to be so narrowly construed. It is true that when § 2.717(b) speaks about the presiding officer's power, it is explicit only about that officer's power to modify a Staff order, but the purpose of § 2.717(b) clearly is to permit integration of an operating license proceeding with Staff orders on matters related to that proceeding. Common sense says that this integration can take place, indeed is often more efficient if it takes place, before the Staff issues an order on a related matter. If a board waits until after the order is issued to take jurisdiction, it will then find stay applications harder to deal with. As legal authority for our position here, we again cite *Diablo Canyon.* The Commission's remarks there about the practical good sense in letting a licensing board hear and decide related issues raised by a materials license application apply no less to issues raised before the Part 70 license is granted than they do to issues raised after the license is granted. We find our reading of *Diablo Canyon* confirmed by the Licensing Board in the *Perry* proceeding: "Whether or not we could grant or deny the [fuel license] application before staff acted is merely a wording formality that would have had no substance were such a contention before us. [*citing Diablo Canyon* and *Zimmer*]." *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), LBP-83-38, 18 NRC 61, 63 (1983).

The *Susquehanna* case the Staff cites is no authority against our ruling. Either the case involves circumstances crucially different from those which face us, or the *Susquehanna* Board's decision not to take jurisdiction over a pending application for a fuel license is not logical. We quote from the case:

> There is precedent in the Commission's proceedings for Licensing Boards to assume jurisdiction over this application once it is filed, and there seems to be ample justification where the receipt of these unirradiated fuel bundle assemblies and their storage on the refueling floor of the Reactor Building relates closely with one or more contentions. However, inasmuch as the grant of an operating license negates the necessity for [a] Part 70 license, the Board declines to assume jurisdiction of this proceeding at the present time. At present, the Board intends to concentrate on expediting the hearing process on the operating license application.

Slip op. at 29. The first sentence of the quotation affirms our account of the law. From the second sentence it would appear that the Part 70 license was not likely to issue before the operating license. Since an operating license includes the authority to receive and store fuel, the Board sensibly declined to delay the operating license proceeding by a Part 70 proceeding. Our circumstances are not the same. It is clear that the Staff is likely to issue a Part 70 license before we are through with the operating license proceeding; indeed such issuance appears imminent. Moreover, given our ruling on the substance of FOE's contentions and LEA's Petition, our assumption of jurisdiction will not delay the operating license proceedings. Finally, if in *Susquehanna* the issuance of a Part 70 license was imminent, then the second sentence of the quotation from the case would make no sense and we would therefore decline to follow it.

To exercise jurisdiction over LEA's Petition, we must make a somewhat different argument, for LEA did not direct its Petition to us. Though it is unusual for a judicial body to exercise jurisdiction where it is not sought by the petitioner, our exercise here is not an act of Constitutional dimensions. *Cf. Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-726, 17 NRC 755, 757 (1983) (no constitutional dimensions to, and therefore no valid purpose served by, a metaphysical discussion of when jurisdiction passes from a licensing to an appeal board when the important consideration is that a ruling on the substantive issues be made without delay). We simply think it makes sense for us to rule on LEA's Petition, for we know the parties and the circumstances of the case. For this very reason, we would fully expect that were we to decline jurisdiction now and let the Petition follow the path LEA intended it to, we would eventually be the licensing board delegated the responsibility of conducting a hearing on the subject of the Petition. Such has been the common practice, a practice which tends to moot any controversy over our jurisdiction.

Were a separate board delegated responsibility over LEA's Petition, that board would have to duplicate some of the effort we are making here, for some of the findings we shall make against the admissibility of FOE's contentions apply to LEA's Petition. If LEA does not appeal our assertion of jurisdiction, or if it does and we are upheld, then we shall have avoided duplication of effort or an unnecessarily circuitous route to our gaining jurisdiction over LEA's Petition. If LEA appeals but we are not upheld, then either we or another board will be given jurisdiction over the Petition. If we are, then we shall treat LEA's Petition as we do in the remaining sections of this order. If another board is named, it will have the benefit of what we say in those sections. Whatever may happen with any appeal LEA may file from our order here, LEA's interests will have been dealt with fairly.

## WHETHER THE CONTENTIONS AND THE PETITION ARE LATE-FILED

Both the Staff and the Applicant argue that FOE's contentions and LEA's Petition are late-filed. They, therefore, apply the criteria set out in 10 C.F.R. § 2.714(a)(1) for entertaining late filings and find that on balance, the criteria weigh against FOE and LEA. It matters little to either the Staff or the Applicant that the Intervenors did not learn of the fuel license application until just a few weeks ago because the Applicant did not serve any of its fuel license filings on the parties. The Staff is willing to take this fact into account only in relation to one of the five criteria in § 2.714(a)(1), namely whether the intervenor has good cause for failure to file on time. The Applicant, however, is inclined to go not even that little distance: "Under the Licensing Board's decision in *Perry,* [which we cite for support above at pp. 863-64], no 'good cause' exists for FOE's lateness inasmuch as 'it has been apparent that [Applicant] would have to receive unirradiated fuel some time prior to low power testing.' [*Perry,* 18 NRC at 63]." Applicant's Answer at 2. The Applicant claims that the Intervenors should have foreseen even that the Applicant might want to store new fuel outside, since there have been other applicants who have stored fuel outside before operation. Tr. 7813. Therefore, the Applicant argues, FOE should have filed its new fuel contentions and LEA its Petition soon after the August

1981 notice of hearing on the application for an operating license. *Id.* The Applicant goes on to argue that at the very least these filings by the Intervenors should have been made right after the June 1983 application for a fuel license was filed. This is based on the Applicant's view that Part 70 proceedings are not related to Part 50 proceedings, and that therefore the Applicant had no duty to inform the parties of its June application, but the Intervenors did have an affirmative duty to use a nearby Public Document Room (PDR) to keep abreast of the public record. *Id.*[3] "In order to admit the late contentions proposed by FOE [and the Petition filed by LEA], the Licensing Board must find, on balance, that the five factors enumerated in 10 C.F.R. § 2.714(a)(1)(i)-(v) weigh in FOE's [and LEA's] favor. *Duke Power Company* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983)." Applicant's Answer at 3.

We do not think that the Intervenors' Part 70 filings are late. We agree with *Perry* that an intervenor could foresee that an Applicant would have to receive unirradiated fuel some time before low-power testing, and perhaps even that such fuel would have to be outside at the site for some finite time. But we do not think that an intervenor should be expected to foresee that an applicant would request that a fuel license be issued before a low-power operating license, or that the fuel might be stored outside for months, or that there would have to be a security plan tailored to such storage because the normal facility security plan would not be implemented as a prerequisite. We can readily imagine that had any intervenor raised fuel license contentions shortly after the notice of hearing on the operating license, the Applicant would have opposed the admission of those contentions on the grounds that they were premature and speculative. The clock did not start running against the Intervenors when the notice of the hearing for the operating license was issued.

But more important, the clock did not start with the June 1983 fuel license application either. There has been since late 1981 a standing order in this proceeding whose language, unless read in the crabbed way in which the Applicant appears to have read it, requires service on the Board and parties of all filings and correspondence among the parties, particularly between the Applicant and the Staff, on matters related to the operating license proceeding. *See* our Memorandum and Order,

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[3] The Board did not ask, and does not know, whether the June 1983 Part 70 application was in fact filed in the local PDR. Give the Applicant's and technical Staff's lack of notice to the Board and parties it would be logically consistent with such lack of appreciation (at best) of the connection of the Part 70 application to this proceeding for the Applicant and Staff not to have assured filing it in the local PDR. If not, this would negate Applicant's argument that Intervenors should have been aware of the application because of their obligation to periodically search the local PDR. In any event, given the result we reach, it is not necessary to determine whether the Part 70 application was timely filed in the local PDR.

October 14, 1981 (unpublished), at 15-16, and the clarification at Tr. 63-64.

Of course, reasonable persons can disagree about what matters are related to this proceeding, but a fuel license application is surely arguably related. Indeed, we understand *Diablo Canyon* to have settled the argument. We do not have to cite Commission case law to the Applicant to support the well-established basic proposition that correspondence and filings on matters which are at least arguably related to this licensing proceeding should be served on the Board and the other parties. Moreover, Applicant's counsel knew, from his participation in *Zimmer,* that licensing boards would consider a fuel license application to be related to an operating license proceeding.

Our standing order on service of papers on related matters was designed to avoid the very situation we now find ourselves in. Not serving the June 1983 application and its amendment and revised amendment has delayed the Intervenors' response to the application and thus has strengthened, or rather, appeared to strengthen, the Applicant's argument that the Intervenors' responses were late-filed. Had we accepted the Applicant's argument, the Applicant, by merely delaying the service of relevant information, would in effect have tightened the standards for admitting contentions. Thus the circumstance before us is an exception to the Commission's general belief that manipulation of the availability of licensing documents (here the device of limited service contrary to expectations) was unlikely to occur. *See Catawba, supra,* 17 NRC at 1047.

The Staff too, of course, was under an obligation to inform us of the fuel license application. It is true that Staff counsel informed the Board and the Intervenors of the amended application, but by then the Applicant was already in a position to argue that the Intervenors' filings were late. We recognize that the Staff is a large organization and that it is therefore sometimes difficult for Staff counsel to be relevantly informed. However, the Staff appears before us in these proceedings as one body. Counsel should know when its client is considering a Part 70 application. It is long past the time for the Staff to implement elementary procedures to assure that its counsel in a hearing is informed of apparently relevant actions being considered by other elements of the Staff. Indeed, this same problem involving lack of notice to Staff counsel and the Board and other parties of a Part 70 new fuel application being considered by the technical staff arose years ago in *Zimmer, supra.* It is regrettable the Staff has not improved its internal procedures in this simple regard.

In the circumstances of this case, namely, where there was a standing order to serve relevant papers, and particularly where the relevance of the papers at issue had already been determined in Commission case law, we think that the criteria of § 2.714(a)(1) and *Catawba* do not apply. Because of the standing order, the Intervenors had a right to expect that relevant papers would be served upon them. Therefore, they were under no obligation to visit a Public Document Room regularly to find out whether the Applicant had filed an application for a fuel license. Our ruling in no way requires us first to find that the Applicant has willfully ignored our standing order. The ruling depends on the mere fact that the June 1983 application was not served on the Board and parties. Because it is unnecessary to the matter before us, and would be digressive, we do not go so far as to investigate whether the Applicant has behaved willfully in an attempt to shield its Part 70 application from challenge by Intervenors and scrutiny in a hearing. We do find that Applicant has given our standing order a crabbed interpretation which we find to be incorrect.

## WHETHER THE CONTENTIONS AND THE PETITION HAVE BASES

Our desire in dealing with these Part 70 filings has been above all to rule on their substance — to deal with the safety concerns they embody. But because of the way in which these Part 70 issues have come before us, we have had to consider procedural questions at some length. FOE's lay representative, Mr. Anthony, allowing himself to be misled by the mere length of the procedural part of our oral ruling, rather than informed by its content, amazed us by claiming that we had based our decision on legal technicalities as to whether we had received certain documents or not, rather than on the safety issues, and that we were "shirking" our duty by not dealing with those issues. Tr. 8028-29.

We hope that Mr. Anthony will make a modest effort to understand what should be obvious in this written confirmation: that we have decided all the procedural issues in his favor. Had the Applicant and the Staff prevailed on the procedural issues, had we ruled that we had no jurisdiction over Part 70 matters, or that the Intervenors' filings were late and did not pass muster under § 2.714(a)(1) and *Catawba,* FOE's contentions would have been ruled inadmissible with hardly a mention of the substance of those contentions. To reach that substance, we first have had to surmount procedural roadblocks. True, we ultimately rule against FOE and LEA, but not because we haven't faced the safety issues. Rather, it is precisely because we have faced those issues —

more thoroughly than Mr. Anthony has — that we rule as we do. We now turn to the substance of FOE's contentions and LEA's Petition. We discuss the contentions first.

## FOE'S CONTENTIONS

FOE's contentions as they appear in Mr. Anthony's February 23 and 28 filings may be put thus: The presence of unirradiated fuel at the Limerick site would endanger the health and safety of the public because: the Board has yet to determine in litigation over FOE's Contentions V-3a and V-3b whether the safety-related buildings at the site can withstand overpressures and impacts from offsite pipeline accidents; the Independent Design Verification Program the Staff has requested the Applicant to make (*see* January 10, 1984 letter from Eisenhut to Bauer) is not yet complete and so it is not known yet that the buildings at the site are safe to store fuel in; the Staff has recently "raised questions about the qualification of the Limerick overhead cranes for handling nuclear fuel since they do not have the required load safety factor" (*see* February 6, 1984 letter from Schwencer to Bauer); offsite emergency plans are not complete and so cannot provide for the safety of the public "in the event of an accident which could set off a fission process while fuel is being transported to or brought into the plant;" stored outside the fuel will be subject to natural hazards such as tornadoes and electrical storms and to the man-made hazards of theft and sabotage, since PECo does not have adequate safeguards for the new fuel; and the fuel is to be stored in a place which is only 350 to 400 feet from the hypothesized site of the design-basis TNT railway car explosion, and which exposes the fuel to possible "activation" by an accident involving the electrical lines running both over and under the storage site.

In responding to these contentions, the Staff and the Applicant have focused primarily on Mr. Anthony's occasional lack of either clarity, specificity, or understanding of the Commission's regulations, and on the absence in the filings of any explanations as to how unirradiated fuel stored as the application says it will be stored could become critical or otherwise endanger the health and safety of the public. For example, Mr. Anthony tends to rely on all the talk of criticality in the fuel license application as evidence that the stored new fuel is dangerous, but we fail to see how the application's explanations of why the fuel will not go critical add up to evidence of danger. *See, e.g.*, FOE's February 23 filing, item 2, and Tr. 7880.

But rather than emphasize that Mr. Anthony has not given us any credible explanation of how unirradiated fuel can cause the public harm, we would emphasize that there exists no such explanation. First, simply on the basis of our own collective knowledge, we are willing here to rule that it is not credible to claim that unirradiated fuel stored as the Limerick fuel will be stored can go critical. The Applicant is right that we are not here to litigate the laws of physics (Tr. 7875), and so we shall not explain why the stored unirradiated fuel at Limerick will not go critical. But the Staff has been helpful by pointing to a few pages in an Appeal Board decision in the *Diablo Canyon* proceeding, *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-334, 3 NRC 809, 817-20 (1976), a few pages which, with brevity and clarity, set out the several conditions which must be met before unirradiated fuel can go critical. We have asked the Staff to provide Mr. Anthony with that Appeal Board decision, not as something to which Mr. Anthony is invited to reply but simply as something through which he can enlighten himself.

As to whether stored new fuel at Limerick can endanger the health and safety of the public through some means not involving criticality: Mr. Anthony has suggested that if a plane crashed into the fuel, or a tower collapsed on it, or the non-conforming crane dropped it, or dropped a heavy object on it, uranium oxide dust could be released to the community and present a "really live public hazard," Tr. 7908-09. Mr. Anthony cites the Supreme Court's recent decision in the Karen Silkwood litigation, *Silkwood v. Kerr-McGee Corp.*, 104 S. Ct. 615 (1984), as one piece of evidence of the harm nuclear fuel can cause. Tr. 7879. But the *Silkwood* decision is no support for Mr. Anthony's contentions: Karen Silkwood was apparently contaminated as a result of handling plutonium, a known alpha-radioactive toxic substance. But unirradiated uranium oxide pellets do not contain any plutonium. Dose rates one foot away from such pellets are negligible. Further, uranium oxide pellets are made of ceramic material that would not become dust under severe impacts. Again we rule on the basis of what we as a Board know: We do not find it credible that non-criticality accidents involving low-enriched, unirradiated uranium oxide fuel could threaten the health and safety of the public.

However, to make sure that this aspect of our ruling that FOE's contentions lack reasonable bases is well-founded, we have ordered the Staff and the Applicant to submit by March 14 affidavits which address whether there is a credible potential that some non-criticality accident involving the low-enriched uranium oxide fuel pellets in the unirradiated new fuel rods for Limerick could cause a violation of the regulations concerning onsite and offsite releases of radiation. Tr. 7920-21. FOE was invited to reply to the affidavits if it wishes to, and LEA too, though the affidavits will concern FOE's contentions more than they will LEA's Petition. Any reply the Intervenors choose to make must be in our hands in Philadelphia by March 21. Of course, the best, but not the required, form of a reply would be an affidavit from an expert. We have received the affidavits from the Staff and the Applicant, and Mr. Anthony informed us yesterday that he intends to reply.

Mr. Anthony often returned to the question of the non-conforming cranes during the prehearing conference. *See, e.g.*, Tr. 7859, 8029, 8033, and 8180. Our ruling on Mr. Anthony's contentions in no way depends on whether the non-conforming crane could somehow crush new fuel. Even if it could, the health and safety of the public would not be endangered. But we shall devote a paragraph to the crane, in part because of Mr. Anthony's insistent concern about the crane, and in part to clarify the discussions of the crane in the February 6, 1984 Staff letter from Schwencer to Bauer and in Section 9.1.5 of the Limerick SER.

