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January 1, 1981 — June 30, 1981

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PREFACE

This is the thirteenth volume of issuances (1 - 1140) of the Nuclear Regulatory Commission and its Atomic Safety and Licensing Appeal Boards, Atomic Safety and Licensing Boards, and Administrative Law Judge. It covers the period from January 1, 1981 to June 30, 1981.

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

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

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

This volume is made up of pages from the six monthly issues of the Nuclear Regulatory Commission publication *Nuclear Regulatory Commission Issuances (NRCI)* for this period, arranged in chronological order. Cross references in the text and indexes are to the NRCI page numbers which are the same as the page numbers in this publication.

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

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

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Cite as 13 NRC 1 (1981) **CLI-81-1**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

John F. Ahearne, Chairman
Victor Gilinsky
Joseph M. Hendrie
Peter A. Bradford

In the Matters of **Docket Nos. 50-247**
50-286

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
(Indian Point, Unit No. 2)

POWER AUTHORITY OF THE STATE OF NEW YORK
(Indian Point, Unit No. 3) **January 8, 1981**

The Commission (1) denies the licensees' motion for reconsideration of its previously announced decision to hold a discretionary adjudicatory hearing to consider long term safety issues raised in connection with a petition filed by the Union of Concerned Scientists requesting, *inter alia;* the shut-down of Units 2 and 3 of the Indian Point nuclear facilities; (2) directs that the proceeding be conducted by an Atomic Safety Licensing Board which is to take evidence and recommend findings and conclusions on disputed contentions for final action by the Commission; and (3) issues procedural and substantive guidelines for the conduct of the hearing. The Commission also reaffirms its earlier decision to permit operation of Units 2 and 3 during the adjudicatory hearing, but announces that it will reexamine the validity of that determination with respect to Unit 2 (which currently is shutdown) before permitting that unit to resume operation.

MEMORANDUM AND ORDER

Background[1]

On May 30, 1980, the Commission issued an order establishing a four-pronged approach for resolving the issues raised by the Union of Concerned Scientists' petition regarding the Indian Point nuclear facilities, and by the decision of the Director, Office of Nuclear Reactor Regulation (NRR), granting in part and denying in part that petition. The order announced the Commission's intention to hold a discretionary adjudication for the resolution of safety issues concerning the plants; initiated an informal proceeding for the purpose of defining the questions to be answered in that adjudication, as well as the criteria to be applied; announced the Commission's plan to address the generic question of the operation of nuclear reactors in areas of high population density through a generic proceeding, to be decided at a later date;[2] and directed the Commission's General Counsel and Director, Office of Policy Evaluation, to establish a Task Force to address the question of the status of the reactors during the pendency of the planned adjudication. In this order, we will deal with the issue of interim operation of the Indian Point units during the adjudicatory hearing and will take the steps necessary to initiate that adjudicatory hearing.

Interim Operation

The Commission must decide whether the Indian Point Units 2 and 3 should continue to operate on an interim basis during the time it takes to complete the adjudicatory hearing we order today. A decision on interim operation is not a decision about the long-term safety of the Indian Point plants.

In his decision on February 11, 1980, the Director of Nuclear Reactor Regulation found that the interim risk of the continued operation of the Indian Point units did not warrant their shutdown while the matter was being further considered. Additionally, the Task Force, formed to conduct a separate investigation of comparative risks of interim operation, com-

[1]The Commission has received a motion from the Union of Concerned Scientists, dated June 23, 1980, requesting the disqualification of Commissioner Hendrie from participation in this matter. In its *Diablo Canyon* decision (In the Matter of Pacific Gas and Electric, 11 NRC 411 (1980), the Commission, with Commissioner Bradford dissenting, stated that requests for the disqualification of a Commissioner would not be entertained by the Commission as a whole but would be referred to the Commissioner whose disqualification was requested. By memorandum of April 23, 1980, Commissioner Hendrie has denied the request for his disqualification.

[2]By this Order, we direct the NRC staff to prepare, as a matter of high priority, a paper setting for options for addressing this generic issue.

pleted its work in June. The conclusion of the Task Force was that the overall risk of the Indian Point reactors is about the same as the typical reactor on a typical site. The Task Force found that although the Indian Point site was considerably more risky than the average nuclear power plant site because of the density of the surrounding population, the design features of the plants reduced the accident risk from Indian Point by a comparable factor. The report acknowledged, however, that the degree of uncertainty for the design comparison was much greater than for the site comparison. Based upon this report, as well as the Director's previous decision, we concluded on July 15 that the risk posed by the operation of the Indian Point facilities did not warrant the suspension of the operating licenses during the adjudicatory proceedings. The Task Force findings and the Director's findings are not the final judgment on the safety of Indian Point Units 2 and 3. That final judgment may only be made after all parties have had the opportunity to examine in detail the Task Force report and other evidence presented by the NRC staff and present additional evidence of their own. In the event that the Licensing Board conducting the adjudication determines that new evidence warrants interim relief, it may at any time recommend a course of action to the Commission. The Task Force Report itself will be distributed free upon a written request to the NRC.

In making this decision, we considered the positions taken by the many commenters. Certain of those positions warrant specific discussion.

UCS has alleged that there are specific safety defects in the Indian Point units which raise questions about whether or not the units comply with NRC regulations. The Director responded to these allegations in his February 11, 1980 Order and UCS responded in turn in the submittal of March 10, 1980. We believe these specific allegations raise issues which are best resolved in the forthcoming adjudicatory proceedings. We have not made a judgment about these allegations and rely in the interim upon the judgment of the Director of NRR. However, we do note that the Task Force report found no significant difference in risk between the Indian Point 2 and 3 designs. It also found that the technical fixes ordered in the Director's decision would be clearly beneficial in reducing risk, but questioned whether the factor of improvement was significant in light of the uncertainties in estimating overall risk. If the Board at any time during the proceeding believes that any of these issues are serious enough to warrant immediate action, it should make an appropriate recommendation to the Commission.

Several commenters contended that the Commission should not permit continued operation because of the lack of an emergency plan for the surrounding area. While a successful plan for evacuation at Indian Point would probably reduce overall risk, the fact is that most operating reactor sites do not yet have an appoved plan and Indian Point is not different in this regard.

New York PIRG requested that we make no decision on interim operation until Senate confirmation of a new chairman. We cannot delay Commission business pending a confirmation process which is beyond our control. Furthermore, such delay would not make a significant difference in this case since the decision on interim operation was unanimous. New York PIRG also requested that the Commission examine a copy of the FEMA review on the status of state and local emergency planning ordered by the President. We have examined this report and it does not change the opinion on emergency planning we expressed above.

