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NUCLEAR REGULATORY COMMISSION ISSUANCES

January 1984

UNITED STATES NUCLEAR REGULATORY COMMISSION

U.S. NUCLEAR REGULATORY COMMISSION

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NUREG-0750
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NUCLEAR REGULATORY COMMISSION ISSUANCES

January 1984

This report includes the issuances received during the specified period from the Commission (CLI), the Atomic Safety and Licensing Appeal Boards (ALAB), the Atomic Safety and Licensing Boards (LBP), the Administrative Law Judge (ALJ), the Directors' Decisions (DD), and the Denials 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.

U.S. NUCLEAR REGULATORY COMMISSION

Prepared by the Division of Technical Information and Document Control,
Office of Administration, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555
(301/492-8925)

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Victor Gilinsky
Thomas M. Roberts
James K. Asselstine
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COMMISSION

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio Palladino, Chairman
Victor Gilinsky
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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
Thomas M. Roberts
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.

Atomic Safety and Licensing Appeal Boards Issuances

ATOMIC SAFETY AND LICENSING APPEAL PANEL

APPEAL BOARDS

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

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[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]

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.

Cite as 19 NRC 13 (1984) **ALAB-759**

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

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[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

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[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] Sta. 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]

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.

Cite as 19 NRC 26 (1984) **ALAB-760**

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

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[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.

Atomic Safety and Licensing Boards Issuances

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LICENSING BOARDS

Cite as 19 NRC 29 (1984) **LBP-84-1**

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.

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

Cite as 19 NRC 36 (1984) **LBP-84-2**

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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II.D.2.*Continued*

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 othe. 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 av. iled itself of its prerogative to delegate, but failed in its responsibility to assure that its contractors carried out their delegated qu-lity 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 nc .onfidence 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 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

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 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):
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> 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].
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> *DAARE/SAFE Contention 9(a)* (Water Hammer):
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> 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):
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> 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. An 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.

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[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.

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[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 ¼-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

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[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.

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[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]

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[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 2×10^{-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

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[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 sensors 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;

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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:

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[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:

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[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

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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

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[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

[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).

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[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

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[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

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[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

[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, 5×10^{-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 5×10^{-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 reactor. 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 pre-hearing 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, nonqualified 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

[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.

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[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 I 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 allegat··ıs 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 retes·:d 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 looseleaf 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-

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[69] Another reason for excluding bolt torquing from the program was that there was to be an overinspection 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]

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[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,

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[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

[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 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.

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[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

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[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.

I. 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, downhole 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).

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[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:
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> * * * * *
>
> (c) does not address the relative significance of alternative assumptions;
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> * * * * *
>
> (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(a) 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, ha 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

[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

Cite as 19 NRC 282 (1984) LBP-84-3

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

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[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.

[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-

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[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

Cite as 19 NRC 288 (1984) **LBP-84-4**

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

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*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

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[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

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[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 LMFBP Demonstration Plant, 93d Cong., 1st Sess. (Feb. 28 and May 4, 1973).

[13] App. 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

[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

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[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]

[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

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[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

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[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

	Organ	REM Case 1	REM Case 2	REM Case 3	REM 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
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

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[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

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[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

[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-

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[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-

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[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 rea: or 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 1×10^{-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 0.25g 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 m.y.bp).

(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 1.48×10^6 rad/Ci inhaled (= 1.48×10^{-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 0.2×10^{-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/m³)
0-8 hours	1.2×10^{-4}
8-24 hours	8.4×10^{-5}
24-96 hours	3.9×10^{-5}
96-720 hours	1.4×10^{-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/m³)
0-8 hours	1.1×10^{-5}
8-24 hours	1.0×10^{-5}
24-96 hours	8.0×10^{-6}
96-720 hours	5.7×10^{-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.

Cite as 19 NRC 391 (1984) **LBP-84-5**

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.

Cite as 19 NRC 393 (1984) **LBP-84-6**

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), provide

> 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

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.

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[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 firs 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 1×10^{-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. *Warth 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 2×10^{-5} curies per year into Pu-239 would result in the formation of 5×10^{-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 Committees' 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 a[illegible]w 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 20, 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

Directors'
Decisions
Under
10 CFR 2.206

DIRECTORS' DECISIONS

Cite as 19 NRC 471 (1984) DD-84-1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

Richard C. DeYoung, Director

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

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[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.

Cite as 19 NRC 478 (1984) DD-84-2

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)]

Cite as 19 NRC 480 (1984) **DD-84-3**

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.[1]

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

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[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.