It appears to be highly unlikely that the non-conforming crane could cause any damage to the unirradiated new fuel. The crane is non-conforming in a very particular respect: It is not the crane itself which the Staff finds to be non-conforming; rather certain lifting devices which are attached to the crane when certain loads are lifted are non-conforming — and not generally so, but only in relation to the heavy weights of the refueling shield (100,000 lbs.) and fuel pool stop logs (120,000 lbs.). NRC regulations require that these lifting devices be able to lift 10 times the weight either of the refueling shield or the stop logs, but the particular devices in question have been found to be able to lift only 4.8 times the weight of the refueling shield and 9.3 times the weight of the stop logs. The Staff is requiring that this non-conformance be resolved before the second refueling outage is over. The Staff does not question that the crane is fit to lift the much lighter new fuel shipping containers (1900 lbs.), or even a number of the containers together. Tr. 8034-35. Moreover, the Applicant says that a combination of physical interlocks on the crane and administrative controls on its use will prevent the crane from lifting heavy loads over places where new fuel will be stored. Tr. 7886.

Between our ruling here on criticality and our tentative ruling on non-criticality accidents with the new fuel, we have addressed and found inadmissible all of FOE's contentions except the one on theft and sabotage. We address it below at p. 874. We shall make a final ruling on FOE's contentions promptly after March 21, the day the replies to the affidavits are due. We think it likely that our ruling will issue before new fuel is received at Limerick; therefore we shall not now issue a stay of fuel receipt at Limerick. Immediately after we had made the oral ruling this order is confirming, Mr. Anthony orally requested a stay. Tr. 7922. We denied the request, on the grounds that stays are not to be issued unless they are necessary. But we will entertain motions for a stay if it is clear that the Applicant will receive fuel before we rule on the affidavits and the replies. We orally ruled, and hereby confirm, that the Applicant must inform the Board and the parties three business days in advance of the time it begins to receive fuel that it plans to do so. Tr. 8045-46. We add now that the notification requirement is extended to include the *Limerick* Appeal Board and the Commission due to the pendency of FOE's appeal and the possibility of appeal by LEA.

If, as expected, our final ruling on the affidavits is consistent with our ruling at this time, there will then be nothing pending before us which would prevent receipt of the fuel pursuant to any license which the Staff may issue. (As of this writing, the Staff has not issued the requested Part 70 license amendment.) Upon our final ruling, which may be made orally on the record, the requirement of three business days' advance notice will remain in effect, unless and until removed by an appellate tribunal.

It would not be easy for us to say that by late March Mr. Anthony will be satisfied that our account of the health and safety impact of new fuel at Limerick is sound. During the prehearing conference in which we made the oral ruling we are now confirming, shortly after we made the ruling, Mr. Anthony asserted that we were going "to hide behind a technical affidavit" (Tr. 8028), that "there will be eventual operation of this plant. It is an inexorable process that appears to me to start with your decision" (Tr. 8030), and that he could not understand how the Applicant would be permitted to use a crane the Staff had judged to be non-conforming. Tr. 8033.

As to the crane, we simply point out what we think was clearly enough said during the Part 70 discussions the week of March 5: The crane in question is non-conforming in a very particular respect: It may not be used with certain lifting devices to lift certain loads. It is not in some vague sense generally non-conforming. Some drivers are licensed to drive only during the day. They are not thereby to be declared non-conforming when driving during the day. We fail to comprehend why Mr. Anthony continues to press his misunderstanding on us, especially when whether the crane is conforming has no logical relation to our decision.

As to whether we plan to hide behind the affidavits, we do not. Neither do we plan to hide behind the replies, should they reveal that what we rule here is wrong. We plan to read the affidavits and the replies, think about them, and rule. As to whether this ruling we confirm today begins an inexorable process which leads to the operation of the Limerick plant, we note that other boards have heard and thoughtfully rejected similar arguments. *See, e.g., Perry, supra,* 18 NRC at 65-66 (no logical relationship of Part 70 ruling to grant or denial of operating license). We plan to continue to hear the litigation of FOE's admitted contentions in this operating license proceeding and then rule according to the merit the record accords them.

## LEA'S PETITION

Some of the analysis which we made concerning the technical bases of FOE's contentions will apply to the technical bases of LEA's Petition, but before we can apply that analysis we must clear yet another procedural hurdle. This hurdle is one more appropriately dealt with now than earlier, for where the earlier hurdles concerning jurisdiction and criteria for admitting late-filed contentions threatened to keep us from considering the substance of the Intervenors' Part 70 filings, the remaining hurdle is that LEA's Petition appears to have no substance to consider: It is not a set of contentions; it is only a request for a hearing, and that is all LEA intended it to be. As the heading of the Petition shows, LEA requested the hearing under the Atomic Energy Act, as amended January 4, 1983, Pub. L. 97-415, Section 12(a). The relevant part of that section is paraphrased in 10 C.F.R. § 50.91(a)(4), which implements Section 12(a). We quote § 50.91(a)(4):

> Where the Commission makes a final determination that no significant hazards consideration is involved and that the amendment should be issued, the amendment will be effective upon issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in § 2.714 has filed a request for a hearing. The Commission need hold any required hearing only after it issues an amendment, unless it determines that a significant hazards consideration is involved.

The Board and LEA agree that, as the first paragraph of § 50.91 shows, the amendment § 50.91(a)(4) refers to is an amendment to an operating license. But the Board and LEA do not agree about how § 50.91(a)(4) applies to amendments to Part 70 licenses; therefore, the Board and LEA do not agree about how § 50.91(a)(4) applies to the Applicants' revised amended fuel license application, which, in effect, is an application for an amendment to the Part 70 license the Applicant has had since last September to receive milligram quantities of sealed source material. LEA reads the quoted section to deny the Commission the power to make amendments effective upon issuance before hearings are held, except when the amendment is both to an operating license and involves no significant hazards consideration. LEA, therefore, regards its Petition as having stayed the effectiveness of any amendment to the Applicant's existing Part 70 license until either the request for a hearing is denied or the hearing is held. *See, e.g.,* Tr. 7840-41, and Tr. 7894-95. Therefore, under LEA's reading of § 50.91(a)(4), LEA has no obligation to file contentions on the new fuel license until well after the Petition has been granted, and we have no substance to consider in supporting a ruling on the Petition.

The advantages from LEA's perspective of its way of proceeding on the fuel license application are clear: By requesting a separate hearing LEA hopes both to gain time to draft admissible contentions and to keep fuel away from Limerick at least until the separate hearing is over. Had LEA brought its concerns before us in the form of a motion to stay receipt of fuel, it would have had to argue a great deal more than it has to argue when petitioning for a hearing.

Treated as a motion to stay receipt of fuel, LEA's Petition must be denied, for it offers neither contentions with some chance of being admitted, nor any strong indication that such contentions are forthcoming. During the prehearing conference the week of March 5, we asked LEA to indicate what matters it could foresee filing contentions on and it mentioned only two: the physical security plan for the new fuel, and the Applicant's request under 10 C.F.R. § 70.24(d) for an exemption from the requirement of § 70.24 for a criticality monitoring system. Tr. 7895. *See* Sec. 2.2.6.1 in June 1983 fuel license application (repeated in the February 6, 1984 amended application).

We assume that the contention on criticality monitoring would allege that such monitoring was to be required of the Applicant. Given the rulings we have made on FOE's contentions related to criticality, we cannot imagine that we would admit such a contention. Moreover, exemptions from the requirements of § 70.24 for low-enrichment new fuel stored at a nuclear power plant site are commonly granted.

Neither can we easily imagine admitting a contention on the physical security plan for the new fuel. Such a security plan is not a complex affair, and, therefore, there is no general cause to be concerned about its adequacy. Since FOE's contention alleges no particular cause to be concerned about the adequacy of the plan, we rule that contention inadmissible for lack of basis and specificity. As LEA firmly points out, any admissible contention concerning the plan would have to be based on a knowledge of the plan, and LEA has not had an opportunity to see it. Tr. 7927-28.

We are not opposed to any sensible arrangement LEA and the Applicant can fashion to give LEA access to the plan, and we will entertain filings of contentions based on knowledge of the plan; but we will not hold up Staff action on the fuel license application while LEA examines the plan seeking bases and specificity for contentions. Our ruling would probably be different if a security plan for unirradiated fuel were as complex as an emergency plan for operating licenses and thus presented as great a possibility of inadequacy, and if LEA had given us a reasonable indication of the specific aspects of the plan with which it is concerned, as LEA did with the emergency plans even before it had seen the plans.

If, after what we think would be an unnecessarily circuitous route, we were delegated jurisdiction over LEA's Petition, we would be faced with LEA's argument that under § 50.91(a)(4), its request for a hearing stays the effectiveness of any Staff order granting the Applicant a fuel license. Again, we would consider any contentions LEA wished to file, but we would not regard the effectiveness of a granted fuel license as stayed, for we do not interpret § 50.91(a)(4)'s silence about licenses other than operating licenses the way LEA does. LEA reads the section to imply a limitation on the power of the Commission to make amendments effective upon issuance: that the Commission has a right to make immediately effective amendments only to operating licenses and only on a showing of no significant hazards considerations. We, however, read the section to imply a limitation on the right of an intervenor to have the effectiveness of an amendment stayed until the completion of the requested hearing. We think it is the purpose of that regulation, and the statutory section behind it, to say that only in relation to operating licenses will amendments ever raise safety issues significant enough to require that the effectiveness of the amendment be stayed pending a hearing. Thus, on our reading of the regulation, amendments to Part 70 licenses normally do not raise safety issues significant enough to require that the effectiveness of the amendment be stayed pending a hearing.

We can imagine coming a step closer to LEA's reading of § 50.91 (which is applicable to operating license amendments) by considering, *arguendo,* that it permits an analogous treatment of amendments to fuel licenses. Read thus, the regulation would permit immediately effective amendments to Part 70 licenses even where a hearing has been requested, where there is a no significant hazards considerations finding. 10 C.F.R. §§ 50.19(a)(4) and 50.92(c). Therefore, even under this analogous application of § 50.91(a)(4) for the sake of argument, our ruling would stand, for LEA has hardly begun to argue that this amendment involves any significant hazards considerations. Given our discussion above, we conclude there are none as to concerns raised by LEA, and that, subject to the expected confirmation by affidavits, there are none as to FOE's contentions.

## APPEALABILITY

If confirmed by our ruling on the affidavits, this will become a final order with respect to all the Part 70 matters raised by FOE. As such, under the Commission's ruling in *Diablo Canyon, supra,* this order will become appealable upon confirmation, after our consideration of the affidavits and any replies. 3 NRC at 74. It is arguably appealable now. FOE has already filed appeals with both the Appeal Board and the Commission.

With respect to LEA's filing, we consider this order to be final now, for LEA has not specified any concern to which a ruling on the affidavits and replies might speak. As we said in our oral ruling, with respect to LEA the ruling became appealable when it was delivered orally, but the deadline for LEA's filing an appeal is to be calculated from the date of service of this written confirmation. By letter of March 9, 1984, LEA informed us that it has not yet decided whether to appeal, and that its decision will depend, in part, on whether it and the Applicant succeed in their present efforts to come to a mutually satisfactory arrangement for giving LEA access to the security plan.

Appeals should be directed to the Commission rather than the Appeal Board, for under the Commission's reading of 10 C.F.R. § 2.785, the Appeal Board does not have jurisdiction over Part 70 matters without a specific delegation from the Commission. *Diablo Canyon, supra,* 3 NRC at 74 n.1. There has, as yet, been no such delegation. As a courtesy, copies of appellate papers should also be served on the *Limerick* Appeal Board and this Board.

The Commission's Office of General Counsel and the Appeal Board have been advised of this ruling.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Lawrence Brenner, Chairman
ADMINISTRATIVE JUDGE

Dr. Peter A. Morris (By LB)
ADMINISTRATIVE JUDGE

Bethesda, Maryland
March 16, 1984

Judge Cole was unavailable to review this written order, but he participated and concurred in the oral rulings which this order confirms.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**ATOMIC SAFETY AND LICENSING BOARD**

**Before Administrative Judges:**

**Sheldon J. Wolfe, Chairman**
**Dr. George C. Anderson**
**Dr. Hugh C. Paxton**

**In the Matter of** **Docket No. 50-482**
**(ASLBP No. 81-453-03-OL)**

**KANSAS GAS & ELECTRIC COMPANY, *et al.***
**(Wolf Creek Generating Station, Unit No. 1)** **March 26, 1984**

The Licensing Board denies an admittedly untimely petition for leave to intervene filed during the course of a hearing which was being held to consider the sole controverted issue of emergency planning. After balancing the factors set forth in 10 C.F.R. § 2.714(a)(1), the Board concluded that the petition, seeking to raise quality assurance/quality control matters, should not be granted.

## RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

In order to determine whether an untimely petition for leave to intervene should be allowed, the Board must balance the five factors set forth in 10 C.F.R. § 2.714(a)(1).

RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

"Good cause" for a late filing depends wholly upon the substantiality of the reasons assigned for not having filed at an earlier date. *South Carolina Electric and Gas Co.* (Virgil C. Summer Nuclear Station, Unit 1), ALAB-642, 13 NRC 881, 887 n.5 (1981).

RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

If the controlling facts relating to the excuse for the untimely filing are not controverted by the petitioner's affidavits, the Board must take them as true. *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-420, 6 NRC 8, 13 (1977), *aff'd,* CLI-78-12, 7 NRC 939 (1978).

RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

Petitioners for leave to intervene, as well as intervenors, are required to diligently uncover and apply all publicly available information. *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041, 1048 (1983); *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), LBP-83-42, 18 NRC 112, 117, *aff'd,* ALAB-743, 18 NRC 387 (1983).

RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

If it is the petitioner's position that its newly acquired organizational existence was sufficient to justify belated intervention, such an explanation for the tardy filing cannot carry the day because the necessary consequence would be that parties to the proceeding would never be determined with certainty until the final curtain fell. No adjudicatory process could be conducted in an orderly and expeditious manner if subjected to such a handicap. *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant, Units 1-4), ALAB-526, 9 NRC 122, 124 (1979).

## RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

Where no good excuse is tendered for the tardy filing, the petitioner's demonstration on the four other factors in 10 C.F.R. § 2.714(a)(1) must be particularly strong. *Mississippi Power & Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-704, 16 NRC 1725, 1730 (1982); *Duke Power Co.* (Perkins Nuclear Station, Units 1, 2 and 3), ALAB-431, 6 NRC 460, 462 (1977).

## RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

The second and fourth factors in 10 C.F.R. § 2.714(a)(1) are of relatively minor importance in the weighing process. *Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-707, 16 NRC 1760, 1767 (1982).

## RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

It is the petitioner's ability to contribute sound evidence — rather than asserted legal skills — that is of significance in considering a late-filed petition to intervene. *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-671, 15 NRC 508, 513 n.14 (1982).

## RULES OF PRACTICE: NONTIMELY PETITION TO INTERVENE

Even though we are told that four of its co-counsel actively participated in the construction hearings, we cannot conclude that the petitioner's participation could reasonably be expected to assist in developing a sound record since the issue that it would litigate here bears no resemblance to any contested issue that confronted the Licensing Board in the construction permit proceeding. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-743, 18 NRC 387, 401 (1983).

# MEMORANDUM AND ORDER
**(Denying NAN's Petition for Leave to Intervene)**

## Memorandum

### I. BACKGROUND

On December 18, 1980, the Nuclear Regulatory Commission published a Notice of Opportunity for Hearing with respect to Applicants' application for the issuance of an operating license (45 Fed. Reg. 83,360). This Notice provided that, by January 19, 1981, "any person whose interest may be affected by this proceeding may file a petition for leave to intervene." The § 2.751a special prehearing conference was held on April 15, 1981. Ultimately, after the completion of discovery, hearings were held January 17-21, January 23-26, and February 14-16, 1984. During these hearings evidence was adduced upon the Intervenors' (Ms. Wanda Christy and Ms. Mary Ellen Salava) emergency planning contentions. There were no other contested matters.

Meantime, in a letter to the Board of January 5, 1984, Ms. Mary Stephens, director of an organization known as Nuclear Awareness Network of Lawrence, Kansas, advised that they intended to submit a late-filed petition to intervene as quickly as possible with respect to quality control/quality assurance. On January 19, 1984, Ms. Stephens, asserting that she had been authorized to represent the members of Nuclear Awareness Network, Inc. (NAN), filed an admittedly untimely Petition for Leave to Intervene and Request for Hearing.

On the last day of the hearing, February 16th, the Board formally closed the record and directed the parties, including the Federal Emergency Management Agency, to submit proposed findings of fact, conclusions of law, briefs and a proposed form of order or decision. However, the Board stated that the closing was conditional and that the record might be reopened because of two recent events. First, the Board noted that, in *New England Coalition on Nuclear Pollution v. NRC,* 727 F.2d 1127 (D.C. Cir. 1984), the Court of Appeals had granted the petition which challenged the Commission's rule to eliminate the need for applicants, which are electric utilities, to establish their financial qualifications. The Board observed that a former party-intervenor in the instant case, Kansans for Sensible Energy, had appealed its dismissal as a party by this Licensing Board, but that the Commission's Appeal Board had held its decision in abeyance pending the result of the District of Columbia Court of Appeal's decision.[1] Second, the Board noted that to date it had not acted upon NAN's untimely petition for leave to intervene.

On February 3 and February 8, 1984, Applicants and the Staff respectively filed responses opposing the granting of NAN's petition for leave to intervene. Pursuant to an unpublished Order of February 9, 1984, NAN, represented by its attorney, filed a response on March 6, 1984. Applicants' and Staff's counsel advised that they did not wish to respond to NAN's response of March 6th and rested upon their submissions respectively submitted on February 3rd and February 8th (*see* unpublished Memorandum, dated March 13, 1984).