We note that the Governor of New York has strongly urged that the plants remain in operation pending the outcome of the proceeding.

Both UCS and New York PIRG sought to address the Commission orally on the subject of interim operation. By a vote of 2-2, that request was denied.

The recent leaks of large amounts of water into the containment and reactor vessel cavity at Indian Point Unit 2 are still being reviewed by the Commission's Office of Inspection and Enforcement. On November 14, 1980, the Commission received a briefing on the status of the investigation at Indian Point Unit 2, and on the implications of the problem for Unit 3. Unit 2 is currently shut down, and must remain so for a period of months, for repair of the fan cooler units and refueling. With respect to Unit 2, prior to resumption of operations, the Commission will determine whether its decision of July 15, 1980, to permit continued operation remains valid. With respect to Unit 3, we decided to stand by our earlier determination to allow operation during the pendency of the adjudication. Our judgment is based upon the information received in the November 14 briefing from the Director of the Office of Inspection and Enforcement, who advised that the containment fan cooler units at Unit 3 are in markedly better condition than those which have been the source of problems at Unit 2, and that Unit 3 has additional safety features not present in Unit 2 in this regard.[3] Our judgment also reflects the fact that the two units are owned and operated by separate entities.

Adjudicatory Proceeding

The Commission has received a motion for reconsideration of that portion of the Commission's order dated May 30, 1980 which directs that an adjudicatory hearing be held on the long-term safety of the Indian Point

[3]See Appendix A, "Comparison of Indian Point Units 2 and 3."

units. The basis for the petition is the Task Force's conclusions that Indian Point poses the same overall societal risk and less of an individual risk than a typical reactor on a typical site. The licensees also contend that the population density is not materially dissimilar from numerous other sites not subject to adjudicatory hearings.

We deny the motion for reconsideration. The licensees would have us treat the Task Force report as the final word on the risks of the Indian Point site, instead of a document designed to aid the Commission in its decision on interim operation. As we stated previously in this order, the Task Force report, compiled in a short time period and not disclosing its detailed methodology and underlying data, will be tested in an adjudicatory setting where parties may present additional or rebuttal evidence. Furthermore, the Task Force report, even if perfectly accurate, does not answer all of the questions the Commission wishes explored by the Licensing Board in a full proceeding. In short, we will not turn a decision on interim operation into a final decision on the long-term acceptability on the Indian Point site.

Licensees also contend that the Indian Point demography is not different from other sites. In fact, according to the Task Force report, Indian Point has the highest population within 10, 30 and 50 miles of any nuclear power plant site in the United States. At 50 miles, its population is more than double any other plant site.

The Commission directs that the discretionary proceeding will be conducted in the vicinity of Indian Point by an Atomic Safety and Licensing Board, using the full procedural format of a trial-type adjudication, including discovery and cross-examination.[4] The purpose of the

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[4]Because of the investigative nature of this proceeding, further guidance is necessary with respect to certain procedural matters. Because the proceeding, although adjudicatory in form, is not mandated by the Atomic Energy Act, it is not an "on the record" proceeding within the meaning of the Atomic Energy Act. Although normal *ex parte* constraints will apply to communications to the Licensing Board, the Commission will not be limited in its ability to obtain information with respect to Indian Point from any source. Because the Commission itself is designating by this Order the issues it wishes to be addressed in the adjudication, it is particularly important that the Licensing Board have discretion to formulate contentions and subissues, upon the advice of the parties, so as to effectuate that purpose. In admitting and formulating contentions and subissues, therefore, the Licensing Board will not be bound by the provisions of 10 CFR Part 2. The Licensing Board may also, without regard to the provisions of 10 CFR Part 2 establish whatever order of presentation it deems best suited to the proceeding's investigative purposes. Except as provided above or elsewhere in this Order, 10 CFR Part 2 will control. If the Board concludes that further relaxation of the rules is necessary for the efficient conduct of the hearing, we expect it to request such authorization from the Commission. The Commission expects the Licensing Board to use its authority under Part 2 to assure the relevance and efficiency of discovery and cross-examination. The Licensing Board shall not reach an initial decision, but as noted in the Order, shall instead formulate recommendations on the questions posed by the Commission. No party will have the "burden of persuasion" as the term is normally used in adjudicatory proceedings; if evidence on a particular matter is in equipoise, the Board's recommendation may be expected

proceeding will be to take evidence and make recommended findings and conclusions on disputed issues material to the question whether the Indian Point Units 2 and 3 plants should be shut down or other action taken. The record of the proceeding, together with recommendations, will then be forwarded to the Commission for the final agency action on the merits of the proceeding. In view of the complexity of this proceeding, and in order that the Commission may make its decision within a reasonable period of time, we stress that the Board should focus clearly upon the questions asked by the Commission.

The Commission's primary concern is the extent to which the population around Indian Point affects the risk posed by Indian Point as compared to the spectrum of risks posed by other nuclear plants. The Commission is concerned with both the total risk to persons and property posed by the Indian Point plants and the risk to individuals living in the vicinity of the Indian Point site, including that resulting from the difficulty of evacuation in an emergency. The Commission intends to compare Indian Point to the spectrum of risks from other nuclear power plants, since the primary basis for the Commission's decision will be how extreme are the individual and societal risks associated with Indian Point compared to the spectrum of risks from other operating stations.

The Commission is also interested in the current state of emergency planning in the vicinity of the Indian Point site and in future improvements in that planning as well as in resolving the specific contentions in the UCS Petition to the effect that some of our regulations are not met in one or both units.

Risks from nuclear power reactors are defined by the probabilities and consequences associated with potential accidents. In directing a comparison of the risks of the Indian Point units with those from a representative group of other operating units, the Commission is fully aware of the uncertanties that attend such quantitive risk assessment calculations (reference NUREG-CR-0400, the Lewis Report, and the Commission policy statement on it.) Despite these uncertainties, risk assessment methods offer the best means available for objective and quantitative comparison of the kind needed here. Further, some of the uncertainty that is associated with risk

to reflect that fact. The staff will be a party to the proceeding, and the licensees will be admitted as parties upon request filed within 30 days of Federal Register notice of the appointment of a Licensing Board. All others wishing to intervene shall file petitions for intervention within 30 days of Federal Register notice of the appointment of a Licensing Board. The appointment of the Licensing Board will be announced by subsequent order of the Commission.

assessment estimates of the absolute values of accident probabilities and consequences does not apply to comparisons such as those sought here.

Several measures of risk are useful for the comparisons the Commission seeks. For individual risks, these include the probabilities of early effects — fatalities and injuries that could occur soon after an accident — and of long-term effects — cancers and genetic effects that could occur more than a year after an accident, all as a function of distance from the reactor.

For societal risks the useful measures include early effects, long-term effects, and property damage and costs in terms of interdiction, decontamination, and crop and milk losses and the possibility that some areas affected by an accident might be uninhabitable for long periods. Societal risk measures should include the distributions of probabilities and consequences as well as the expected risks or mean annual values of the consequences. Risk measures of these kinds for the Indian Point units and for a representative group of other operating nuclear power plants were presented in the report of the Commission's Task Force on Interim Operation of the Indian Point, NUREG-0715, and were found useful by the Commission in its consideration of the interim operation matter.

In developing the record of the proceeding, the Board should address a series of questions as follows:

1. What risk may be posed by serious accidents at Indian Point 2 and 3, including accidents not considered in the plants' design basis, pending and after any improvements described in (2) and (4) below?

2. What improvements in the level of safety will result from measures required or referenced in the Director's Order to the licensee, dated February 11, 1980? (A contention by a party that one or more specific safety measures, in addition to those identified or referenced by the Director, should be required as a condition of operation of the facility or facilities, would be within the scope of this inquiry.)

3. What is the current status and degree of conformance with NRC/FEMA guidelines of state and local emergency planning within a 10-mile radius of the site and, of the extent that it is relevant to risks posed by the two plants, beyond a 10-mile radius? In this context, an effort should be made to establish what the minimum number of hours warning for an effective evacuation of a 10-mile quadrant at Indian Point would be. The FEMA position should be taken as a rebuttable presumption for this estimate.

4. What improvements in the level of emergency planning can be expected in the near future, and on what time schedule, and are there other specific offsite emergency procedures that are feasible and should be taken to protect the public?

5. Based on the foregoing, how do the risks posed by Indian Point Units 2 and 3 compare with the range of risks posed by other nuclear power plants licensed to operate by the Commission? (The Board should limit its inquiry to generic examination of the range of risks and not go into any site-specific examination other than for Indian Point itself, except to the extent raised by the Task Force.)

6. What would be the energy, environmental, economic or other consequences of a shutdown of Indian Point Unit 2 and/or Unit 3?

7. Does the Governor of the State of New York wish to express an official position with regard to the long-term operation of the units?

The Commission would like to receive the Board's recommendations no later than one year from this date.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 8th day of January, 1981.

Appendix A — Comparison of Indian Point Units 2 and 3

In the aftermath of the event at Indian Point Unit 2 resulting from containment fan cooler leakage, an analysis was made to determine whether Indian Point Unit 3, which is of a nearly identical design, had any features which would preclude the type of event which had occurred at Indian Point Unit 2. At the time of that event, Indian Point Unit 3 was shut down for maintenance and inspection. The analysis indicated the following:

1. The maintenance history on the containment fan coolers is significantly better at IP-3 compared to IP-2; therefore, major leakage inside containment is much less likely to occur. Although the better condition is probably largely because IP-3 fan coolers are newer, at the present time the fact is they *are* in significantly better condition and are expected to remain so during the upcoming cycle.

   At IP-3, there have been no leaks in the piping associated with the fan coolers (such as the main contributing leak to the IP-2 event in a 10″ service water return pipe). IP-3 has replaced five motor cooler units in their history after experiencing leakages up to approximately 2 gpm maximum from those units.

   Also, there are no "episeal" or "adams clamp" patches on the IP-3 coolers (there are numerous patches of both types on IP-2, some of which have had to be re-repaired). IP-3 has used "hard" solder (90/5/5) to build up a patch over several small leaks. Those patches, while not considered permanent, have proven more satisfactory than the IP-2 method.

   Finally, the fan-cooler service water isolation valves at IP-3 have all been rebuilt even though no recent problems have been experienced, and each fan cooler unit has passed the Technical Specification required 0.36 gpm/cooler leak rate test (this includes all valves, coils, pipes, etc., not just the isolation valves).

2. There are more indications in the control room of the sump levels in containment than there were at IP-2.

   a. The sump pump on/off levels of the vapor containment (VC) sump are adjusted so that five levels lights (three on one column and two on another column) will turn on before water spills onto the 46' elevation floor (as opposed to four at IP-2). Since two are normally on even after the sump pumps have pumped the sump at each plant (the lowest 2 lights) that means 3 additional lights will come on at IP-3 as opposed to 2 at IP-2, before water spills onto the 46' elevation floor.

b. A new capacitive detector device will detect approximately 1″ of water on the 46' floor, with an audible control room alarm.

c. At IP-3, the recirculation sump is normally kept dry so that increasing levels in containment will also be detected by the two *additional* level indicating columns in that sump before water could flow into the reactor cavity (at IP-2, the recirculation sump is kept full of borated water, thereby negating usefulness of these indicators).

One of the two level indicating columns in *each* sump must be operable by Technical Specifications for continued plant operation.

3. Several features are present in the reactor cavity to prevent and detect collection of water there.

a. Two new pumps have been installed which will not operate in such a way as to be subject to trips on thermal overload, as might have been the case with the previous pumps. The pumps have been installed with a "siphon breaker" (3/4″ vacuum relief line in the discharge loop above the 46' floor, where it will discharge into the VC sump).

b. A column has been installed in the cavity that will activate two independent audible alarms in the control room when approximately 1″ and approximately 3″ of water respectively are in the bottom of the cavity.

c. A search has been conducted for other siphon paths into the reactor cavity, resulting in sealing of one conduit connection on the 46' floor which representd a potential siphon path.

d. Two unlabeled lights inside containment that were incorrectly assumed to indicate cavity pump operation (when on) at IP-2 have been properly labeled at IP-3 (they *do* indicate cavity pump operation at IP-3, unlike IP-2 where they indicate moisture in the cavity).

e. The 46' floor has been "surveyed" using a water-filled tygon hose, with the result that water depth on the 46' floor at the sump before water would flow into the cavity would be approximately

5-1/2″ (compared to variously reported 2″ to 4″ at IP-2 due to a reverse slope in the IP-2 46' floor).

4. In addition to Technical Specification requirements already mentioned (0.36 gpm leakage/fan cooler, one float column operable/sump) several plant requirements, some with calibration procedures, exist for equipment important for detection/prevention of "IP-2" type events.

   a. Level switches and the capacitive level indicator must be calibrated by procedure each refueling outage.

   b. Dew point detectors and weir level (containment fan cooler condensate and/or leakage flow detector) must be calibrated every two years.

   c. Plant procedures require each shift recording and supervisory review of trends on the rotometer flow meter/totalizer installed on the line from the VC sumps to tanks outside containment. Changes in that flow would signal leaks in containment (by an increase) or the possibility of pump failure (by a decrease).