## II. DISCUSSION[2]

As it must, having filed its petition for leave to intervene exactly 3 years after the opportunity for timely intervention had expired, NAN admits that the petition is untimely. (Petition at 1). NAN asserts that it is a nonprofit Kansas corporation established in 1983 "for the express purpose of providing education, research, lobbying, and testimony on issues relating to nuclear power, waste, and related matters," that two identified members live within 20 miles of the Wolf Creek construction site,[3] and that other members live, work and recreate within the Wolf Creek geographic area and have interests that may be affected by the outcome of the proceeding. (Petition at 1-2). Petitioner further alleges that within the last 30 days (prior to January 19, 1984) six former Wolf Creek workers voluntarily made statements to its director, Ms. Mary Stephens, which "strongly suggest" that the Applicants' general contractor has encouraged or permitted procedures and practices contrary to the conditions in the Safety Evaluation Report and applicable federal regulations. Should the Board permit intervention, it asserts that, based upon these statements, it would timely file contentions alleging:

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[1] In a recent policy statement, the Commission directed its adjudicatory bodies to continue to treat the rule as valid and stated that it expected to complete an adequate response to the D.C. Circuit's decision before the Court issues its mandate. 49 Fed. Reg. 7981 (1984). Because of this directive, in a Memorandum and Order of February 28, 1984 (unpublished), the Appeal Board stated that KASE's appeal would continue to be held in a deferred status.

[2] In light of our ultimate conclusion that the petitioner should not be allowed to enter the proceeding at this late date, it is unnecessary for us to reach and decide whether NAN has standing to intervene as a matter of right or, lacking standing, whether it meets the criteria established for allowing intervention as a matter of discretion.

[3] Petitioner attached to its Response of March 6, 1984, affidavits of these two members attesting that they had authorized NAN's director to file the petition on their behalves. It also attached its director's affidavit which states, *inter alia*, that she travels and recreates within 20 to 25 miles of the nuclear facility.

(a) That the deliberate policies practiced and permitted by Daniels Construction Co. as general contractor at Wolf Creek are contrary to and make mockery of quality assurance/quality control requirements putatively imposed on this project;

(b) That construction workers were directed by Daniels' foremen to perform work in safety-related areas at variance with established procedures creating doubt as to the physical soundness of the structure;

(c) That Daniels' foremen directed construction workers to mislead quality control personnel and at least one Daniels' foreman forged and falsified work documents for safety-related matters.

NAN states that, in support of its contentions relating to the breakdown of quality assurance/quality control, it proffers the testimony of these six former employees of the general contractor. It then proceeds to identify these six individuals and summarizes their statements alleging incidences of improper and/or defective quality assurance/quality control. (Petition at 3-7).

In order to determine whether this untimely petition for leave to intervene should be allowed, we must balance the five factors set forth in 10 C.F.R. § 2.714(a)(1). In pertinent part, this section provides that:

> Any person whose interest may be affected by a proceeding and who desires to participate as a party shall file a written petition for leave to intervene. . . . The petition and/or request shall be filed not later than the time specified in the notice of hearing . . . . Nontimely filings will not be entertained absent a determination . . . that the petition and/or request should be granted based upon a balancing of the following factors . . . :
>
> (i) Good cause, if any, for failure to file on time.
> (ii) The availability of other means whereby the petitioner's interest will be protected.
> (iii) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.
> (iv) The extent to which the petitioner's interest will be represented by existing parties.
> (v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding.

### (i) Good cause, if any, for failure to file on time

Much argument was occasioned by NAN's assertion at page 7 of the petition that it was unaware of the existence of these serious allegations until mid-December 1983, when its director was contacted by a representative of the workers, who was not identified by NAN. At page 8 of the petition, however, NAN avers that the construction workers first approached its director in December of 1983. Since these allegations of safety-related violations were not available until these individuals decided to make public this evidence, NAN urges that these allegations constituted newly arising information and good cause has been established for its failure to file on time.

Applicants, however, attached to their Response of February 3, 1984, a joint affidavit of their QA manager and of their QA management consultant who attest that they had interviewed the six former workers identified by NAN, and that each worker had stated, among other things, that NAN's director had contacted them and that they had not initiated the communication with Ms. Stephens. Applicants' two QA personnel also attest that (1) each of the six former workers were sheet metal craftsmen who had worked on heating, ventilating and air-conditioning systems and that each had indicated his first-hand knowledge of quality problems at Wolf Creek was limited to HVAC systems, and that (2) each worker had either denied some of the allegations attributed to them by NAN, corrected or revised allegations attributed to them by NAN, and/or advised that most, if not all, corrective actions had been taken and that they have no present safety concerns. Signed but unsworn statements of four of these former workers, as witnessed by the Applicants' QA manager, were also attached to the Response. (Attachments 2-5). Accordingly, Applicants argue that since NAN's recounts of allegations are in large part gross misrepresentations and since NAN initiated contact with the workers (and thus could have done so long ago), good cause has not been established to excuse the belated filing of the petition.[4] (Applicants' Response at 15). Unaccountably, Applicants did not secure and attach affidavits of the six former workers. Four of these individuals signed unsworn statements; apparently the two remaining individuals declined. Equally inexplicable is the fact that, ignoring its assertion at page 8 of its petition, NAN urges that "[n]owhere does Petitioner state or imply that the workers contacted" its director. (NAN's Response at 3). Moreover, we note that while NAN asserts that "[i]t is certainly true that NAN Director Stephens contacted the referenced six workers but only after and in response to her being contacted by their official representative" (*id.* at 7), Ms. Stephens' affidavit neither addressed nor supported this factual allegation. Finally, we note that, again without any explanation for not doing so, NAN did not reinterview any of the six former workers after receiving Applicants'

[4] Relying upon this information received from the Applicants that NAN's director had initiated the contacts with the six former workers and thus could have made these inquiries earlier, the Staff was unwilling to rely upon NAN's primary justification for its untimely filing. (Staff's Response at 10).

Response of February 3, 1984, did not append to its Response counter-statements or affidavits of four of these six individuals if indeed their statements appended to Applicants' Response were faulty, and/or did not append counter-statements or affidavits of all six individuals, if the joint affidavit of Applicants' QA manager and QA management consultant misstated information or drew erroneous conclusions from the interviews with the six former workers. Instead, at page 4 of its Response, NAN attacks the credibility of Applicants' QA manager and QA management consultant in that it is their "QA/QC program [which] is directly challenged by Petitioner."

Amidst this swirl of arguments, however, there are undisputed facts. First, at page 12 of the joint affidavit, Applicants' QA manager and QA management consultant attest that Mr. Neil Campbell's allegation of forgery of weld control records by a Daniels Construction Company sheet metal foreman had been the subject of I&E Report 81-10.[5] Exhibit C, attached to Applicants' Response, contains a Notice of Violation dated April 21, 1982, and Investigation Report 81-10, the latter of which, as signed on September 22, 1981, reflects that: (a) the investigation of this allegation involved 66 hours by two NRC investigators and two NRC inspectors in May, June and August of 1981; (b) the investigation identified one Weld Control Record Supplement Sheet which contained nine QC inspector signatures suspected to be forgeries; (c) an FBI laboratory analysis confirmed that these signatures were forgeries; and (d) efforts to identify the person responsible for these forged signatures were unproductive. Exhibit E reflects that Applicants advised that corrective actions were completed on April 16, 1982, and Exhibit I, dated April 12, 1983, reflects that the Commission closed out the violation in I&E Report 83-06. Second, at page 11 of the joint affidavit, Applicants' two QA personnel attest that Mr. Campbell's allegation that he had been repeatedly ordered to stamp false D numbers on welds had been the subject of I&E Report 81-12. Exhibit D, attached to Applicants' Response, contains a Notice of Violation dated April 21, 1982, and Investigation Report 81-12, the latter of which, as signed on September 22, 1981, reflects that the investigation of this allegation, among others, by two NRC investigators and one NRC inspector took place in June, July and August 1981, and that interviews regarding changing of welder identification numbers on HVAC hangars confirmed that this had occurred and that no justification for these actions could be provided. Ex-

[5] Mr. Campbell is one of the six former workers at Wolf Creek identified in NAN's Petition and is alleged to have made these statements to NAN's director. (Petition at 5). We have given no weight to hearsay statements in the joint affidavit alleged to have been made by him since Applicants failed to submit Mr. Campbell's affidavit, and did not even furnish his signed but unsworn statement.

hibit F, dated May 21, 1982, reflected that Applicants advised NRC that corrective action would be taken. Exhibit I, dated April 12, 1983, reflects that the NRC inspector reported that, as a result of the number of discrepancies disclosed on 120 HVAC hangars/supports, all safety-related hangar supports were 100 percent reinspected and that necessary rework had been completed on January 20, 1983. Third, said QA personnel attested at page 13 of the joint affidavit that I&E Reports 81-10 and 81-12 had been placed in the local NRC Public Document Room in May 1982, and that, as reflected in attached Exhibit L, newspaper articles of April 23, 1981, of April 30, 1982, and of May 3 and May 5, 1982 had addressed these violations and investigations.

Relying on these facts, not disputed by NAN in its subsequent response, Applicants argue that NAN has failed to show good cause because "Petitioner does not explain why it did not seek out workers or other sources of QA/QC information years ago on the basis of available information." (Applicants' Response at 15). NAN's rejoinder at page 4 of its Response is that:

> Applicant, having at this point abandoned common sense, would have this Board craft an impossible standard. Applicant's argument would, if adopted, require prospective intervenors to not only scour newspaper accounts and voluminous NRC-required filings, but to conduct daily exit-interviews at the construction site. Only in that way could Petitioner have learned of the complained-of work on safety-related plant . . .

We are guided by the Appeal Board which has held that whether there is "good cause" for a late filing depends wholly upon the substantiality of the reasons assigned for not having filed at an earlier date.[6] However, here not only are the controlling facts uncontroverted by NAN's affidavits and thus ones which we must take as true,[7] but NAN concedes that it could have learned of QA/QC problems by looking at newspaper accounts and the submissions required by the NRC. At least as early as April 1981, via the newspaper account, and by no later than May 1982, via placements of the Inspection and Enforcement Reports in the local public document room, NAN knew or should have known of these problems. Instead of acting promptly, NAN waited until January 19, 1984 before filing its petition for leave to intervene out-of-time. Intervenors are required to diligently uncover and apply all publicly available

[6] *South Carolina Electric and Gas Co.* (Virgil C. Summer Nuclear Station, Unit 1), ALAB-642, 13 NRC 881, 887 n.5 (1981).

[7] *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-420, 6 NRC 8, 13 (1977), *aff'd*, CLI-78-12, 7 NRC 939 (1978).

information.[8] Moreover, if it is NAN's position that its newly acquired organizational existence in 1983 was sufficient to justify belated intervention, such an explanation for the tardy filing cannot carry the day because the necessary consequence would be that the parties to the proceeding would never be determined with certainty until the final curtain fell. Assuredly, no adjudicatory process could be conducted in an orderly and expeditious manner if subjected to such a handicap.[9]

We conclude that NAN's tardiness was unjustified, and, in these circumstances where no good excuse is tendered, the petitioner's demonstration on the four other factors must be particularly strong.[10]

**(ii) and (iv) The availability of other means whereby the petitioner's interest will be protected, and the extent to which the petitioner's interest will be represented by existing parties**

It is clear that these two factors must be weighed in NAN's favor. There is no issue other than the matter of emergency planning which has been litigated in this proceeding. However, these two factors are of relatively minor importance.[11]

**(iii) The extent to which petitioner's participation may reasonably be expected to assist in developing a sound record**

The Petition does not tell us clearly whether NAN intends to present as witnesses the six former workers and, as indicated above, we are concerned that, without explanation, NAN's Response of March 6th did not even advert to the written statements appended to Applicants' Response of February 3, 1984. Moreover, while NAN asserts that its members possess technical expertise in relevant areas and that its counsel is experienced as a former Deputy General Counsel to the Kansas Corporation Commission and that he would be assisted by four co-counsel

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[8] *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041, 1048 (1983). While this and other cases are addressed to nontimely filing of contentions, these decisions of the Commission and the Appeal Board have equal application to nontimely petitions to intervene. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), LBP-83-42, 18 NRC 112, 117, *aff'd*, ALAB-743, 18 NRC 387 (1983).

[9] *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant, Units 1-4), ALAB-526, 9 NRC 122, 124 (1979).

[10] *Mississippi Power & Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-704, 16 NRC 1725, 1730 (1982); *Duke Power Co.* (Perkins Nuclear Station, Units 1, 2 and 3), ALAB-431, 6 NRC 460, 462 (1977).

[11] *Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-707, 16 NRC 1760, 1767 (1982).

whose experience includes, *inter alia,* active participation in the earlier Wolf Creek construction hearings (Petition at 10; Response at 7), we weigh this factor against the Petitioner. We have before us only NAN's conclusional assertions, and thus NAN did not sustain its burden with respect to this factor. It is the ability to contribute sound evidence — rather than asserted legal skills — that is of significance in considering a late-filed petition to intervene.[12] Moreover, the QA/QC issue that NAN would litigate here bears no resemblance to any contested issue that confronted the Licensing Board in the construction permit proceeding.[13] Finally, we are not told that NAN's members are experts in QA/QC matters.

**(v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding**

NAN admits that, if its petition for leave to intervene is granted and its contentions are thereafter admitted, the issues before the Board will necessarily be broadened, the parties will require time to prepare for the hearing and the hearing will likely be extended for a period of months. (Petition at 11). This factor must be weighed against NAN.

## III. CONCLUSION

Because of the lack of availability of other means to protect its interest and because its interest will not be represented by existing parties, factors (ii) and (iv) weigh in NAN's favor. However, these two factors are of relatively minor importance. The important other factors (i, iii and v) must be weighed against NAN and decisively tip the balance against permitting intervention at this late date.

## Order

Nuclear Awareness Network Inc.'s Petition for Leave to Intervene and Request for Hearing is denied. Pursuant to 10 C.F.R. § 2.714a, within ten (10) days after the service of this Memorandum and Order, NAN may appeal to the Atomic Safety and Licensing Appeal Board.

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[12] *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-671, 15 NRC 508, 513 n.14 (1982).

[13] *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-743, 18 NRC 387, 401 (1983).

Judges Anderson and Paxton join but were unavailable to sign this issuance.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Sheldon J. Wolfe, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland,
this 26th day of March 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF NUCLEAR REACTOR REGULATION**

**Harold R. Denton,** Director

**In the Matter of** **Docket No. 50-373**
**(10 C.F.R. § 2.206)**

**COMMONWEALTH EDISON COMPANY**
**(LaSalle County Station,**
**Units 1 and 2)**

**AND ALL LIGHT-WATER REACTORS** **March 16, 1984**

The Director of the Office of Nuclear Reactor Regulation denies petitions by Edward M. Gogol alleging that there are severe errors, defects and loopholes in the integrated leak rate testing (ILRT) methodology now in use. The petitions sought a variety of relief including requests for immediate action such as placing the LaSalle Unit 1 of the Commonwealth Edison Company in cold shutdown, ceasing further construction and licensing activities with respect to LaSalle Unit 2 and Byron Unit 1 and shutting down reactors with insufficient evidence of adequate containment leak rate testing.

## NUCLEAR REGULATORY COMMISSION: RULEMAKING AUTHORITY

Should a petitioner pursuant to 10 C.F.R. § 2.206 wish to initiate a rulemaking, the procedures set forth in 10 C.F.R. § 2.802 should be followed.

## RULES OF PRACTICE: SHOW-CAUSE PROCEEDINGS

The Director will not institute proceedings in response to a petition under 10 C.F.R. § 2.206 to consider an issue the Commission is treating generically through rulemaking.

## TECHNICAL ISSUE DISCUSSED: CONTAINMENT LEAK RATE TESTING

The Commission's requirements for integrated leak rate testing are set out in 10 C.F.R. § 50.54(o) and Appendix J to 10 C.F.R. Part 50. While the Commission's requirements for integrated leak rate testing continue to provide reasonable assurance that the public health and safety is adequately protected, the NRC Staff has under way a review of leak rate testing requirements to see whether modifications to these requirements are appropriate. The Commission has placed leak rate testing for water-cooled power reactors on its Regulatory Agenda.

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

## INTRODUCTION

On November 29, 1983, Edward M. Gogol (Petitioner) filed a "Petition for Emergency Relief re: Primary Containment Leakage and Integrated Leak Rate Testing at All U.S. Nuclear Power Reactors" with the Commission. A virtually identical document, also dated November 29, 1983, was directed to the Office of Nuclear Reactor Regulation. Finally, on November 29, 1983, the Petitioner submitted his "Petition for Emergency Relief re: Primary Containment Leak Rate at LaSalle Units 1 and 2" to the Regional Administrator for Region III of the Nuclear Regulatory Commission. All three petitions (hereinafter referred to as the Petitions) address substantially the same technical issue, specifically, the adequacy of the integrated leak rate testing which is conducted on U.S. nuclear power reactor containments. Consequently, the Petitions have been consolidated and have been treated pursuant to 10 C.F.R. § 2.206 of the Commission's regulations. As the issue raised is primarily one related to the licensing standards for commercial nuclear reactor facilities, the Office of Nuclear Reactor Regulation has provided the substantive response to all Petitions.