B. Long Term

With the above noted exceptions, many of the preventative and mitigative features described above are not defined as "safety-related" and/or they do not have formal operability requirements.

However, IP-3 personnel have been "tuned" to look for this type of event by IE Information Notice 80-37 concerning the IP-2 event, and by extensive discussion with NRC personnel. The NRC staff believes that in the near term, a flooding event at IP-3 is unlikely, and that if it did happen it would be promptly detected and corrected long before consequences become as severe as they did at IP-2.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-466**

HOUSTON LIGHTING AND POWER COMPANY
(Allens Creek Nuclear Generating Station, Unit No. 1) **January 5, 1981**

The Appeal Board dismisses as interlocutory an intervenor's appeal of the Licensing Board's order rescinding any prior orders of that Licensing Board which had granted procedural assistance pursuant to 10 CFR 2.750(c).

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Interlocutory appeals are barred in terms by the Commission's Rules of Practice. 10 CFR 2.730(f); *Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), ALAB-370, 5 NRC 131 (1977).

APPEARANCES

Mr. John F. Doherty, Houston, Texas, intervenor, *pro se.*

MEMORANDUM AND ORDER

1. Last July, the Commission established "a one-year pilot program of procedural assistance in adjudicatory proceedings on applications for licenses and amendments thereto, except for antitrust proceedings, to parties other than the applicant * * *". 45 *Fed. Reg.* 49535 (July 25, 1980).

In the implementation of this program, several of the Rules of Practice were amended. Among other things, a new subsection (c) was added to 10 CFR 2.750, authorizing (except in antitrust proceedings) the supplying "of one free transcript to a party, other than the applicant, upon request by that party". *Id.* at 49537.

On December 3, 1980, the Comptroller General of the United States issued a letter decision (B-200585) in which he concluded that certain portions of the procedural assistance program, including that embodied in 10 CFR 2.750(c), were precluded by Section 502 of the Energy and Water Development Appropriation Act, 1981, Pub. L. No. 96-367, 94 Stat. 1331, 1345.[1] On the strength of this determination, the following day the Chairman of the Commission sent a memorandum to the Secretary and the Executive Director for Operations in which he ordered an immediate halt to the program to allow the General Counsel and the Commission "an opportunity to examine [the] decision and reach a conclusion as to what our future action should be". The Chairman went on to instruct that "[a]ny documents that are in the process of being transmitted should be held and no further processing should occur without further direction from the Commission".

2. On December 2, 1980, the Licensing Board had held a prehearing conference in this construction permit proceeding involving the proposed Allens Creek nuclear facility. One of the participants in the conference had been intervenor John F. Doherty. Together with other intervenors, Mr. Doherty had previously requested and been granted the procedural assistance authorized by 10 CFR 2.750(c).

On December 9, the Licensing Board entered an order in which it (1) called attention to the Comptroller General's ruling and Chairman Ahearne's directive in response thereto; and (2) rescinded, to the extent covered by the ruling and directive, "any previous orders or issuances which adverted to and/or granted procedural assistance to any [i]ntervenors". On December 10, the parties were orally notified of the substance of that order.

Dissatisfied with the termination of his entitlement to receive a copy of the transcript of the December 2 prehearing conference, Mr. Doherty seeks relief from us by way of "appeal". It is his apparent view that he has been retroactively deprived of the vested right to a free transcript which had been conferred upon him by the adoption of Section 2.750(c) last July. We are also told that he had relied on that alleged right to his detriment in that he

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[1]That Act contains the NRC appropriation for FY 1981. Section 502 provides that:

> None of the funds in this Act shall be used to pay the expenses of, or otherwise compensate, parties intervening in regulatory or adjudicatory proceedings funded in this Act.

does not now have any record of what transpired at the prehearing conference.

3. Because, insofar as here relevant, the Licensing Board's December 9 order was entirely interlocutory in character, Mr. Doherty's appeal from it is barred in terms by the Commission's Rules of Practice. 10 CFR 2.730(f); *Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), ALAB-370, 5 NRC 131 (1977), and cases there cited. Although the appellate papers might nonetheless be treated as a petition for directed certification under 10 CFR 2.718(i),[2] Mr. Doherty would not be aided were we to do so.

The Licensing Board manifestly was bound by the immediately effective instruction of the Chairman of the Commission that, pending further Commission directive (and there has been none to date) no additional transcripts of adjudicatory proceedings were to be supplied to parties at public expense. That instruction likewise must be honored by us, as well as by all other components of the Commission (including the Office of the Secretary, which had general responsibility for the administration of the procedural assistance program). Thus, Mr. Doherty has pressed his grievance in the wrong forum.

In these circumstances, we need not undertake to consider Mr. Doherty's thesis that the Commission remained obligated to furnish him with a transcript of the December 2 prehearing conference even after the Comptroller General had authoritatively ruled that such a step would involve the unlawful expenditure of appropriated funds. It is worthy of passing note, however, that a copy of that transcript is available for inspection in the local public document room for the Allens Creek facility located in the Sealy Public Library, Sealy, Texas. That community, in the neighborhood of the Allens Creek site, is approximately 45 miles from the center of Houston, where Mr. Doherty lives.[3] While no doubt he would prefer to have his own personal copy which could be consulted at his convenience, the fact remains that Mr. Doherty has ready access to the transcript at a not prohibitive distance from his residence. Unless and until the Commission determines that it is both legally permissible and desirable to reinstate the provisions of Section 2.750(c), it appears that he will have to take advantage of that access.

Appeal *dismissed.*

It is so ORDERED.

[2] *See Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 482-83 (1975).

[3] Sealy and Houston are connected by a major interstate highway.

FOR THE APPEAL BOARD

Barbara A. Tompkins
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Richard S. Salzman

In the Matter of **Docket Nos. 50-369 OL**
50-370 OL

DUKE POWER COMPANY
(William B. McGuire
Nuclear Station, Units 1
and 2) **January 6, 1981**

Upon review pursuant to 10 CFR Part 2, Appendix B, of the Licensing Board's order of November 25, 1980 authorizing a license for fuel loading, initial criticality and zero power physics testing of Unit 1 of the McGuire facility, the Appeal Board concludes that no stay is warranted and affirms the order to the extent reviewed.

RULES OF PRACTICE: EFFECTIVENESS OF LICENSING BOARD DECISIONS

Except as provided in 10 CFR Part 2, Appendix B, Licensing Board decisions which authorize licensing action become effective only after the Appeal Board and Commission actions outlined in that Appendix have taken place.

MEMORANDUM AND ORDER

1. On November 5, 1979, the Commission amended its Rules of Practice (10 CFR Part 2) by, *inter alia,* the addition of an Appendix B entitled

"Suspension of 10 CFR § 2.764 and Statement of Policy on Conduct of Adjudicatory Proceedings."[1] 44 *Fed. Reg.* 65050 (November 9, 1979). In relevant part, Appendix B provides that Licensing Board decisions "shall not become effective until the Appeal Board and Commission actions outlined [in the Appendix] have taken place." Insofar as the appeal boards are concerned, that action is as follows:

> Within sixty days of the service of any Licensing Board decision that would otherwise authorize licensing action, the Appeal Board shall decide any stay motions that are timely filed. For the purpose of this policy, a "stay" motion is one that seeks to defer the effectiveness of a Licensing Board decision beyond the period necessary for the Appeal Board and Commission action described herein. If no stay papers are filed, the Appeal Board shall, within the same time period (or earlier if possible), analyze the record and decision below on its own motion and decide whether a stay is warranted. It shall not, however, decide that a stay is warranted without giving the affected parties an opportunity to be heard.
>
> In deciding these stay questions, the Appeal Board shall employ the procedures set out in 10 CFR 2.788. However, in addition to the factors set out in 10 CFR 2.788(e), the Board will give particular attention to whether issuance of the license or permit prior to full administrative review may: (1) Create novel safety or environmental issues in light of the Three Mile Island accident; or (2) prejudice review of significant safety or environmental issues. In addition to deciding the stay issue, the Appeal Board will inform the Commission if it believes that the case raises issues on which prompt Commission policy guidance, particularly guidance on possible changes to present Commission regulations and policies, would advance the Board's appellate review. If the Appeal Board is unable to issue a decision within the sixty-day period, it should explain the cause of the delay to the Commission. The Commission shall thereupon either allow the Appeal Board the additional time necessary to complete its task or take other appropriate action, including taking the matter over itself. The running of the sixty-day period shall not operate to make the Licensing Board's decision effective. Unless otherwise ordered by the Commission, the Appeal Board will conduct its normal appellate review of the Licensing Board decision after it has issued its decision on any stay request.

[1]Section 2.764 provides for the immediate effectiveness of initial decisions directing the issuance or amendment of a construction permit or operating license.

Ibid; footnote omitted.

2. On November 25, 1980, the Licensing Board entered an unpublished order in the operating license proceeding involving the McGuire nuclear facility. In that order, the Board acted upon the motion of the applicant for summary disposition with regard to its entitlement to a license for Unit 1 authorizing fuel loading, initial criticality, zero power physics testing and low-power testing. The Board resolved the matter in the applicant's favor as to all of those activities except low-power testing. Respecting that phase, it concluded that a genuine issue of material fact had been raised by the intervenor, Carolina Environmental Study Group.

Although the order does not bear the "decision" label, it does authorize "licensing action" and therefore comes within the ambit of Appendix B. No motion for a stay (or indeed exceptions) having been filed by any party, our task is to review the order and the underlying record on our own initiative to determine whether a stay is nonetheless warranted.

Our examination of the portion of the record pertaining to the motion for summary disposition persuades us that the Board below correctly granted the motion insofar as fuel loading, initial criticality and zero power physics testing are concerned. (We are not called upon to consider, and thus express no opinion on, the Board's denial of the balance of the motion.) Applying the criteria specified by the Commission in its Statement of Policy, we (1) conclude that no stay of the November 25 order is warranted; (2) affirm the order to the extent here reviewed; and (3) advise the Commission that, in our judgment, the order raises no issues requiring its prompt policy guidance.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Dr. W. Reed Johnson

In the Matter of | **Docket No. 50-344 (Control Building)**

PORTLAND GENERAL ELECTRIC COMPANY, *et. al.*
(Trojan Nuclear Plant)

January 6, 1981

The Appeal Board grants a joint motion by the licensee, staff and the State of Oregon requesting that it (1) reopen the record in this proceeding in order to receive a stipulation executed by them and certain documents previously furnished the Board by the staff and (2) modify a license condition to take account of the stipulation. Because the stipulation is dispositive of the single issue presented by the State of Oregon's appeal, the Board dismisses the appeal as moot.

APPEARANCES

Mr. Frank W. Ostrander, Jr., Assistant Attorney General of Oregon, Portland, Oregon, for the State of Oregon.

Messrs. Maurice Axelrad and **Albert V. Carr, Jr.,** Washington, D.C., and **Mr. Ronald W. Johnson,** Portland, Oregon, for the licensee, Portland General Electric Company, *et al.*

Mr. Jospeh R. Gray for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

1. Last July, the Licensing Board issued an initial decision in which it authorized the modification of the Trojan facility's control building to bring it into conformity with prevailing seismic requirements. LBP-80-20, 12 NRC 77. By the addition of a license condition, identified in the decision as "2.C.11" but then incorporated into the Trojan operating license as "2.C.(12)", the Board directed, *inter alia*, that the modification program be carried out in accordance with plans which the licensee[1] had devised and introduced into evidence. 12 NRC at 112. The condition further mandates that any "deviations or changes" from those plans be "accomplished in accordance with the provisions of" 10 CFR 50.59. *Ibid.*

Insofar as here relevant, that Section allows the holder of an operating license to make changes in the facility or procedures "as described in the safety analysis report" without obtaining prior Commission approval so long as neither an alteration of the technical specifications incorporated in the license nor an unreviewed safety question is involved.[2] Such changes must be reported to the Commission on an annual basis or at such shorter intervals as may be specified in the license.[3]

At least by implication, the initial decision thus rejected the assertion of the State of Oregon,[4] embodied in its proposed findings of fact, that the licensee should be required by license condition to report deviations or changes on an accelerated basis.[5] Dissatisfied with this rejection, Oregon simultaneously both moved for reconsideration before the Licensing Board and appealed to us.[6] In an unpublished order entered on September 4, 1980, the motion was denied on the ground that the record did not establish a need for accelerated reporting of "minor changes or deviations undertaken pursuant to Section 50.59". Order, pp. 2-3. Oregon thereupon moved forward with the appeal.

2. On December 18, shortly after the briefing of the Oregon appeal had been completed, the licensee's counsel advised us that the parties had reached agreement on the matter in dispute and that that agreement was

[1]As in the initial decision, the co-owners of the Trojan facility are collectively referred to herein as the "licensee."