The Petitions allege that there are severe errors, defects, and loopholes in the integrated leak rate testing (ILRT) methodology now in use including the standards of the American Nuclear Society (ANS) and the American National Standards Institute (ANSI), specifically ANS N45.4-1972 and ANSI/ANS 56.8-1981. It is alleged that the ILRT methodology now in use offers no guarantee that actual leak rates are acceptably low. The Petitions seek a variety of relief including requests for immediate action such as placing the LaSalle Unit 1 of the Commonwealth Edison Company in cold shutdown, ceasing further construction and licensing activities with respect to LaSalle Unit 2 and Byron Unit 1 (also of the Commonwealth Edison Company), and shutting down reactors with insufficient evidence of adequate containment leak rate testing. For the reasons set forth in my letter to the Petitioner of December 16, 1983, I declined at that time to take any immediate action based upon the preliminary evaluation conducted by the NRC Staff of the Petitions and other relevant information. I indicated at that time that the NRC Staff would continue to review the Petitions and that the Office of Nuclear Reactor Regulation would issue a formal decision with regard to them in the reasonably near future. On January 6, 1984, the Petitioner submitted an "Addendum to Petition for Emergency Relief" (Addendum) to the Office of Nuclear Reactor Regulation, requesting documentation in the ILRT area. My letter to the Petitioner of February 9, 1984 informed him that the Addendum would be treated as an FOIA request to the extent it sought documents within the possession of the NRC. With respect to the request that all data from the ILRT tests of the LaSalle, Byron and D.C. Cook plants not in the possession of the NRC be obtained and placed in the public document rooms to permit access for the general public, I declined to take such action at that time and stated that I would issue a formal decision regarding this matter in the reasonably near future. My decision in these matters follows.

## DISCUSSION

The Commission's requirements for integrated leak rate testing are set out in 10 C.F.R. § 50.54(o) and Appendix J to 10 C.F.R. Part 50. These requirements call for preoperational and periodic leak rate testing of commercial nuclear power plants in accordance with ANS N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." Experience gained with integrated leak rate testing of commercial nuclear facilities has been drawn together in an industry consensus document, specifically, ANSI/ANS 56.8-1981, "Containment System Leakage Testing Requirements," which provides detailed measures for performing the integrated leak rate testing required by Appendix J.

The Petitions allege that there are serious errors, defects, and loopholes in both ANS N45.4-1972 and ANSI/ANS 56.8-1981, and that the ILRT methodology now in use offers no guarantee that actual leakage rates are acceptably low. The specific defects alleged in the Petitions include:

(1) The equation used to calculate containment air mass at any given time is wrong;

(2) The final calculated leakage rate may be "fudged" in a variety of ways to presumably yield an invalid leak rate; and

(3) There are "loose" requirements for the permanent archiving of raw test data and other data essential for test evaluation.

With regard to the first alleged defect, the equation presented in ANSI/ANS 56.8-1981 for calculation of containment air mass is not wrong as alleged in the Petitions. The manner in which the mean containment temperature is calculated for use in the equation, however, is important. In this regard, ANSI/ANS 56.8-1981 does not prescribe how to calculate the mean containment temperature. Either a mass-weighted mean temperature or a volume-weighted mean temperature would be acceptable. While the use of a volume-weighted mean temperature is technically more correct, for reasonably well stabilized containment test conditions, as required by Appendix J to 10 C.F.R. Part 50, the error that could result from using the mass-weighted mean temperature is not significant. To illustrate the point that either a mass-weighted mean temperature or a volume-weighted mean temperature would produce acceptable results, five sets of data from the LaSalle Unit 1 ILRT were analyzed using both techniques. The results are set out in Appendix A (unpublished) attached to this decision. The difference in the two methods is quite small; *i.e.*, on the order of 0.35°R or about 0.06%. What is necessary is that integrated leak rate testing be properly conducted to assure stable conditions and that the test data be properly evaluated.

In examining this first alleged defect, the NRC Staff has reviewed both documents referred to in the Petitions. It should be noted that the NRC Staff was aware of the content of these documents prior to the date on which these Petitions were filed. The NRC Staff is also aware of the work of Dr. Zinovy Reytblatt, which is also referred to in the Petitions.

A properly conducted test of containment leak rate obviates not only the first concern raised in the Petitions with respect to the equation used to calculate the containment air mass, but also the second deficiency alleged, namely that final calculated leakage rates may be "fudged." A properly conducted test would not likely be flawed by the types of deficiencies alleged in the Petitions such as unjustified discarding of data or the use of unjustified weighting coefficients. Such manipulation of data would, of course, be a violation of the Commission's regulations and would subject licensees to NRC enforcement action. To ensure compliance with the Commission's requirements regarding integrated leak rate testing, NRC inspectors regularly observe the tests conducted by licensees and document the results of their observations. For example, the integrated leak rate test inspections for the LaSalle Units 1 and 2 and the Byron Unit 1 are documented in the following inspection reports:

Inspection Report 50-373/82-25 (June 9, 1982) — LaSalle Unit 1

Inspection Report 50-373/82-32 (July 29, 1982) — LaSalle Unit 1

Inspection Reports 50-373/83-28; 50-373/83-23 (DE) (July 28, 1983) — LaSalle Unit 2

Inspection Reports 50-454/83-40 (DE); 50-455/83-30 (DE) (October 11, 1983) — Byron Unit 1

Based on these inspections, the licensee's integrated leak rate testing for both LaSalle County Units 1 and 2 have been determined to meet current regulatory requirements contrary to the allegations of the Petitions. Leak rates at these facilities are within acceptable limits.

Although the inspection effort at Byron Unit 1 is not yet complete, the NRC Staff will similarly determine the acceptability of that integrated leak rate testing. A similar inspection effort is undertaken with respect to integrated leak rate testing on all other licensed commercial power reactors, thus providing assurance that the Commission's requirements are met.

With respect to the third deficiency alleged by the Petitions, namely that requirements for permanent archiving of raw data and other data essential to test evaluation are insufficient, licensees of commercial power reactors are required to retain records which furnish evidence of activities affecting quality of safety-related items, including reactor containment, pursuant to 10 C.F.R. § 50.71, 10 C.F.R. § 50.34, and Criterion XVII of Appendix B to 10 C.F.R. Part 50. Furthermore, the technical specifications, which form a part of the operating license for each plant, require the permanent retention of records associated with in-service inspections and tests required by technical specifications. Integrated leak rate testing is one of the in-service inspections and tests called out in the technical specifications. Finally, the NRC Staff has assured itself that records do in fact exist and are being retained at the LaSalle and Byron and Cook facilities.

With respect to access to records associated with ILRT, the NRC has access to all records of licensees related to their licenses as may be necessary to effectuate the purposes of the Atomic Energy Act of 1954, as amended. *See* 10 C.F.R. § 50.70(a). The bulk of such records are maintained by the licensees and unavailable for examination by the general public. With respect to licensee's records related to ILRT, the Petitioner requested that such records be obtained by the NRC and placed in the public document rooms. I see no clear benefit to Petitioner's request and consequently that request is denied. The volume of records maintained by licensees is enormous. To honor Petitioner's request would set in motion a practice that could eventually overwhelm the NRC files, both physically and financially with no clear benefit to the public. Remedies are available to the public should there be concerns with regard to a licensee's data or its evaluation. A petition pursuant to 10 C.F.R. § 2.206 as filed by Mr. Gogol is one such remedy. The NRC has pursued the ILRT matter and the records involved and assured itself that the Commission's regulations are being met. It will do likewise when other issues are brought to it for consideration. To honor Petitioner's request would incur substantial burdens and costs without a clear corresponding benefit.

In an effort to more fully understand the concerns raised by the Petitions and at the suggestion of Mr. Gogol, the NRC Staff met with Dr. Reytblatt on January 4, 1984. A copy of the summary of this meeting is attached as Appendix B (unpublished). At this meeting, Dr. Reytblatt expressed more fully his concerns with ILRT methodology and made specific reference to a critique prepared by him of ANSI/ANS 56.8-1981. As can be seen from the summary of the meeting with Dr. Reytblatt, a number of his concerns were resolved at the meeting itself. At the meeting, Dr. Reytblatt noted that many of his concerns had already been provided to the Commission in documented submittals. In fact, Dr. Reytblatt has made a number of submittals to the NRC from May 26, 1982 to July 26, 1983 critiquing ILRT methodology. These submittals were unsolicited and classified by Dr. Reytblatt as proprietary. Consequently, no detailed discussion of these submittals is presented here. However, the NRC has reviewed these submittals and the overall conclusions of the review are:

(1) No safety issue has been identified by Dr. Reytblatt of which the NRC was unaware or which requires NRC action;
(2) Dr. Reytblatt's technical concerns are adequately accounted for if the containment's test conditions are properly stabilized;
(3) Dr. Reytblatt's assumptions on the range of parameters to be encountered during an ILRT cannot realistically be expected to occur; and
(4) The calculated containment leak rate using current methods would not be significantly altered by use of Dr. Reytblatt's proposed technical refinements.

Integrated leak rate testing of nuclear power plant reactor containment is a substantial undertaking. The testing itself is a major undertaking, as is the analysis of the test data. While the Commission's requirements for integrated leak rate testing continue to provide reasonable assurance that the public health and safety is adequately protected, these requirements are now over 10 years old and a substantial base of experience exists to apply in seeking improvements to the regulations. In fact, one modification to 10 C.F.R. Part 50, Appendix J in the area of Type B tests was recently made. *See* 45 Fed. Reg. 2330 (1980) and 45 Fed. Reg. 62,789 (1980). The NRC Staff has under way a review of leak rate testing requirements with a view to see whether other modifications to these requirements are appropriate. Possible improvements could include clarification of the procedures and conditions governing the conduct of integrated leak rate tests. Substantial efforts have been undertaken in this area. As noted above, an industry consensus document, ANSI/ANS 56.8-1981, has been developed in this area. Both Petitioner and Dr. Zinovy Reytblatt are well aware of these activities and developments from their participation in the activities of Working Group ANS-56.8 of the Standards Committee of the American Nuclear Society. The concerns raised in the Petitions have been presented to the NRC Staff on a number of occasions in the past in both oral and written manner by the Petitioner and Dr. Reytblatt. Consequently, both the nuclear industry and the NRC Staff have long had the benefit of these concerns. And, as noted above, consideration of these concerns could result in appropriate modifications to Appendix J to 10 C.F.R. Part 50 at a future date.

The Commission has placed leak rate testing for water-cooled power reactors on its Regulatory Agenda. 48 Fed. Reg. 52,931 (1983) and NUREG-0936, "NRC Regulatory Agenda" Vol. 2, No. 3 (November 1983). At the time a Notice of Proposed Rulemaking is issued, the Petitioner, along with other interested members of the public, will be given an opportunity to comment. As a general rule, the Director will not institute proceedings in response to a petition under 10 C.F.R. § 2.206 to consider an issue the Commission is treating generically through rulemaking. *See Maine Yankee Atomic Power Co.* (Maine Yankee Atomic Power Station), DD-83-3, 17 NRC 327, 329 (1983) and cases there cited. To the extent the Petitioner sought initiation of a rulemaking, his Petitions were misdirected. *See* 10 C.F.R. § 2.802. Should Petitioner wish the Commission to initiate a rulemaking in this matter, the procedures set forth in 10 C.F.R. § 2.802 should be followed. However, as noted above, ILRT is already on the Commission's Regulatory Agenda.

## CONCLUSION

In summary, compliance with the Commission's current regulations regarding integrated leak rate testing of commercial nuclear facilities provides reasonable assurance that the public health and safety are adequately protected. An important consideration in assuring compliance with these regulations is proper conduct of the tests and evaluation of the test data. To this end, the NRC Staff has in place an inspection program to monitor such testing and observe data reduction. NRC Staff findings are routinely documented in inspection reports for the affected facilities. Specific findings of these inspection efforts for LaSalle Units 1 and 2 have been reviewed and the NRC Staff has determined the integrated leak rate testing for these facilities has been properly conducted and that these facilities are in compliance with the Commission's regulations in this area. Requirements for archiving of data have also been reviewed and are satisfactory to ensure availability of data for future review should the need arise. Consequently, I conclude that the overall state of integrated leak rate testing regarding commercial nuclear power facilities is adequate to assure the public health and safety. Accordingly, I decline to take any of the actions solicited by the Petitions. For reasons stated above, I also decline to grant the request sought by Petitioner's Addendum, specifically that all licensee's ILRT data be placed in the public document room.

The Petitioner's request for action pursuant to 10 C.F.R. § 2.206 is denied. As provided by 10 C.F.R. § 2.206(c), a copy of this decision will be filed with the Secretary for the Commission's review. This decision is made without prejudice to the Petitioner's filing of a petition for rulemaking in accordance with 10 C.F.R. § 2.802.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland,
this 16th day of March 1984.

[The attachments have been omitted from this publication but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung**, Director

**In the Matter of** | **Docket No. 50-397 (10 C.F.R. § 2.206)**

**WASHINGTON PUBLIC POWER SUPPLY SYSTEM (WPPSS Nuclear Project No. 2)**

**March 19, 1984**

The Director of the Office of Inspection and Enforcement denies a petition of the Coalition for Safe Power requesting that the Nuclear Regulatory Commission institute show-cause proceedings pursuant to 10 C.F.R. § 2.202 to determine whether the construction permit for the Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) should be revoked, a stay of construction imposed, the pending application for an operating license denied, and hearings instituted before an Atomic Safety and Licensing Board. The petition alleged as its supporting bases deficiencies primarily in the construction and management of the WNP-2 facility.

## TECHNICAL ISSUE DISCUSSED: QUALITY ASSURANCE

It would be unreasonable to hinge the grant of an NRC operating license upon a demonstration of error-free construction. What is required is a careful consideration of whether all ascertained construction errors have been cured and whether the errors indicate that there has been a breakdown in quality assurance procedures of sufficient dimension to raise legitimate doubt as to the overall integrity of the facility and its safety-related structures and components. *Union Electric Co.* (Callaway Plant, Unit 1), ALAB-740, 18 NRC 343, 346 (1983).

RULES OF PRACTICE: SHOW-CAUSE PROCEEDINGS

An order to show cause is appropriate in those instances in which the NRC concludes, based upon alleged violations by the licensee or potentially hazardous conditions or other facts, that enforcement action should be taken but that a basis could reasonably exist for not taking the enforcement action proposed. *See* 10 C.F.R. § 2.202(a)(1) and 10 C.F.R. Part 2, Appendix C, § IV.

RULES OF PRACTICE: SHOW-CAUSE PROCEEDINGS

Sufficient grounds must be present for the NRC to institute a show-cause proceeding. The standard to be applied in determining whether to issue a show-cause order is whether substantial health or safety issues have been raised.

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

## Introduction

On October 14, 1983, the Coalition for Safe Power (Petitioner) filed its "Show Cause Petition from the Coalition for Safe Power Requesting Revocation of the Construction Permit and Denial of an Operating License for Washington Public Power Supply System Nuclear Project No. 2" (Petition). The Petition requested that the Nuclear Regulatory Commission institute show-cause proceedings pursuant to 10 C.F.R. § 2.202(a) to determine whether the construction permit for the Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) should be revoked, a stay of construction imposed, the pending application for an operating license denied, and hearings instituted before an Atomic Safety and Licensing Board. The Petition alleged as its supporting bases deficiencies primarily in the construction and management of the WNP-2 facility. Receipt of the Petition was acknowledged by letter of November 9, 1983. By letter of December 20, 1983, I advised the Petitioner that a review of the Petition had been conducted jointly by the Offices of Nuclear Reactor Regulation, Inspection and Enforcement, and Region V, and that the issues raised in the Petition had been evaluated. That evaluation concluded that the issues raised in the Petition should not preclude the issuance of an operating license for the

WNP-2 facility.[1] Based in part on the results of that evaluation, the Office of Nuclear Reactor Regulation on December 20, 1983, issued a license to the Washington Public Power Supply System (WPPSS or licensee) to permit fuel loading and low-power testing for the WNP-2 facility. My letter of December 20, 1983 further informed the Petitioner that, on the basis of the evaluation conducted to date, I did not intend to grant the relief sought by the Petition and that my formal decision, in accordance with 10 C.F.R. § 2.206(b), would be issued in the reasonably near future.

My formal examination of the Petition follows. The Petition raises essentially five issues, each of which is treated in turn.

## Background

To meaningfully discuss the issues raised by the Petition, some background information is required.

WPPSS was issued Construction Permit No. CPPR-93 (Nuclear Project No. 2) by the Atomic Energy Commission in 1973, which authorized construction of the WNP-2 plant. The WNP-2 plant is located near Richland, Washington and consists of one 1100-MWe boiling water reactor of General Electric design and related facilities for use in the commercial generation of electric power.

Early construction activities at WNP-2 were routine; however, in 1978, NRC inspections revealed signs of poor-quality construction. Intensified NRC inspection efforts resulted. Several investigations were conducted in response to noted construction problems and allegations. In May 1980, meetings were held between the NRC and licensee upper management to focus needed attention on observed problem areas.

An NRC investigation completed in February 1980 resulted in the imposition of a Civil Penalty in the amount of $59,500 for identified structural deficiencies in the sacrificial shield wall (SSW); failure to establish and use a suitable test program for the SSW; failure to provide control of special construction processes; various procedural inadequacies; and generally inadequate record-keeping practices. A

[1] No hearings were held regarding issuance of an operating license. A hearing notice was issued on July 26, 1978. *See* Receipt of Application for Facility Operating License; Notice of Consideration of Issuance of Facility Operating License; and Notice of Opportunity for Hearing, 43 Fed. Reg. 32,338 (1978). An intervention petition was filed in response to the notice.