[2]Section 50.59(a)(1).

[3]Section 50.59(b).

[4]Oregon has participated in the proceeding under the "interested State" provisions of 10 CFR 2.715(c).

[5]More particularly, Oregon proposed that the deviation or change be reported either prior to its commencement or within 14 days after the licensee initially decided to implement it (depending on the nature of the deviation or change).

[6]In an unpublished July 28, 1980 order, we instructed the Board below to pass upon the reconsideration motion on the merits notwithstanding the pendency of the appeal.

reflected in a stipulation. On December 24, the stipulation — duly executed by counsel for the licensee, the NRC staff and Oregon — was transmitted to us. In an accompanying joint motion, we were asked (1) to reopen the record to receive both the stipulation and certain documents which had been previously furnished to us by the staff on November 24, 1980;[7] and (2) to modify license condition 2.C.(12) to take account of the agreement.

We *grant* the motion in its entirety. Accordingly, the Director of the Office of Nuclear Reactor Regulation is *directed* to amend forthwith the introductory portion of condition 2.C.(12) in the Trojan operating license so as to read as follows:

> (12) **Control Building Modifications.** The Licensee is authorized to and shall proceed with modifications to the Control Building in order to restore substantially the originally intended design margins. The modification program shall be accomplished in accordance with PGE-1020, "Report on Design Modifications for the Trojan Control Building", as revised through Revision No. 4, and as supplemented by PGE Exh. 27 (Licensee's Testimony ("Broehl, *et al*)." on Matters Other Than Structural Adequacy of the Modified Complex, March 17, 1980). Any deviations or changes from the foregoing documents shall be accomplished in accordance with the provisions of 10 CFR part 50.59. Prior to completion of the modification, any reports under this condition required by 10 CFR 50.59(b) shall be made to the NRC for information in accordance with the following schedule:
>
> (i) Any deviations or changes which require or cause the Licensee to perform calculations to ensure compliance with 10 CFR 50.59 shall be reported prior to commencement of the deviations or changes.
>
> (ii) All other deviations or changes shall be reported within fourteen (14) days after the Licensee initially decided to implement them.
>
> (iii) A copy of all reports submitted to the NRC pursuant to 10 CFR 50.59 shall be sent to the Office of Nuclear Reactor Regulation.
>
> The Control Building modification program shall further be subject to the following:

[7]These documents were (1) an October 14, 1980 letter from J.L. Crews, Chief, Reactor Operations and Nuclear Support Branch, Office of Inspection and Enforcement, Region V, to the Portland General Electric Company; and (2) a September 25, 1980 internal Region V memorandum which was enclosed therewith.

Because the foregoing license amendment is fully dispositive of the single issue presented by the Oregon appeal as briefed, that appeal is now moot. It is hereby *dismissed* on that ground.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. W. Reed Johnson

In the Matter of — **Docket Nos. 50-516, 50-517**

LONG ISLAND LIGHTING COMPANY and NEW YORK STATE ELECTRIC & GAS CORPORATION
(Jamesport Nuclear Power Station, Units 1 and 2)

January 15, 1981

On applicant's representation that the Jamesport plant will not be built, the Appeal Board grants applicant's motion to terminate this construction permit proceeding as moot. The Appeal Board also vacates the Licensing Board decisions (LBP-78-17, 7 NRC 826 (1978) and LBP-78-41, 8 NRC 750 (1978)) authorizing construction of the facility, dismisses the pending appeal therefrom, and directs that the outstanding construction permits be revoked.

APPEARANCES

Mr. W. Taylor Revely, III, Richmond, Virginia, for Long Island Lighting Company and New York State Electric & Gas Corporation, **applicants.**

Mr. Bernard M. Bordenick for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

The Licensing Board authorized the Director of Nuclear Reactor Regulation to issue construction permits for the twin-unit Jamesport nuclear facility and he did so on January 4, 1979. New York law, however, requires public utilities to obtain certificates from the State's Board on Electric Generation Siting and the Environment before starting to build. While appeals from the Licensing Board's decisions were pending, the State Siting Board announced orally in January 1980 that it would approve only a single coal-fired plant at the Jamesport site and confirmed that ruling in writing on September 8, 1980.

Although the State Board "has yet to act definitively" on petitions to reconsider, the applicants informed us on December 19, 1980 that "there is no credible possibility * * * that it will decide to authorize nuclear units at Jamesport", and that the Long Island Lighting Company's "Board of Directors voted on November 26, 1980, to end the Jamesport nuclear project." Representing that Jamesport will not be built irrespective of the outcome of this appeal, the applicants move to terminate this proceeding as moot, as done in analogous circumstances in *Rochester Gas and Electric Corp.* (Sterling Project, Unit 1), ALAB-596, 11 NRC 867, 869 (1980).

Only the staff has answered the applicant's motion. It advises us that the Jamesport site needs no significant redress because the applicants never began construction, and endorses terminating this proceeding as they suggest.

Applicants' motion is *granted*. The appeal is *dismissed* and the Licensing Board decisions authorizing construction of the Jamesport facility (LBP-78-17, 7 NRC 826 (1978) and LBP-78-41, 8 NRC 750 (1978)) are *vacated* on grounds of mootness; the construction permit proceeding is *terminated* and the Director of Nuclear Reactor Regulation shall *revoke* the outstanding construction permits with due notice to all parties.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

Cite as 13 NRC 27 (1981) **LBP-81-1**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Administrative Judges:

Marshall E. Miller, Chairman
Robert M. Lazo
B. Paul Cotter, Jr.

In the Matter of | **Docket Nos. 50-413A**
50-414A

DUKE POWER COMPANY,
NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION,
SALUDA RIVER ELECTRIC COOPERATIVE
(Catawba Nuclear Station, Units 1 and 2)

(Antitrust)

January 13, 1981

The Licensing Board denies a request for an antitrust hearing on an application for an amendment to the construction permit for the Catawba plant authorizing the transfer of ownership interests in the plant.

RULES OF PRACTICE: INTERVENTION

In order to be admitted to a proceeding as an intervenor as a matter of right, a petitioner for intervention must allege both (1) some injury that has occurred or will probably result from the action involved, and (2) an interest arguably within the "zone of interests" protected or regulated by the statute sought to be invoked (and which and the tribunal is empowered to administer).

NRC ANTITRUST REVIEW: SCOPE

NRC antitrust jurisdiction is not plenary; the Commission is authorized to condition licenses on antitrust grounds only where necessary to insure that the activities so licensed will neither create nor maintain situations inconsistent with the antitrust laws. *Detroit Edison Company* (Enrico Fermi Atomic Power Plant, Unit No. 2), ALAB-475, 7 NRC 752, 756 (1978).