By "Order Subsequent to the Prehearing Conference on January 25, 1979," the Atomic Safety and Licensing Board, on March 6, 1979, concluded that no justification for granting intervention in the operating license proceeding existed. *Washington Public Power Supply System* (WPPSS Nuclear Project No. 2), LBP-79-7, 9 NRC 330 (1979). On October 9, 1979, the Atomic Safety and Licensing Board issued a "Notice of Dismissal of the Proceeding."

letter was also issued on June 17, 1980 pursuant to 10 C.F.R. § 50.54(f) seeking assurance from the licensee that the WNP-2 quality assurance program would be improved and implemented to ensure adequate quality of construction. In addition, the letter requested that the licensee develop a plan for determining the quality of past work. In July 1980, another NRC investigation resulted in the identification of twelve items of noncompliance which demonstrated a continuing problem in the control of safety-related work being performed by contractors at WNP-2.

The licensee, in response to the enforcement actions, issued stop-work orders to all WNP-2 site contractors to permit initiation of appropriate corrective measures. The NRC issued a Confirmatory Action Letter (CAL) to the licensee in July 1980 regarding a stop-work order applicable to the principal mechanical contractor. NRC hold points were placed into effect to ensure timely review of the licensee's corrective actions prior to restart of work by the principal mechanical contractor. The NRC inspection force for WNP-2 was supplemented to provide increased audit capability.

The licensee's corrective measures included significant changes to the quality assurance program, including a 100% review and revision of the quality-related work procedures. Major personnel changes were made to the site management organization, and a new construction manager (Bechtel) was brought in to review the adequacy of previous work and provide surveillance over new work. In 1981, safety-related construction work was permitted to resume.

In response to NRC concerns about the quality of past construction work, the licensee initiated a Quality Verification Program (QVP)[2] to determine the quality of work completed up to 1980. This effort, performed in large part by the various contractors involved, was conducted under surveillance by the newly hired construction manager (Bechtel) in accordance with approved procedures. The NRC staff performed independent audits of the overview effort, inspected samples of QVP documentation, reviewed report findings and independently verified selected pieces of WNP-2 hardware.

[2] The initially conceived program was titled Reverification of Completed Safety-Related Work (RCSW); it encompassed both work restart and the projected hardware reinspection activities. After work restart, the reinspection activity was titled the Quality Verification Program (QVP). Detailed record review activities were later also encompassed by the QVP, as were other special review and rework programs at the site. The QVP was loosely referred to as the reverification program. A separate design-oriented review was performed during 1982-1983, titled the Independent Design Verification Program (IDPV). The QVP and IDVP were together considered the Plant Verification Program.

The NRC Construction Appraisal Team (CAT)[3] also included various pieces of hardware in its special site inspection during 1983. This inspection identified various discrepancies between as-installed pipe supports and the construction drawings, reinforcement steel placement deficiencies, several welding concerns and questionable bolting installations. These matters were referred to the NRC Region V inspection staff for follow-up and resolution in accordance with standard practices.

From the time of the identification of the major construction problems noted above, the NRC staff has conducted a series of Systematic Appraisals of Licensee Performance (SALP) reviews in connection with the WNP-2 facility. These reviews provided a basis for the NRC to evaluate the positive and negative attributes of the licensee's performance. These reviews assisted in improving both licensee performance and the effectiveness of the NRC regulatory program.

The first SALP Board findings at WNP-2 for 1980 centered primarily on significant weaknesses in various aspects of the licensee's quality assurance program. Other findings dealt with a lack of control over the quality of work by the site contractors. The associated SALP report discusses concerns over the number and type of noncompliance items in areas such as safety-related structures, piping and pipe supports, electrical installations and record-keeping practices.

The SALP Board findings for 1981 recognized significant changes the licensee had made in the WNP-2 construction organization. Experienced management personnel had been brought in to implement newly established project management programs to better control the quality of construction. However, the Board also noted that the licensee had been remiss in moving ahead in some areas without making associated changes to the program plan and developing and issuing needed procedures in a timely manner. Other items were identified which highlighted concerns over incomplete corrective measures, design changes, piping supports, and timeliness of responses to NRC licensing matters.

The SALP Board findings for 1982 found the overall performance of the licensee to be acceptable. However, two weaknesses were identified in the areas of design and installation of electrical systems and the implementation of proposed corrective action relative to a reported construction deficiency.

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[3] The Construction Appraisal Team, developed as an NRC headquarter function, focuses primarily on determining the quality of safety-related structures, systems and components by direct hands-on inspections. To a lesser extent, the licensee's quality assurance program is relied on by the team to confirm the findings of the hands-on inspection results. The CAT is made up of highly qualified NRC and independent contractor personnel selected to provide in-depth evaluations of plant hardware.

The most recent SALP Board review on the WNP-2 project for 1983 found that the licensee has been responsive to the previous findings of the SALP reviews and that acceptable corrective actions have been initiated, supervised and directed at the highest level of management. While the Board's findings were favorable, the licensee was strongly encouraged to be especially alert for signs of performance deterioration during completion of the project.

The SALP reviews at WNP-2 served, in large part, to focus attention on the licensee's capability to provide prompt and effective corrective actions to identified construction problems. SALP Board findings, as a matter of policy, have been acted upon by the cognizant NRC regional and site staff as needed to bring about desired changes in inspection emphasis and follow-up of identified weaknesses.

During the latter part of 1982, NRC staff discussions with the licensee led to the development and implementation of an Independent Design Verification Program (IDVP) effort at WNP-2. This effort was directed toward gaining additional assurance regarding the as-built design of the facility and was carried out by licensee staff personnel who were independent from WNP-2 design and construction responsibilities. This effort centered on three safety-related reactor systems. Also, studies were conducted to evaluate operational interaction between the reactor systems. An independent consultant firm, Technical Audit Associates Incorporated (TAA), evaluated the technical adequacy of the IDVP and audited its entire implementation. The NRC staff has completed its evaluation of the program and its findings and concludes that there are no indications of significant deficiencies in the WNP-2 design process and that the design verification program provides additional confidence that acceptable QA design practices were followed during construction of the facility.

Before discussing each one of the five major areas identified in the Petition, it is important to recognize that the Petition provides no new information. The Petition consists almost exclusively of excerpts taken from findings of NRC Inspection Reports. The NRC inspection program recognizes that deficiencies will be found as a result of inspection activities. Corrective action is required for every violation of NRC requirements. *See* 10 C.F.R. § 2.201. Consequently, all the allegations in the Petition which stem from inspection findings have been the subject of corrective action and have been adequately resolved.

With respect to the other allegations raised in the Petition, the NRC has been generally aware of these matters and had taken action with respect to them to the extent appropriate.

The Petition does not provide new information but only restates that which the NRC was already aware of and had already addressed in various inspection and investigation reports. Consequently, the response to the Petition has been organized around the five principal issues raised by the Petitioner to indicate how the corrective action process worked to resolve the various concerns, rather than by a detailed discussion of each of the scores of allegations contained in the Petition, which have already been looked at and resolved once before during the inspection process.

## Principal Issues Raised by Petitioner

### I. THE QUALITY ASSURANCE PROGRAM AT THE WNP-2 FACILITY

The first issue raised by the Petitioner concerns allegations of numerous failures by WPPSS to implement an adequate quality assurance program at the WNP-2 facility as required by 10 C.F.R. Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The Petitioner discusses a number of areas wherein it is alleged that the quality assurance provisions of Appendix B have not been met. Such areas include design control, record control, worker qualifications, and materials control. The Petitioner refers to various inspection findings and reviews conducted by either Region V or the Office of Inspection and Enforcement in support of the contention that quality assurance at the WNP-2 facility is deficient. The NRC inspection activities routinely find deficiencies in the performance of construction activities at nuclear power plants. It is not unusual, therefore, that such deficiencies were identified in the NRC Inspection Reports referenced in the Petition. NRC inspection activities involve the auditing of construction of nuclear facilities with the purpose of assuring that the overall construction program in place at a construction site is effective in ensuring that proper quality standards are maintained. Such inspection activities routinely result in enforcement actions and the identification of unresolved items. Isolated deficiencies in the licensee's program do not necessarily undermine the program to such an extent as to give rise to a significant safety concern. Given the magnitude of construction activities associated with completing a nuclear power plant, even numerous deficiencies in such construction activities do not necessarily give rise to a significant safety concern. As has been recently recognized by the

Atomic Safety and Licensing Appeal Board,[4] it would be unreasonable to hinge the grant of an NRC operating license upon a demonstration of error-free construction. What is required in any inquiry is a careful consideration of whether all ascertained construction errors have been cured and whether the errors indicate that there has been a breakdown in quality assurance procedures of sufficient dimension to raise legitimate doubt as to the overall integrity of the facility and its safety-related structures and components.[5]

The following specific categories of quality assurance issues raised by the Petition have been resolved as follows:

## A. Design Control

The Petitioner contends that the licensee experienced an ongoing failure to conform to applicable design criteria, specifically Criterion III of 10 C.F.R. Part 50, Appendix B, Design Control. The Petition references several NRC design and construction inspection observations which discuss the standby service water system (SSWS). The Petitioner concludes that there has been a general failure by the licensee to conform to applicable design criteria with respect to the SSWS. The Petition further presents NRC findings regarding electrical cable separation and concrete structures, and concludes that lack of design control in these areas has resulted in failure to meet the appropriate design criteria. The Petition then concludes that there is no reasonable assurance that Criterion III of Appendix B has been met. These matters are discussed below.

In the 1981 Inspection Report 50-397/81-17, referenced in the Petition, the NRC inspector observed that forty-nine questions had been identified by the architect-engineer during a design audit of various quality systems, including the SSWS. The NRC inspector noted the absence of anyone's assessment of the significance of these questions relative to the effectiveness of the design verification quality assurance program and concluded the design verification process appeared to deserve further review. The licensee committed at that time to address the inspector's concerns prior to continued execution of the program. The NRC subsequently reviewed details of the existing design verification process, both at the site and the home office of the architect-engineer, and concluded that the process in-place was acceptable.

In 1982 the NRC observed that Drawing Interim Revision Sheets on the SSWS had not been compiled and incorporated into composite draw-

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[4] *Union Electric Co.* (Callaway Plant, Unit 1), ALAB-740, 18 NRC 343, 346 (1983).

[5] *Id.*

ings at the frequency called for by procedures and questioned whether working drawings were correct. This was considered to be a relatively minor administrative matter and no items of noncompliance or unacceptable work were identified. The drawings were subsequently updated and controls were adopted and initiated to assure that working drawings were correct.

The SSWS did not provide the design coolant flow for certain pieces of equipment during the initial preoperational tests. On August 11, 1983, the licensee properly reported this information to NRC in accordance with 10 C.F.R. § 50.55(e) and took appropriate corrective actions, including the installation of flow-restricting orifices, valve position adjustments, and system cleaning. Additional permanent orifices will be installed to preclude the need for continued valve position adjustments.

With respect to electrical cable separation, the 1982 SALP Board expressed concern and the need for a clear definition of acceptance criteria for ensuring the electrical independence of redundant safety-related circuits. This area relates to NRC observations and findings documented in various inspection reports and management meetings from 1978 through 1983 regarding the licensee's efforts to interpret industry standards and define field inspection criteria for electrical cable installation separation inspection. This matter has been reviewed by the NRC licensing technical staff of the Office of Nuclear Reactor Regulation and final agreements have been reached on the appropriate criteria and the nature of field inspections. The NRC staff noted the resolution of this matter during the 1983 SALP review.

With respect to electrical cable installation, the CAT identified some errors in cable pull slips but did not identify any improperly installed cables associated with these errors. Also, field variations noted within a group of single-division wall penetrations do not violate separation requirements. With respect to cable tray separation, the licensee has now considered installation tolerances and has conducted physical walkdown inspections of tray installations and has taken appropriate corrective actions for noted discrepancies.

With respect to concrete structures, the CAT identified a number of deviations in the structures. The CAT identified reinforcing steel spacing discrepancies in several areas, and an inability to locate some steel dowels or determine reinforcing steel patterns within the areas of limited excavation of concrete. The licensee subsequently performed additional concrete excavations and obtained additional data. NRC inspectors, including CAT representatives, performed additional inspections of licensee actions. The nature and extent of the discrepancies and their impact on the ability of the structure to take its design loads has been considered by the licensee assuming worst-case loadings and discrepancies. Evaluations were also conducted by independent third parties. The as-built structures were found to be capable of substantially exceeding design loads. The deviations would have initially been acceptable under the American Concrete Institute Code had they been evaluated before rather than after the fact. No programmatic changes were required since all civil structures had been completed and no additional work is contemplated. The structural significance was also evaluated and found acceptable by NRC licensing technical staff of the Office of Nuclear Reactor Regulation. These conclusions are documented in Supplements 4 and 5 to the Safety Evaluation Report (SER)[6] for the WNP-2 facility.

After the July 1980 stop-work action, the licensee conducted a work restart and work reverification program as part of the QVP and reviewed the past design change control and nonconformance control systems. This included reviews of samples of the different kinds of design change procedures used on the project and reviews of the engineering disposition of nonconformance report documents. This effort was reviewed by the NRC staff which concluded that the design change control process had been improved substantially and was now acceptable. Since that time, design change control has been satisfactory.

To provide additional assurance of proper design control, the licensee conducted its IDVP to review the design activities performed by the architect-engineer, using licensee engineers not associated with the WNP-2 project design and construction responsibilities. Overview audits were conducted by an independent audit organization, TAA. The NRC staff has reviewed the results of the IDVP and concludes that it does provide reasonable assurance that the WNP-2 facility has been designed in accordance with Criterion III of 10 C.F.R. Part 50, Appendix B, Design Control.

### B. Record Control, Worker Qualifications, Material Control, and Maintenance and Preservation

The Petitioner contends that NRC inspection data show that: (1) the licensee has had a continuing inability to produce adequate documentation required by NRC regulations; (2) engineering, quality assurance

[6] NUREG-0892, "Safety Evaluation Report Related to the Operation of WPPSS Nuclear Unit No. 2," and Supplements 1 through 4. Supplement 4 was issued in December 1983; Supplement 5 is in preparation and will be issued shortly.

and craft personnel were not properly qualified and/or trained; (3) measures had not been established for the identification and control of certain materials to prevent the use of incorrect material; and, (4) there has been a continuing problem with site housekeeping, cleanliness control, and preventative maintenance for safety-related systems or components. These areas are addressed below.

The subject areas have been periodically audited by NRC inspectors as a part of the NRC's routine inspection program to assure that an acceptable level of performance is being maintained by the licensee and its contractors. The examples the Petitioner describes are generally reflective of our inspection experience at this and other construction sites. The licensee has been responsive to the inspection findings and has taken the necessary corrective actions to resolve each of the issues identified.

The licensee experienced problems with generation of quality assurance records prior to July 1980. Default of some contractors contributed to the loss of some records, as did deferrals of final reviews of certain work packages. Through 1980 and after, efforts were undertaken by the licensee to identify the discrepancies in past records and resolve any omissions, discrepancies, and deficiencies. Corrective actions were taken in the specific areas mentioned in the Petition. The licensee's corrective action programs, including quality control reinspections, have been monitored by NRC inspectors. Corrective actions and resolutions, including CAT findings, have been documented in NRC Inspection Reports. No deficiencies remain outstanding.

With respect to worker qualifications, the licensee, as a part of the QVP, conducted reviews to determine the nature of design-change-type actions by engineering personnel, and found such actions acceptable. These reviews were also used to evaluate engineering qualifications. Past work performed by crafts and inspected by field quality control personnel contained discrepancies which may have been due to qualifications problems or lack of training. Work and records associated with such work have been reexamined under the QVP and the as-built programs to provide additional confidence of worker and inspector qualifications. After the July 1980 period, the indoctrination and training programs for craft personnel and quality control inspectors were reinforced.

The Sandler Affidavit contends that worker qualifications were still considered to be a problem at WNP-2 as late as November 1982 based on a discussion with an NRC inspector. Specific issues, including the discussion with the NRC inspector, as related to the Sandler Affidavit regarding welding engineers and quality assurance personnel qualifications are discussed in Appendix A (unpublished). Resolution of these items, including CAT findings, has been documented in NRC inspection reports.

With respect to material control, there have been isolated cases of material control discrepancies even though the licensee and its contractors had established general and specific programs for material identification and control in accordance with NRC regulations. The CAT observed that an incorrect grade of nuts appeared to have been installed in some pump couplings, pipe flanges, and pressure relief valves. Of twenty-one bolted connections inspected, identifying markings were not observed on five. Markings indicated that an inferior grade had been used on six others. Subsequent removal and inspection of the nuts showed that some markings had been concealed on the underside of the nut. The licensee conducted a complete reinspection of the fasteners for mechanical equipment and identified that only Grade 2H and Grade 7 nuts had been used. These are of equivalent physical and chemical properties for the temperature service of the equipment since physical properties only differ above approximately 900°F. For flanges, the architect-engineer considered the highest fastener loading with the lowest commercially available grade bolting and found this within the Code-allowable bolt stress. The licensee also corrected the minor discrepancies identified in mixed nuts in the material bins at the Bechtel warehouses and audited the other bins. This audit was performed in conjunction with the investigation of apparent improper bolting materials in equipment flanges and couplings.