NRC ANTITRUST REVIEW: SCOPE

Contentions raising financial or safety issues are beyond the scope of NRC antitrust hearings.

NRC ANTITRUST REVIEW: REQUEST FOR HEARING

In order to invoke the NRC's antitrust jurisdiction, a request for a hearing and intervention must:

(1) describe the situation allegedly inconsistent with the antitrust laws which is the basis for intervention;
(2) describe how that situation conflicts with the policies underlying the Sherman Act, Clayton Act, or Federal Trade Commission Acts;
(3) describe how the situation allegedly inconsistent with the antitrust laws would be created or maintained by activities under the license;
(4) identify the specific relief sought; and
(5) explain why the relief sought fails to be satisfied by the license conditions, if any, which have been proposed by the Department of Justice.

RULES OF PRACTICE: INTERVENTION (DISCRETIONARY)

Petitions for intervention may be granted as a matter of discretion to petitioners lacking standing as a matter of right who may make some contribution to the proceeding. Factors to be considered include:

(1) a petitioner's showing of significant ability to contribute on substantial issues of law or fact which will not be otherwise properly raised or presented;
(2) the specificity of such ability to contribute on those substantial issues of law for fact;
(3) justification of time spent on considering the substantial issues of law or fact;
(4) provision of additional testimony, particular expertise, or expert assistance;
(5) specialized education or pertinent experience.

MEMORANDUM AND ORDER

(Denying Request for Antitrust Hearing)

Pursuant to the provisions of §105c of the Atomic Energy Act of 1954, as amended, on July 11, 1980, the Commission sought additional antitrust advice from the Attorney General of the United States in connection with the purchase of ownership interests in Catawba Nuclear Station, Unit 1.[1] This nuclear plant is owned by Duke Power Company ("Duke"), whose participation was the subject of an antitrust review conducted by the Department of Justice ("Department") in 1973. The Department originally recommended that an antitrust hearing be initiated, but the necessity for such a hearing was then obviated when Duke agreed to have certain conditions attached to its license.

The Department reviewed the instant situation resulting from the proposed sale by Duke of a 75% ownership in Catawba Unit 1 to the North Carolina Electric Membership Corporation (NCEMC, 56.25%) and to the Saluda River Electric Cooperative (Saluda River, 18.75%). It found that this sale was the result of discussions between Duke and the cooperative systems in its service area that occurred after the cessation of the original antitrust proceeding. The Department reviewed information submitted by seventy neighboring electric systems, and by letter dated October 29, 1980, the Attorney General advised NRC that no antitrust hearing was necessary with respect to the proposed transfer of ownership interests.[2]

Pursuant to §105c of the Atomic Energy Act of 1954, as amended [42 U.S.C. §2135(c)(5)], and to its own policy, the Commission upon receiving the Department's advice duly published a notice in the *Federal Register*, giving an opportunity for intervention to any person whose interest may be affected by this proceeding.[3] Any person with the requisite interest could file by December 15, 1980, a petition for leave to intervene and request a hearing on the antitrust aspects of the application (10 CFR §2.714).

A handwritten letter from Harvard G. Ayers, dated December 15, 1980, was received by the Commission on December 19. Mr. Ayers, whose address is Rt. 3, Box 662, Boone, North Carolina, stated:

> "I am a member of the Blue Ridge Electric Membership Corporation of northwestern North Carolina, and as such I am seriously concerned with the pending purchase by the North Carolina EMC's (which

[1]42 U.S.C. §2135(c).

[2]Staff's Answer to Request for an Antitrust Hearing by Harvard G. Ayers, dated January 9, 1981, p. 3.

[3]45 *Federal Register* 75393-94 (November 14, 1980).

include BREMCO) of 56.25% of Duke Power's Catawba 1 reactor. I feel the safety of this Westinghouse unit is clearly in question, vis-a-vis the McGuire containment adequacy question being pursued by the ASLB at this time. Further I question the financial advisability of the NCEMC purchase - we are in essence giving Duke Power a blank check for construction costs plus a guaranteed profit. Because of these and other reasons, I request that the NRC hold hearings on this matter preferably in Boone."

The Staff filed an answer to the request for an antitrust hearing on January 9, 1981. This answer opposed the Ayers request because (1) the letter failed to establish standing to request an antitrust hearing; (2) the letter does not satisfy any of the general criteria set forth in 10 CFR §2.714 or the other antitrust criteria governing such request; and (3) discretionary intervention is not warranted. The Staff further requested the Board to treat this matter in an expedited manner in light of unusual circumstances, involving the formal issuance of the license amendment on December 23, 1980 because the appropriate divisions of NRC Staff were unaware of the Ayers' letter and request.[4]

Although Mr. Ayer's letter is somewhat informal as a pleading, we will consider it on the merits as a petition for leave to intervene and request for a hearing on the antitrust aspects of the application, timely filed pursuant to 10 CFR §2.714. This case is very similar on the facts to *Detroit Edison Company* (Enrico Fermi Atomic Power Plant, Unit No. 2), LBP-78-13, 7 NRC 583 (1978), *aff'd.* ALAB-475, 7 NRC (1978). Under the principles there discussed, the instant intervention petition will be denied.

Intervention as a matter of right is governed under our practice by judicial standing doctrines, which require the petitioner to allege both (1) some injury that has occurred or will probably result from the action involved ("injury in fact" test), and (2) an interest arguably within the "zone of interests" protected or regulated by the statute sought to be invoked (and which the tribunal is empowered to administer).[5]

The petition asserts that Mr. Ayers is a member of the Blue Ridge Electric Membership Corporation of northwestern North Carolina ("Blue Ridge"), which is a member of one of the applicants (NCEMC) which seeks to become a co-owner and co-licensee of Catawba Unit 1. Petitioner is therefore not a ratepayer of the present licensee (Duke), nor of the potential additional licensees (NCEMC or Saluda River). His electric rates will not

[4] Staff's Answer, pp. 3-5.