NRC observations in late 1978 through early 1980 identified weaknesses in the licensee's site housekeeping and system cleanliness control, and equipment preventive maintenance program during construction. Repetitive findings in this area in 1979 resulted in NRC enforcement actions which subsequently led to effective corrective measures by the licensee. Continued NRC attention to this area identified a few additional minor discrepancies in 1981. The enforcement history in this area does not support the existence of significant uncorrected defects in plant structures or equipment. System flushing and preoperational testing provided additional means for the identification and correction of conditions which may have resulted from past weaknesses in the cleanliness, housekeeping and preventive maintenance programs.

With respect to preventive maintenance, the CAT observed that no deficiencies were noted with the preventive maintenance requirements or actions taken for the sample of thirty-six components reviewed. The CAT further observed that the system appeared to be effective. The

CAT concluded that the system currently in place is consistent with regulatory requirements but requires further updating, which is now being done by the licensee.

### C. Quality Class II

The Petitioner alleges that there is no reasonable assurance that the licensee has or will apply installation and inspection techniques to Quality Class II (QCII)[7] equipment important to safety and governed by Criterion II of Appendix B. While the Petitioner acknowledges that Appendix B is generally applied to Quality Class I (QCI) materials and equipment,[8] the Petitioner alleges that there are numerous instances where QCII equipment is important to safety but Appendix B is not applied. The Petitioner describes CAT questions regarding quality assurance measures for supports of Quality Class II/Seismic Category I installations which could impact safety-related equipment in the event the support fails.

The licensee has always applied quality assurance measures to QCII/Seismic Category I installations. For piping and component supports, this included design by the architect-engineer to QCI standards and installation to QCI procedures. In 1981, subsequent to the June 1980 work stoppage, the new construction manager concluded that rework and new installations of QCII/Seismic Category I items did not require all of the same documentation and inspection activities required for QCI items as was the case prior to the stop-work order. The program was therefore revised to permit field inspection by the construction field engineers responsible for this work. The NRC reviewed this change and determined it to be acceptable and equivalent to practices at other nuclear plant construction sites.

The Petitioner notes that the CAT found that the various quality control inspection and as-built programs have not been totally effective in identifying installed hardware that does not meet design requirements. CAT also found that the accuracy of previous inspection and as-built information for QCII/Seismic Category I supports and restraints did not appear to provide sufficient confidence in the acceptability of this hardware. An NRC Notice of Violation was issued relative to this matter. The licensee corrective actions, to problems identified by the CAT, included inspections and engineering evaluations, including use

[7] QCII items are those items that do not have a safety function but their failure could impact safety-related equipment.

[8] QCI materials and equipment have been defined as safety-related components and are specifically identified in the licensee's Final Safety Analysis Report (FSAR).

of a third-party engineering organization to assess the installations. The evaluation data showed that none of the deviations found would have a significant effect on the structural integrity of the supports. The evaluation data were reviewed by the NRC staff which concluded that the as-built QCII/Seismic Category I structures and supports provide adequate margin of safety and are consistent with NRC requirements. The Office of Nuclear Reactor Regulation with responsibility for licensing the WNP-2 facility has reviewed the CAT concerns with representatives of the CAT organization and concluded that there are no outstanding issues.

**D. Procedures**

The Petitioner contends that procedures have not been properly produced, reviewed, and utilized, and that there has been a failure to properly use procedures at WNP-2 affecting preoperational testing, environmental sampling, reverification, and systems lineup, which continues to this day.

The procedures in use at WNP-2 were generated by various contractors engaged in work on the project, to govern their internal activities and to interface with other organizations at the site. All contractors were required to have quality assurance programs for safety-related work. Each contractor's internal quality assurance organization included a quality control section for direct inspection of hardware to assure compliance with procedures and/or specifications. The contractor organizations included quality assurance audit sections, for assessing internal compliance with procedures by all elements of the organization. In addition, the construction manager (Burns and Roe) had oversight responsibility over the site contractors, which included review of each contractor's work procedures. Such reviews were conducted by segments of the Burns and Roe organization staffed with personnel with appropriate qualifications. Technical, contractual or quality assurance aspects were considered by the organizational element most familiar with the subject, and procedures were routed to such elements for review.

The Burns and Roe organization included technical groups to assist the contractors in handling and processing of field-identified problems prior to submittal to design engineering. The construction manager also included a quality assurance staff to perform daily surveillance over the activities of the site contractors, and to perform formal audits of the contractors and the construction manager's own internal organization. As a part of the surveillance function, the construction manager's quality assurance staff received copies of the contractor's work procedures for comment regarding inclusion of quality assurance program requirements. The coordinating function for both technical and quality assurance comments was generally performed by project engineering.

The NRC staff generally performed monthly inspections of work procedures, in-process work, and records. The inspectors looked for compliance with applicable codes, standards, commitments to the NRC, and additional specification requirements imposed by the architect-engineer.

As a result of these reviews and other associated work experiences, the various work procedures were revised many times. Some revisions were necessary to resolve ambiguities or errors, improve methods of performance, and to reflect design changes. Some changes were made to prevent recurrence of situations identified by auditors or NRC inspectors.

As a result of the NRC inspection findings from 1978 through 1980, indicating numerous deficiencies in implementing procedures, the licensee stopped work in July 1980 and initiated a complete review of the work procedures of each contractor engaged in safety-related work. Contractors were required to perform comprehensive reviews of their quality discrepancy documents to identify negative trends. These trends were considered when the new procedures were reviewed to assure that program changes would be implemented to preclude recurrence. Their revised procedures were then reviewed by a task force of independent reviewers under direct management of the licensee. These reviews were generally completed in early 1981. The task force compiled the significant discrepancies identified during the reviews and provided these and the backup data sheets to the contractors for consideration in the subsequent record review and hardware reinspection programs, to ascertain adequacy of work completed prior to upgrade of the procedures. The licensee also performed technical re-review on a sample of work procedures for inactive and pre-purchased contracts as a result of issues raised by an NRC management team in 1983. The procedure review and work reverification activities were monitored by an NRC Resident Inspector and Region V inspection staff between July 1980 and December 1983. The NRC staff finds that the licensee had implemented proper procedures for the QVP. These procedures were reviewed, approved, and monitored by Bechtel. The QVP accomplished its intended mission.

In June 1983, the CAT inspection examined various types of hardware at the site, including items subject to the reinspection program and the applicable procedures. The CAT found minor hardware discrepancies and raised questions related to procedure content and/or adherence.

Licensee management promptly responded to these items and undertook their satisfactory resolution.

The inspection history shows that there have been individual problems with the production and use of procedures at WNP-2. With a project of this magnitude, omissions and errors cannot be precluded. Licensee management has been advised of procedural deficiencies and cases of failure to follow procedures, including the specific cases raised in the Petition. In each case, results of subsequent NRC inspections indicate that licensee management has been responsive, corrective action has been initiated, and the items have been satisfactorily resolved, including the CAT issues referred to by the Petitioner.

In discussing the procedures issue, the Petitioner claims that the Sandler Affidavit attached to the Petition demonstrates procedural deficiencies. The issues raised by the Sandler Affidavit had been examined prior to receipt of the Petition by the NRC and either found acceptable or satisfactorily resolved. The Sandler Affidavit is discussed in detail in Appendix A to this decision. The conclusion of the NRC staff is that all the issues raised by the Sandler Affidavit have been satisfactorily resolved.

## E. Corrective Actions and As-Built Plant

The Petitioner contends that the NRC inspections to date demonstrate or strongly suggest that the licensee has not complied with the NRC quality assurance criteria for corrective action in that the licensee has not addressed the underlying programmatic causal factors of individual problems and has an inability to identify, analyze and ensure proper and timely completion of corrective actions. The Petitioner points specifically to continuing difficulties with the Bechtel as-built program[9] for examining pipe supports and restraints. Given the alleged inability of the licensee to take timely corrective action, it is not clear to the Petitioner why the NRC continued to rely on the good-faith effort of the licensee.

The NRC staff has examined the examples presented in this section of the Petition and has determined that each one has been satisfactorily resolved. Based on NRC staff inspection experience, the licensee has routinely addressed causal factors leading to deficiencies in its actions to prevent recurrence and has repeatedly demonstrated the ability to take

[9] The Bechtel as-built program was a 1982-1983 effort to identify differences between as-installed pipe supports and installation records. It encompassed the consideration of NRC Bulletin 79-14, regarding seismic analysis of pipe supports.

effective corrective actions. As a consequence, there is reasonable assurance that the plant has been constructed substantially in accordance with the conditions of the construction permit and NRC regulations. This has been confirmed by an examination of the as-built pipe supports and restraints, including the utilization of an independent third party to perform evaluations and independent calculations of a sample of these items. Corrective actions have generally been of sufficient timeliness.

In March 1983, an NRC special management team reviewed the status of implementation of the QVP commitments made to NRC by the licensee in July 1980. The team found ten areas where the licensee's interpretation or implementation of program commitments did not appear consistent with the NRC's initial understandings or expectations. Most of the NRC questions related to the scope or implementation of reviews of records and material supplied by inactive site contractors or offsite material suppliers. It is noted that the licensee had previously informed the NRC of changes in commitments prior to their implementation, both by phone and in bi-monthly reports. However, the NRC staff did not specifically agree to these changes at the time. The licensee was cooperative in addressing the issues raised by the NRC, and implemented additional reviews and field inspection activities to satisfy the NRC staff that the reverification would be effective and adequately implemented.

The 1982 SALP Board recommended that additional effort appeared warranted to ensure implementation of corrective action decisions regarding significant construction deficiencies reported under 10 C.F.R. § 50.55(e). This was based upon inspection findings that the licensee had not instituted an effective tracking system for assuring that directives for corrective actions were in fact accomplished and accomplished satisfactorily. The 1983 SALP acknowledged licensee quality assurance actions, including adequate corporate audits to assure that all corrective actions were completed, but also noted the need for continued vigilance in verifying the adequacy and implementation of corrective actions.

The CAT found that the Bechtel as-built program had not been totally effective in identifying hardware deficiencies. However, the CAT observed that the deficiencies found during the team's inspection of QCI piping and supports would probably not endanger system function. The licensee's actions on this matter included: (1) a sample reinspection performed by the WPPSS staff, including the CAT inspection sample, and evaluation of the findings; (2) a Burns and Roe evaluation of the Bechtel as-built program and discrepancies identified; and (3) Stone and Webster (third-party architect-engineer) performance of independent field measurements and an assessment of design capability of a sample of pipe supports.

The engineering evaluations and assessments of worst-case conditions, performed by Burns and Roe and Stone and Webster, concluded that none of the deviations impacted the design, function, or operability of the specific supports and that similar deviations in other supports would not significantly affect their structural integrity. The evaluations and assessments were reviewed by the NRC staff of the Office of Nuclear Reactor Regulation and it determined that none of the deviations had a significant effect on the structural integrity of the support. The NRC staff conclusions were documented in Supplements 4 and 5 to the SER for the WNP-2 facility.[10]

In summary, the NRC staff has considered the matters referenced in the Petition and has determined that these matters have been satisfactorily addressed or resolved.

### F. Test and Startup

The Petitioner contends that the NRC inspection findings demonstrate that the startup organization at WNP-2 is unqualified and its activities reflect the same deviations from FSAR and Appendix B requirements as has the QA/QC program as a whole. The Petitioner references CAT findings that the startup organization has failed to develop adequate documentation to ensure that sufficient corrective actions were taken when deficiencies were identified.

The NRC Staff has examined the examples identified in the Petition and does not consider them significant. The NRC inspection findings referenced in the Petition, regarding test and startup activities were of minor significance and were adequately resolved by the licensee. The licensee revised the startup program to eliminate the deficiencies referenced by the Petitioner through establishment of separate functions, *i.e.*, the startup personnel would perform those tasks for which they were specifically qualified and other construction and quality-related inspections would be handled by others specifically trained and qualified in these areas.

Startup activities are unlike construction activities in that they principally involve the conduct of operational-type tests to determine if the equipment meets its design function. Findings are referred to other organizations for evaluation, repair, or modifications. The weaknesses that

[10] *See* note 6, *supra*.

have been identified in the startup program are principally in the area of documentation and do not necessarily suggest inadequacies in the plant hardware.

The startup organization is different from the construction organization. Different individuals are involved and the staff is much smaller. Thus, there is insufficient basis for concluding that inadequate performance of the construction organization is reflective of inadequate performance of the startup organization.

The CAT did observe that adequate documentary evidence of corrective actions could be provided for only thirteen Inspection Reports (IRs) from a group of fifty-six reviewed. Each of these IRs was originated by a person within the electrical contractor organization to document what that person perceived to be departure from the electrical specifications. Although each such IR involved equipment which was no longer the responsibility of the electrical contractor, and no longer under his quality assurance program control, these IRs were offered to the licensee startup organization for consideration. Some of the discrepancies noted by these IRs were simply observations of conditions associated with in-process work being performed by the startup organization. The IRs were not a part of the startup organization's quality assurance program, and the startup engineers apparently ignored those which related to such in-process work. For other matters, the startup engineers translated the information on the IR into the rework or repair control document (Startup Deficiency Report) prescribed by the startup organization's quality assurance manual. The CAT considered that the IRs related to in-process work should have had some sort of documentation of the startup engineer's decision. However, the CAT did not identify any IR items which were actual deficiencies and which were overlooked by the startup organization. The licensee responded to the CAT concern by reexamining all such IRs that had not been dispositioned by the startup organization by instituting a procedure to document actions on such IRs in the future. These actions were subsequently reviewed by an NRC inspector and determined to be satisfactory.

The startup organization at WNP-2 has performed satisfactorily. In those instances where deficiencies were identified, appropriate corrective actions were taken.

In summary, the NRC staff does not agree with the various conclusions reached by the Petitioner concerning the licensee's quality assurance program. The NRC staff has been fully aware of the items identified by the Petitioner; it was the NRC staff who identified and reported these items. It was also the NRC staff who tracked these items to completion to assure sufficient corrective action. For most of the cases, it was the

NRC staff member who originally identified the item who was also involved in the assessment of the resolution of the issue involved. The NRC inspectors had access to all detailed records, personnel and physical hardware to aid in their assessments.

## II. FAILURE TO MEET GENERAL DESIGN CRITERIA

A second issue in the Petition concerns conformance of the facility with 10 C.F.R. Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The Petitioner alleges nonconformance with a variety of the General Design Criteria in such areas as electrical circuits, structures, and fluid systems. The Petitioner alleges that these nonconformances make operation of the facility an unacceptable risk. The Petitioner further asserts the need for an Independent Design Verification Program (IDVP) to be completed prior to the licensing of operation of the WNP-2 facility. The staff has recognized the need for such a program at the WNP-2 facility and indeed such a program has been undertaken and completed by WPPSS.

At the time of my December 20, 1983 letter to the Petitioner, the NRC staff's preliminary review of the IDVP had been completed. The NRC staff had not completed its detailed review of the licensee's report and the Technical Audit Associate's reports of the IDVP effort. The Staff has now completed its review of these reports and has confirmed its conclusions that there are no indications of significant deficiencies in the WNP-2 design process. The IDVP that was conducted provides additional confidence that acceptable quality assurance design practices were followed at WNP-2 including the ability to correctly translate applicable NRC design criteria into plant design drawings and specifications.

With respect to the specific references in the Petition concerning areas where the General Design Criteria have allegedly not been met, specifically the electrical circuits, structures and fluid systems, the NRC has been aware of each of the items of concern. As a routine NRC program function, the licensee's progress towards resolution of each item has been tracked by the NRC to ensure complete and acceptable corrective action. Some of these matters have been discussed above. For most of the specific items discussed in the Petition, the individual NRC staff member who originally identified the item was also involved in the assessment of the licensee's resolution of that issue. Consequently, the NRC staff is reasonably assured that each of the items referenced in the Petition, including those having to do specifically with WNP-2 plant electrical circuits, structures and fluid systems, have been resolved.

As stated in my letter to the Petitioner of December 20, 1983, the Office of Nuclear Reactor Regulation (NRR) has completed a thorough review of the WNP-2 facility, including its conformance to the General Design Criteria, and has concluded that the facility, as constructed, meets these criteria.

## III. THE SANDLER AFFIDAVIT

A third issue raised by the Petition relates to allegations made by Mr. Stewart Sandler, in an affidavit attached to the Petition, concerning lack of quality construction and effective quality assurance at the WNP-2 site, principally in the area of welding. The Sandler Affidavit has been evaluated by the NRC staff. This evaluation is attached as Appendix A (unpublished). The results of the evaluation show that all issues raised by the affidavit have been satisfactorily resolved.

## IV. WPPSS MANAGEMENT

The fourth issue raised by the Petitioner concerns the competence of WPPSS management to properly operate the WNP-2 facility. The Petitioner alleges that WPPSS management has failed to maintain an adequate quality assurance program to ensure that design and construction of WNP-2 has met applicable requirements. The Petitioner refers to a variety of sources including SALP and CAT findings as supporting the Coalition's position that the WPPSS management is not qualified to operate the WNP-2 facility.

Contrary to the above, the NRC staff has found that the licensee has been responsive to NRC concerns regarding management weaknesses, particularly since issuance of the Civil Penalty and 10 C.F.R. § 50.54(f) request in June 1980. Further, the NRC staff has reviewed the licensee's managerial qualifications during the operating license review and concluded that licensee management is qualified and competent to manage the WNP-2 facility for operations. Specific findings pertaining to the contentions raised in the Petition are discussed below.

Over the 1978 to 1980 time period, the NRC staff identified licensee management difficulties in obtaining effective implementation of quality assurance programs by the various site contractors. Inspection findings and management meetings resulted in various corrective actions during this period, and eventually led to the stop-work decision by the licensee and definition of a site-wide corrective action program in July 1980. A new management team was brought in to supplement and/or replace those individuals who had been ineffective in controlling the project. Included in the program was the termination of some contractors. The licensee's management demonstrated a commitment to assess the adequacy of prior work and the work methods for future work. This included the introduction of significant personnel resources in the form of a restart task force in 1980 and a new construction completion and construction management contractor (Bechtel) in early 1981. The resources and experience of the Bechtel organization strengthened the management team and the corrective action programs coincident with completion of construction of the plant.

The NRC staff has reviewed the licensee's and Bechtel's efforts to assure that all quality deficiencies have been identified and addressed. In addition to programs of records reviews and hardware reinspections, program revisions were initiated to assure that discrepant conditions were documented for proper corrective action. The licensee's management also established a telephone hotline for employees to openly or anonymously report observed quality discrepancies. Also, employee exit-interviews included inquiries regarding knowledge of quality problems.

The licensee's independent consultant (TAA) took note of the current management's improved attitude towards quality in its September 1983 final report.[11] Acceptable cooperation of licensee management was also demonstrated during the NRC CAT inspection.

The Petitioner notes that, in 1982, an NRC inspector was informed by the ASME authorized nuclear inspector (ANI) that Bechtel was not properly implementing its quality assurance program in several areas, including training of crafts, availability of work procedures, departures from work procedures, and insufficient material identification and segregation. The NRC in its review of this matter found that Bechtel construction management had requested the ANI to first address issues to management in meetings or by other informal means and then document his concerns if warranted. Such a request by Bechtel was questionable in the view of the NRC staff in that it could reduce the effectiveness of the quality assurance program. Corrective actions by Bechtel included resolution of the ASME and the NRC inspector's specific concerns and implementation of routine documentation by the Bechtel quality assurance department. Contrary to the allegation in the Petition that this item remains open, the item was closed in NRC Inspection Report 50-397/82-14.

[11] "An Independent Evaluation of the Quality Verification Program and Quality Control Effectiveness," Vol. I, p. 9, September 1983.

The NRC staff considers that licensee and Bechtel management have now demonstrated a sense of responsibility for the establishment and implementation of the quality assurance program and associated compliance with NRC regulations.

The CAT did identify several issues related to management competence in its audit of the WNP-2 facility. NRC Region V issued a Notice of Violation on August 30, 1983 containing six items of noncompliance regarding these issues. Region V referred several of the items to the Office of Nuclear Reactor Regulation for evaluation to assure that corrective action had a sound technical basis. Both NRR and the Region V staff coordinated further reviews with the CAT staff. This included two follow-up inspections by CAT inspectors who had been involved in the original inspection. All of the CAT issues have been satisfactorily resolved. While the CAT perceived that identified hardware deficiencies required increased management attention to assure prompt satisfactory resolution, the CAT did not perceive these deficiencies to represent a pervasive management breakdown.

Finally, the technical and management competence of WPPSS to operate the WNP-2 facility has been reviewed by the NRC staff in accordance with the requirements of 10 C.F.R. § 50.40(b) and the Standard Review Plan (NUREG-0800), Section 13.1. The results are reported in the WNP-2 SER[12] issued in March 1982. The organizational changes made by WPPSS have also been reviewed and are reported in Supplements Nos. 1 and 3 of the SER issued in August 1982 and May 1983, respectively. The NRC staff has concluded that the licensee has complied with all appropriate Commission requirements in the area of management competence and is qualified to operate the WNP-2 facility.

## V. CONDUCT OF NRC PERSONNEL

The fifth and final issue raised in the Petition questions the propriety of the conduct of NRC personnel in their review of matters related to the WNP-2 facility. Paragraphs 51, 71, and 94 of the Petition allege a lack of decisive actions on the part of Region V to ensure that WPPSS met commitments and regulatory requirements.[13] In Paragraph 95, the Petitioner alleges NRC "improprieties" including informal release of information to licensees and further alleges that the NRC Office of Inspec-

[12] *See* note 6, *supra*.

[13] Paragraph 83 alleges that an inspection item concerning implementation of the WPPSS Quality Assurance Program has remained open for an extended period of time. This matter is discussed at p. 920, *supra*.

tor and Auditor (OIA) did not thoroughly investigate the "improprieties."

The Petition has been referred to the NRC's Office of Inspector and Auditor for review and consideration to determine whether any improper conduct occurred on the part of NRC personnel. The Office of Inspector and Auditor has reviewed the Petition and believes that no action by OIA in response to the Petition is warranted at this time.[14]

Based upon my review of the extensive inspection and enforcement effort conducted by Region V, and by the Office of Inspection and Enforcement, I am convinced that these efforts form an adequate basis upon which to make determinations regarding the possible existence of any health and safety concerns raised by the allegations as contained in the Petition. While the Petitioner asserts that the CAT findings show that Region V has not vigorously pursued its inspection and enforcement responsibilities, I find the opposite to be true. The CAT's findings of no pervasive breakdown in the quality assurance activities at the WNP-2 facility confirm that Region V has been effective in overseeing the response of WPPSS to the earlier quality program breakdown to reduce construction errors to an acceptable level.

## Conclusion

The Petitioner argues at length that the circumstances identified by the Petition warrant the exercise of this agency's discretion to issue to WPPSS an order pursuant to 10 C.F.R: § 2.202(a) to show cause why the construction permit for the WPPSS Nuclear Project No. 2 should not be revoked, a stay of construction imposed, the pending application for an operating license denied and a proceeding initiated before the Atomic Safety and Licensing Board.[15] An order to show cause is appropriate in those instances in which the NRC concludes, based upon alleged violations by the licensee or potentially hazardous conditions or other facts, that enforcement action should be taken but that a basis could reasonably exist for not taking the enforcement action proposed. *See* 10 C.F.R. § 2.202(a)(1) and the "General Policy and Procedures for NRC Enforcement Actions," 10 C.F.R. Part 2, Appendix C, § IV. The information provided by the Petitioner is, in almost all instances, derived from the

[14] OIA memo to Director, Office of Inspection and Enforcement, dated January 6, 1984.

[15] Given the issuance of an operating license to WPPSS for the WNP-2 facility, much of the relief sought by the Petitioner is moot. However, had the Petitioner identified deficiencies warranting action such as suspension, modification or revocation of the operating license, such actions would have been taken.

results of NRC inspection activities. The various deficiencies raised by the Petitioner, to the extent that they exist, have been satisfactorily addressed by WPPSS either through its response and corrective action to specific violations or in support of its application for an operating license. In those instances where allegedly new information has been provided by the Petitioner, *e.g.*, the Sandler Affidavit, the NRC staff had already been generally aware of those allegations, had examined them, and had found them to be without merit.

Sufficient grounds must be present for the NRC to institute a show-cause proceeding. The Petitioner, as discussed above, fails to state a sufficient basis for the institution of show-cause proceedings. The standard to be applied in determining whether to issue a show-cause order is whether substantial health or safety issues have been raised.[16] In this instance, both the NRC inspection program and the licensing process have resulted in a careful review of the design and construction of the WPPSS facility. This process culminated in the completion of a satisfactory IDVP program at the WNP-2 facility. Given the substantial basis for a finding that the public health and safety will be reasonably assured following operation of the WNP-2 facility, I decline to institute a show-cause proceeding.

Accordingly, Petitioner's request for action pursuant to 10 C.F.R. § 2.206 has been denied as described in this decision. As provided by 10 C.F.R. § 2.206(c), a copy of this decision will be filed with the Secretary for the Commission's review.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 19th day of March 1984.

[Appendix A has been omitted from this publication but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

[16] *Consolidated Edison Co. of New York* (Indian Point, Units 1, 2 and 3), CLI-75-8, 2 NRC 173, 176 (1975).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

**OFFICE OF INSPECTION AND ENFORCEMENT**

**Richard C. DeYoung**, Director

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| **In the Matter of** | **Docket No. 50-275** |
| | **(10 C.F.R. § 2.206)** |
| **PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Unit 1)** | **March 26, 1984** |

The Director of the Office of Inspection and Enforcement denies a petition under 10 C.F.R. § 2.206 filed by the joint intervenors in the Diablo Canyon operating license proceeding. The joint intervenors contended that the low-power license for Diablo Canyon Unit 1 should be revoked or at least remain suspended on the basis of the licensee's failure to report a 1977 audit of the quality assurance program of the licensee's prime piping contractor. Although the Director finds that the failure to report the audit constituted a material false statement under the Atomic Energy Act, the Director did not find revocation or suspension of the license to be an appropriate remedy for the reporting failure.

## REPORTING OBLIGATIONS: 10 C.F.R. § 50.55(e)

Section 50.55(e) does not require the reporting of every design or construction deficiency, but requires holders of construction permits to evaluate identified deficiencies and report significant deficiencies as defined by the regulation.

### ATOMIC ENERGY ACT: MATERIAL FALSE STATEMENTS

The licensee is found to have made a material false statement by not reporting an audit of its prime piping contractor's quality assurance program where quality assurance was an issue being heard in the operating license proceeding and the audit on its face appeared to contradict the licensee's testimony in the proceeding.

### ATOMIC ENERGY ACT: MATERIAL FALSE STATEMENTS

The fact that an item is not reportable under 10 C.F.R. § 50.55(e) may not obviate reporting under the "full disclosure" standards of section 186 of the Atomic Energy Act.

### NRC ENFORCEMENT POLICY

Not every violation of Commission requirements mandates the severe sanction of license revocation. The choice of sanctions for violations of NRC requirements rests within the sound discretion of the Commission.

### NRC ENFORCEMENT POLICY: MATERIAL FALSE STATEMENTS

In view of the minimal significance of the material false statement (*i.e.*, failure to report) here, and upon consideration of enforcement actions for other material false statements, a Notice of Violation is the most appropriate enforcement action for the failure to report the quality assurance audit.

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

On October 20, 1983, counsel for the joint intervenors in the Diablo Canyon operating license proceeding filed a request before the Commission to revoke the low-power license for Unit 1 of the Diablo Canyon Nuclear Power Plant or, alternatively, to continue the suspension of the license. The joint intervenors' request rests on the alleged failure of the Pacific Gas and Electric Company (PG&E or the licensee) to report the existence of a 1977 audit performed by Nuclear Services Corporation (NSC) of Pullman Power Products' quality assurance program. Pullman is the principal piping contractor for the Diablo Canyon Plant. PG&E opposed the joint intervenors' request in an answer dated October 25, 1983. On November 8, 1983, the Commission rescinded in part its prior suspension of Facility Operating License No. DPR-76 so as to permit fuel loading and pre-criticality testing at Unit 1. CLI-83-27, 18 NRC 1146 (1983). In its Memorandum and Order partially reinstating the low-power license, the Commission referred the joint intervenors' request to the Staff for consideration under 10 C.F.R. § 2.206.

Upon consideration of the joint intervenors' request and other relevant information, the Staff agrees that PG&E should have reported the NSC audit. However, for the reasons set forth in this decision, the Staff does not believe that the extreme remedy of either license suspension or revocation is warranted under these circumstances.

## FACTUAL BACKGROUND

In July 1977, PG&E requested Pullman to obtain an independent audit of Pullman's work at Diablo Canyon. PG&E concurred in Pullman's selection of NSC to perform the audit.[1] NSC conducted the audit between August 22 and September 20, 1977, and sent its report to Pullman on October 24, 1977. In its summary of its report, NSC found "little evidence available to verify the adequacy of the work performed" before early 1974; it concluded that, though documentation was available increasingly since early 1974, "the present program and controls still do not meet 10 C.F.R. 50, Appendix B requirements" for the reasons identified in the report.[2] Upon its review of the NSC audit, Pullman disagreed with NSC's conclusion that necessary documentation did not exist for pre-1974 work. Pullman noted that NSC had failed to examine installed work and had misapplied the applicable codes and regulatory criteria. Pullman asserted that it met Appendix B requirements, but indicated that the audit results were useful in identifying areas in which the quality assurance program could be upgraded.[3] Pullman submitted the final report of its review of the NSC audit to PG&E on April 11, 1978.

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[1] *See* Affidavit of Russell P. Wischow at 1-2, attached to PG&E's Answer to Joint Intervenors' Supplement to Motion to Reopen the Record on the Issue of Construction Quality Assurance (Sept. 21, 1983), which was filed with the Atomic Safety and Licensing Appeal Board.

[2] NSC Audit at 42. The NSC audit and the related Pullman and PG&E reports are attached to the PG&E filing referenced in note 1, *supra*.

[3] Pullman Report, section 4, "Observations," and section 5, "Summary."

Apparently, PG&E did not receive a copy of the NSC audit until February 1978 when Pullman provided a draft of its review of the audit and the NSC audit report to PG&E.[4] PG&E reviewed the NSC audit and also performed an audit of Pullman's installation work. This audit by PG&E was conducted from April 2 through June 1, 1978, and resulted in a report to J.D. Worthington, PG&E Executive Vice President, on June 13, 1978, and a separate report to R.S. Bain, PG&E Manager of Station Construction, on June 16, 1978. While PG&E concluded that, contrary to the NSC audit's findings, Pullman essentially met applicable requirements, PG&E opened two nonconformance reports and four minor variation reports to initiate corrective actions as the result of its review. PG&E generally agreed with Pullman's assessment of the failings of the NSC audit.

At the time that the NSC audit was conducted and was being reviewed by Pullman and PG&E, the Atomic Safety and Licensing Board, on its own initiative, was considering the issue of quality assurance in the Diablo Canyon operating license proceeding. On May 25, 1977, the Board denied the joint intervenors' motion of April 29, 1977, to add a quality assurance contention to the proceeding. At the same time, the Board directed PG&E and the Staff to present evidence on the quality assurance program for Diablo Canyon. The hearing was conducted on October 18 and 19, 1977. Russell Wischow, the Director of the PG&E Quality Assurance Department, testified for PG&E. A panel of three witnesses from NRC's Region V office and the Office of Nuclear Reactor Regulation testified for the Staff. Mr. Wischow described the quality assurance program and testified that the program had generally been effective in detecting defects and in ensuring their correction. The Staff testified that implementation of the Diablo Canyon quality assurance program had been adequate. Counsel for the joint intervenors declined to cross-examine either Mr. Wischow or the Staff's witnesses. PG&E filed its proposed findings of fact and conclusions of law on the quality assurance issue on November 11, 1977, in which PG&E asserted that its quality assurance program had uncovered and then had corrected defects in construction and that its quality assurance program for design and construction of the plant was acceptable. The joint intervenors opposed those findings on February 28, 1978. PG&E replied to the joint intervenors' opposition on March 14, 1978, and reiterated its view that the quality assurance program was acceptable. The Staff filed its proposed findings on March 17, 1978. The Board issued its decision on quality

[4] Affidavit of Russell P. Wischow, *supra* note 1, at 2-3.

assurance in a "Partial Initial Decision" in 1981. The Board found that the quality assurance program for the design and construction phase of Diablo Canyon complied with 10 C.F.R. Part 50, Appendix B, and that implementation had been acceptable. LBP-81-21, 14 NRC 107, 116 (1981).[5]

## REPORTABILITY OF THE NSC AUDIT

The basic issue raised by the joint intervenors' request is whether PG&E had an obligation to report the NSC audit. Reporting obligations to the Commission may arise from various sources, *e.g.*, license conditions, regulations such as 10 C.F.R. Part 21 and 10 C.F.R. § 50.55(e), and section 186 of the Atomic Energy Act.[6] The joint intervenors contend that, by failing to report the NSC audit, PG&E violated 10 C.F.R. § 50.55(e) and committed a material false statement under section 186 of the Atomic Energy Act.

### A. Reportability of the NSC Audit Under 10 C.F.R. § 50.55(e)

The joint intervenors believe that the NSC audit was reportable under 10 C.F.R. § 50.55(e) because the audit revealed a breakdown in Pullman's and PG&E's quality assurance programs. Under 10 C.F.R. § 50.55(e)(1), the holder of a construction permit is required to:

> notify the Commission of each deficiency found in design and construction, which, were it to have remained uncorrected, could have affected adversely the safety of operations of the nuclear power plant at any time throughout the expected lifetime of the plant, and which represents:
> (i) A significant breakdown in any portion of the quality assurance program conducted in accordance with the requirements of Appendix B to this part....

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[5] In November 1981, shortly after issuance of a low-power license for Unit 1, the Commission suspended the license in view of the discovery of deficiencies involving quality assurance for design activities. CLI-81-30, 14 NRC 950 (1981). The Appeal Board reopened the operating license record on design quality assurance by Memorandum and Order dated April 21, 1983 (unpublished).

[6] The NSC audit may also have been reportable under PG&E's responsibility to make appropriate Board notifications. Since the Appeal Board's decision in *Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-143, 6 AEC 623, 625-26 (1973), parties to NRC adjudicatory proceedings have been held to an absolute obligation to alert NRC adjudicatory tribunals to new information that is relevant and material to the matters being adjudicated. *See also Tennessee Valley Authority* (Browns Ferry Nuclear Plant, Units 1, 2 and 3), ALAB-677, 15 NRC 1387, 1394 (1982). The enforcement of the obligation to make Board notifications is within the purview of the Commission's adjudicatory tribunals. The Staff itself is responsible to ensure that new, relevant and material information of which the Staff becomes aware is provided to the Boards and parties.

This regulation does not require the reporting of every deficiency in design or construction that could ultimately affect the safety of plant operations. Rather, the deficiency must be significant. The licensee must evaluate identified deficiencies in design and construction to determine whether a particular deficiency is significant. In determining whether the deficiency represents a significant breakdown in the quality assurance program or one of the three other types of significant deficiencies under § 50.55(e), the regulation permits the licensee reasonable latitude in determining a deficiency's significance.

PG&E evaluated the NSC audit and Pullman's response and concluded that Pullman's quality assurance program generally met applicable requirements. PG&E initiated its review after receiving the NSC audit with Pullman's own review of it. Pullman had reviewed the audit and determined that the findings did not substantiate major deficiencies in Pullman's quality assurance program. Pullman also noted that NSC had not reviewed or identified any hardware or installation deficiencies in Pullman's work, though such a review had been intended to be within the scope of the NSC audit. PG&E reviewed the NSC audit and Pullman's response and also audited the as-built condition of components and supports fabricated and installed by Pullman. PG&E concluded that Pullman's response to the NSC audit was generally correct. As a result of its findings, PG&E opened two nonconformance reports and four minor variation reports to ensure corrective action for identified deficiencies in the programmatic description of the quality assurance program and in the implementation of procedures and for several minor installation deficiencies. PG&E did not conclude in its report that the identified deficiencies were "significant" or that Pullman's quality assurance program had suffered a "significant breakdown."

In recent months the Staff has reviewed the findings of the NSC audit and has examined extensively those findings that would affect the quality of installed hardware.[7] The Staff examined Pullman's records and procedures and the licensee's audits of Pullman's activities. The Staff also interviewed various Pullman personnel, particularly those with experience at the site in the early 1970's. *See* NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Supp. No. 21, at 2-157 (December 1983); NRC Region

[7] No one on the Staff recalls specifically whether the NSC audit was reviewed by NRC inspectors in 1977 or 1978. However, an inspector may have seen the audit or at least PG&E's report of its review of the NSC audit during a July 1978 inspection. The inspection report only indicates that NRC examined a number of PG&E quality assurance audits performed between May 25 and July 6, 1978, the same time frame within which the PG&E review of the NSC audit was issued. *See* NRC Region V Inspection Reports Nos. 50-275/78-10 and 50-323/78-10, at 10 (July 25-26, 1978), attached to the PG&E filing referenced in note 1, *supra*.

V Inspection Reports Nos. 50-275/83-34 and 50-323/83-24; 50-275/-83-37 and 50-323/83-25. The Staff did not identify any significant breakdown in Pullman's quality assurance program or safety concerns with the installed hardware.[8] Additionally, an NRC contractor reviewed some 100 radiographs, independently measured weld attributes, and examined records of Pullman's work. The NRC contractor's review did not establish the existence of welding problems.

Although the timeliness of its evaluation could have been improved, PG&E's failure to make a report under § 50.55(e) does not appear unreasonable. Based on the Staff's review of the NSC audit, Pullman's response, PG&E's review, and pertinent inspection reports during the period, the Staff does not believe that the Pullman quality assurance program suffered a significant breakdown in 1977 such that PG&E was obliged to submit a report under § 50.55(e).[9]

### B. Reportability of the NSC Audit Under Section 186 of the Atomic Energy Act

Apart from 10 C.F.R. § 50.55(e), PG&E may have had an obligation to report the NSC audit under the "full disclosure" doctrine that has developed in NRC case law interpreting section 186 of the Atomic Energy Act. In holding that an omission of material information could constitute a material false statement under section 186 of the Atomic Energy Act, the Commission has imposed an obligation on licensees and applicants to ensure that relevant and material information is promptly furnished to the Commission. *Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC 480 (1976), *aff'd sub nom. Virginia Electric and Power Co. v. NRC,* 571 F.2d 1289 (4th Cir. 1978).

Materiality of an omission or statement depends on "the context in which information appears and the stage of the licensing process involved" and "whether information has a natural tendency or capability

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[8] In response to the joint intervenor's supplemental motion to reopen the record on construction quality assurance before the Atomic Safety and Licensing Appeal Board, the Staff has also taken the position that the NSC audit report did not reveal a major breakdown in the Pullman quality assurance program. *See* NRC Staff's Response to Joint Intervenors' Supplement to Motion to Reopen the Record on Construction Quality Assurance (Oct. 6, 1983) and attached Affidavit of Gonzalo H. Hernandez, Jr. (Oct. 4, 1983). The Appeal Board denied the joint intervenors' motion to reopen on October 24, 1983. In its Memorandum and Order issued on December 19, 1983, which details the rationale for its decision, the Appeal Board stated, "[w]e have carefully reviewed the NSC audit report and the responses of Pullman and the applicant. These lead us to conclude that the deficiencies identified by NSC in 1977 did not evidence a significant or systematic failure of the quality assurance program." ALAB-756, 18 NRC 1340, 1354 n.35 (1983).

[9] In view of PG&E's findings regarding the NSC audit's results, reporting under 10 C.F.R. Part 21 would not have been required since neither a defect nor noncompliance was present that could create a substantial safety hazard.

to influence a reasonable agency expert." *Id.*, 4 NRC at 491. Put another way, "materiality should be judged by whether a reasonable staff member should consider the information in question in doing his job." *Id.* at 486. The Commission has noted that "[a]t the hearing stage . . . where agency decisionmaking is imminent, arguably relevant data must be promptly furnished if the agency is to perform its function." *Id.* at 488. Intent to mislead or to withhold information is *not* a prerequisite to the finding of a material false statement under section 186.[10]

Here, PG&E had an obligation to submit the NSC audit to the Commission before it had reached the conclusion that the NSC audit had not revealed significant deficiencies in Pullman's quality assurance program because of the apparent conflict with PG&E's quality assurance testimony.[11] By not reporting the NSC audit, PG&E committed a material false statement by omission. The obligation to report the NSC audit arose primarily because the Board had held a hearing to develop a record on quality assurance in the operating license proceeding. Although the Board had determined *sua sponte* to receive evidence on quality assurance, that fact did not absolve the licensee from any reporting obligation.

One can only speculate about the specific actions that would have been prompted if PG&E had reported the NSC audit; however, the NSC audit would likely have had some influence on the Board's and the Staff's examination of the quality assurance issue. PG&E had testified on October 18, 1977 that its quality assurance program, including that of its contractors, was sufficient to ensure adequate design and construction of the Diablo Canyon Plant. Within a few days of the hearing, Pullman, PG&E's prime piping contractor, received the NSC audit report which on its face suggested serious inadequacies in Pullman's quality assurance program. Thus, the audit's findings appeared to conflict with the testimony of PG&E which portrayed an adequate, effective quality assurance program. Given the interest of the Board in establishing a record on quality assurance, the Board may well have kept open the record until evidence was received on the validity and significance of the NSC

[10] *See North Anna, supra,* 4 NRC at 486-87. However, the degree of carelessness or intent in failing to provide material information is a pertinent consideration in determining whether and what enforcement action is appropriate for a given material false statement.

[11] This may be an instance in which the failure to provide information would constitute both a failure to meet the obligation to keep the adjudicatory boards informed and a material false statement by omission. Although the obligations are derived from different sources, the obligations under the Board notification policy and under section 186 are very similar. Moreover, two of the omissions for which the applicant was held liable in *North Anna* were based upon the applicant's failure to adduce evidence before the Licensing Board. *See Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), LBP-75-54, 2 NRC 498, 532-33 (1975).

audit's findings. Furthermore, the Staff would likely have followed PG&E's review and resolution of the audit's findings.

Although PG&E determined ultimately that the NSC audit had not in fact detected a significant quality assurance breakdown, PG&E did not make that determination until June 1978. Prior to June, the parties had filed proposed findings on quality assurance, and no decision had been rendered by the Board on the quality assurance issue. Given the pendency of the quality assurance issue, PG&E should have provided the NSC audit to the Commission. The audit was reportable *not* because it was an audit, but because the audit report appeared to contain more significant findings than might be expected of a typical audit. These findings appeared to contradict the record developed in the operating license proceeding and, most likely, would have resulted in follow-up review by the Staff to ensure proper resolution of the audit's findings.

PG&E apparently did not have the NSC audit until February 1978, when PG&E received the audit with Pullman's draft review indicating that NSC's conclusions were generally invalid. This fact does not absolve PG&E from any reporting responsibility. Pullman obtained a copy of the audit in October 1977. In *North Anna,* the Commission held that the applicant or licensee is chargeable with the knowledge of information in the possession of its contractors and consultants. *See North Anna, supra,* CLI-76-22, 4 NRC at 486; LBP-75-54, 2 NRC 498, 504-06, 523 (1975); *cf.* 10 C.F.R. Part 50, Appendix B, Criterion I; *Atlantic Research Corp.,* CLI-80-7, 11 NRC 413, 421-22, 424 (1980). In any event, PG&E received the NSC audit in February 1978 with Pullman's draft response. Although PG&E would ultimately determine that the NSC audit did not reveal significant quality assurance deficiencies, PG&E should have reported the NSC audit when PG&E received it, rather than waited to complete its review. At best, the status of the audit was indeterminate when PG&E received it, but, in light of the Commission's interest in the quality assurance issue and the potential conflict between PG&E's earlier testimony and the audit's findings, PG&E should have submitted the NSC audit to the Commission. Reportability under the facts here is a close call. In other cases, licensees and applicants have been expected to provide information during the hearing stage of the licensing process even where its materiality was uncertain.[12] To decide otherwise here would weaken the incentives for licensees to ensure that the Commission is informed of potentially relevant and material information.

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[12] *See North Anna, supra,* LBP-75-54, 2 NRC at 523 and CLI-76-22, 4 NRC at 488. *Compare McGuire, supra,* 6 AEC at 625 n.15, in the context of the obligation to make Board notifications. *See also supra* note 11.

One could argue that, for purposes of reporting construction deficiencies, the Commission has established a specific reporting threshold in 10 C.F.R. § 50.55(e) which requires only the reporting of certain deficiencies. Nonetheless, the Commission has imposed a distinct reporting obligation through the "full disclosure" doctrine developed under section 186 of the Atomic Energy Act. While 10 C.F.R. § 50.55(e) establishes a reporting standard for most instances in which construction deficiencies are identified, licensees have an obligation under section 186 to report information not otherwise reportable under 10 C.F.R. § 50.55(e), particularly where a particular matter is being adjudicated before an NRC tribunal.

Although the Commission and its licensees are more sensitive to reporting issues today, the standards applied in the foregoing analysis were in place in 1977 when the NSC audit was performed. Accordingly, the Staff believes that the NSC audit should have been reported and that the failure to report constitutes a material false statement under section 186 of the Atomic Energy Act.

## ENFORCEMENT ACTION FOR THE REPORTING FAILURE

Having determined that PG&E made a material false statement, the question remains whether any enforcement action should be taken. The joint intervenors would have the Commission revoke the low-power license for Diablo Canyon Unit 1 or continue its suspension.

Not all violations of NRC requirements, including material false statements, warrant the extreme remedy of license revocation or suspension. *See Petition for Emergency and Remedial Action,* CLI-78-6, 7 NRC 400, 405-06 (1978); *Washington Public Power Supply System* (WNP Nos. 4 & 5), DD-82-6, 15 NRC 1761, 1766 n.9 (1982). The choice of enforcement sanctions for violations of NRC requirements rests within the sound discretion of the Commission, based on consideration of such factors as the significance of the underlying violations and the effectiveness of the sanction in securing lasting corrective action. The Commission's current policy on the application of enforcement sanctions is set forth in 10 C.F.R. Part 2, Appendix C.[13] The enforcement policy classi-

[13] At the time PG&E failed to report the NSC audit, the effective enforcement policy was the one issued on December 31, 1974. That policy did not classify material false statements under the categories of "violations," "infractions," and "deficiencies" used to rank the relative severity of violations of NRC requirements. In those instances in which civil penalties were imposed for a material false statement, the amounts of civil penalties were equivalent to the range of penalties imposed for items of noncompliance in the "violation" category. The categories of violations and the schedule of civil penalties for violations are reproduced in *Atlantic Research Corp.*, ALAB-594, 11 NRC 841, 856-59 (1980).

fies different types of violations by their relative severity and describes circumstances in which formal sanctions, including orders, civil penalties, and notices of violation, are appropriate.

The severity categories for violations involving material false statements are addressed in Supplement VII of the current enforcement policy. Applying this guidance to the material false statement at issue here, the Staff would classify PG&E's failure to report the NSC audit as a Severity Level IV violation. Classification at this level is appropriate for two basic reasons. First, the Staff is not aware of any evidence which suggests that the failure to report resulted from a deliberate, calculated effort to conceal or withhold the NSC audit. Thus, the material false statement here does not carry the degree of intent or recklessness which would warrant classification at Severity Levels I or II. Second, the failure to report — though material — is not significant enough to warrant classification at Severity Level III. Although the Staff would probably have ensured that PG&E or its contractor had evaluated the audit report and had initiated appropriate corrective actions as might be required, the NSC audit would not have changed the Staff's position at the time on quality assurance because ultimately PG&E concluded and the Staff agreed that the NSC audit did not identify a significant quality assurance breakdown. In any event, PG&E took appropriate corrective actions without Staff action. In sum, while the NSC audit would have influenced the Staff in the sense that the Staff would have probably sought more information, the NSC audit would not have resulted in a different Staff position on the quality assurance issue.

Third, in comparison with some Severity Level III material false statements in other cases, this violation is less significant. For example, in the Pilgrim case, the licensee represented that it was in compliance with an NRC regulation when it had not, in fact, met the applicable requirement.[14] In Brunswick, the material false statement involved the licensee's inaccurate representation of its corrective actions in response to an NRC Notice of Violation.[15] These two instances are more severe than the material false statement at issue here, particularly in view of the fact that the affirmative representations in those cases were false and were made in response to express Staff requests for information. The Staff has, in another case, applied the Severity Level IV classification to a material false statement which the Staff did not consider significant. *See Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units

[14] *See* NUREG-0940, Vol. 1, Nos. 1 and 2, at I-8 (September 1982) (EA-81-63).

[15] *See* Letter to E.E. Utley, Carolina Power & Light Co., from J.P. O'Reilly, NRC Region II Administrator (EA-83-88; Jan. 10, 1984).

1 & 2), DD-83-17, 18 NRC 1289 (1983). In view of these precedents, the Staff has concluded that Severity Level IV is the appropriate classification for the violation in this case.

In view of the minimal significance of this particular material false statement, license revocation or continued suspension is inappropriate. As noted above, the failure to report the NSC audit does not appear to have been deliberate or willful.[16] Even if this particular instance were considered in conjunction with the material false statement for which PG&E was cited in early 1982, escalation of enforcement sanctions to the level of license revocation or suspension would not be warranted. Moreover, continued suspension or revocation would not appear to be an appropriate remedy here. The material false statement for which PG&E received a Notice of Violation on February 11, 1982 involved an inaccurate characterization of its receipt of draft reports of the seismic reverification program. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-82-1, 15 NRC 225 (1982). The false statement was compounded by the failure of other PG&E officials to correct the false statement although they knew it to be false at the time. This violation, although more significant than the current violation, did not result in suspension or revocation of the license or even in the imposition of a civil penalty.

Furthermore, the material false statement currently under consideration predated by several years the enforcement action taken in 1982. In connection with that enforcement action, PG&E was required to take appropriate corrective actions. In letters dated March 23, April 15 and 28, 1982, PG&E described its corrective action to ensure good communication between PG&E and the NRC to prevent the recurrence of material false statements or similar reporting failures. The Staff would expect such corrective actions to preclude in the future the type of reporting failure involved in the failure to report the NSC audit.

A Notice of Violation pursuant to 10 C.F.R. § 2.201 is an appropriate sanction for a material false statement of the type made here. Civil penalties are not usually imposed for Severity Level IV violations. In view of the circumstances surrounding this violation including its age and minimal safety significance, a civil penalty for the failure to report

[16] For an instance in which the making of deliberately false statements regarding the status of licensed activities led to license revocation, *see* American Testing Laboratories, Inc., Order to Show Cause and Order Temporarily Suspending License, 48 Fed. Reg. 28,371 (1983); Order Revoking License, 48 Fed. Reg. 57,182 (1983).

the NSC audit would serve no remedial purpose and, accordingly, a Notice of Violation at most is the appropriate sanction here.[17]

## CONCLUSION

For the reasons stated in this decision, PG&E committed a material false statement by failing to report the 1977 NSC audit. Because license revocation or the continuation of the suspension of the low-power license is inappropriate for this material false statement, the intervenors' request for such relief is *denied.*

A copy of this decision will be provided to the Commission for possible review in accordance with 10 C.F.R. § 2.206(c). Unless the Commission otherwise directs, the Staff will issue a Notice of Violation regarding this matter after the conclusion of the period within which the Commission may review this decision.

Richard C. DeYoung, Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland,
this 26th day of March 1984.

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[17] Having concluded that a civil penalty is not appropriate in these circumstances, I do not need to reach the question whether imposition of a civil penalty would be barred by the statute of limitations in 28 U.S.C. § 2462.

CLI-84-6 was inadvertently omitted from the March 1984 issuances and not assigned a CLI number until April. Therefore, this order can be found at 19 NRC 975.

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