[5] *Portland General Electric Company* (Pebble Springs, Nuclear Plant, Units 1 and 2), CLI-76-27, 4 NRC 610, 613-14 (1976); *Virginia Electric & Power Company* (North Anna Power Station, Units 1 & 2), ALAB-363, 4 NRC 631, 632 (1976). *See also Data Processing Service v. Camp*, 397 U.S. 150, 153 (1970); *Sierra Club v. Morton*, 405 U.S. 727 (1972).

be affected by any action of applicant utilities, but only by possible actions of Blue Ridge after rate-setting proceedings by the appropriate State regulatory body. Under a similar factual situation in *Fermi, supra,* the Appeal Board stated:

> "Petitioner seeks to invoke the Commission's antitrust jurisdiction. That jurisdiction is not plenary, however; the Commission's writ to enforce the antitrust laws does not run to the electric utility industry generally. Neither does it reach all actions by utilities that generate electricity with nuclear-powered facilities. Rather, Congress authorized this Commission to condition nuclear power plant licenses on antitrust grounds only where necessary to insure that the activities so licensed would neither create nor maintain situations inconsistent with the antitrust laws. The reason for the grant, as the Commission has explained, was 'a basic Congressional concern over access to power produced by nuclear facilities,' because the industry was nurtured by public funds and the legislature was anxious that nuclear power 'not be permitted to develop into a private monopoly via the [NRC] licensing process.' Put another way, the preservation and encouragement of competition in the electric power industry through 'fair access to nuclear power' is the principal motiviating consideration underlying Section 105c of the Atomic Energy Act." (Footnotes omitted) 7 NRC at 756-57).

The petitioner's apprehensions were not addressed to a large utility seeking to keep nuclear power away from cooperatives, which was the subject of some Congressional concerns. Indeed, her concerns were quite the opposite. The Appeal Board continued:

> "Boiled down, Mrs. Drake's arguments amount to dissatisfaction with the cooperatives' management decision to satisfy an expected need for more baseload power by acquiring part of the Fermi nuclear plant. She would prefer some other course; she fears this one will raise her electrical rates inordinatley But the Nuclear Regulatory Commission and its adjudicatory boards do not sit to supervise the general business decisions of the public utility industry nor to second-guess the judgment of those who do; that task is entrusted to others. Injuries from those causes are beyond the zone of interests that Section 105c of the Atomic Energy Act was designed to protect or regulate." (7 NRC at 757-58).

In this proceeding, Mr. Ayers purports to "question the financial advisability of the NCEMC purchase." Such a concern or contention by a ratepayer is clearly beyond the scope of the "zone of interests" that §105c of the Atomic Energy Act was designed to protect or regulate.

The Ayer's letter further states: "I feel the safety of this Westinghouse unit is clearly in question, vis-a-vis the McGuire containment adequacy question being pursued by the ASLB at this time." Intervention petitions and requests for hearing cannot properly raise antitrust issues and health and safety issues in the same proceeding.[6] In addition, the notice of opportunity for hearing to which Mr. Ayers apparently responded referred only to request for "a hearing on the antitrust aspects of the application."[7] The safety concerns described in the letter obviously are not within the ambit of antitrust issues.

In determining whether a hearing request is sufficient to invoke the Commission's antitrust jurisdiction, the request must also meet the following requirements:

(1) describe the situation allegedly inconsistent with the antitrust laws which is the basis for intervention;
(2) describe how that situation conflicts with the policies underlying the Sherman Act, Clayton Act, or Federal Trade Commission Acts;
(3) describe how the situation allegedly inconsistent with the antitrust laws would be created or maintained by activities under the license;
(4) identify the specific relief sought; and
(5) explain why the relief sought fails to be satisfied by the license conditions, if any, which have been proposed by the Department of Justice.[8]

The Ayers letter does not describe a situation inconsistent with the antitrust laws in the slightest degree, let alone the specificity necessary to trigger a hearing.

In addition to standing as a matter of right, a petition for intervention may be granted as a matter of discretion to certain petitioners who may make some contribution to the proceeding. *Portland General Electric Co.* (Pebble Springs Nuclear Plant, Units 1 and 2), CLI-76-27, 4 NRC at 610, 616-17 (1976). Although the Ayers letter is cast as a request for a hearing and does not explicitly request intervention, we will consider it as such for

[6]*Houston Lighting & Power Co.* (South Texas Project, Unit Nos. 1 & 2), ALAB-381, 5 NRC 582, 589 (1977); *Public Service Co. of Indiana (Marble Hill Nuclear Generating Station, Units 1 & 2), ALAB-316, 3 NRC 167 (1976).*

[7]45 *Federal Register* at 75394.

[8]*Kansas City Gas & Electric Co.* (Wolf Creek Generating Station, Unit No. 1), ALAB-279, 1 NRC 559 (1975) (Wolf Creek I). *See also Louisiana Power & Light Co.* (Waterford Steam Generating Station, Unit 3), CLI-73-7, 6 AEC 48 (1973) (Waterford I); *Louisiana Power & Light Co.* (Waterford Steam Generating Station, Unit 3), CLI-73-25, 6 AEC 619 (1973) (Waterford II); *Pacific Gas & Electric Co.* (Stanislaus Nuclear Project, Unit 1), LBP-77-26, 5 NRC 1017 (1977).

purposes of discussing discretionary intervention. In determining whether to grant intervention as a matter of discretion, we must consider all the facts and circumstances of a particular case, including some of the factors set forth in 10 CFR §2.714(a) and (d). *See Virginia Electric Power Co.* (North Anna Power Station, Units 1 & 2), ALAB-363, 4 NRC 631 (1976). Factors to be considered include:

(1) a petitioner's showing of significant ability to contribute on substantial issues of law or fact which will not be otherwise properly raised or presented;
(2) the specificity of such ability to contribute on those substantial issues of law or fact;
(3) justification of time spent on considering the substantial issues of law or fact;
(4) provision of additional testimony, particular expertise, or expert assistance;
(5) specialized education or pertinent experience.[9]

The one-page letter of Mr. Ayers fails to make even a rudimentary showing of any of the factors set forth in §2.714(a) and (d) or of those other factors listed above. Since it fails to set forth any anticompetitive concerns, there cannot be demonstrated any "significant ability to contribute on substantial issues of law or fact." Also, the letter fails to make any allegations that would lead to a reasonable expectation that Mr. Ayers would provide expertise, expert assistance, or additional testimony that would be helpful to any proceeding. This seems especially true since there is no Catawba antitrust proceeding now under way, the Attorney General has advised that no antitrust hearing is necessary, and the Ayers letter is the lone request for a hearing. In such circumstances a petitioner's showing on the criteria for discretionary intervention must be particularly strong.[10]

There does not appear to be any basis for determining that Mr. Ayers could or would make a "valuabe contribution...to our decision-making process" in an antitrust context. This case falls within the principles set forth in Fermi, *supra,* where it was stated:

> "There remains whether Mrs. Drake should be permitted to intervene as a matter of discretion. The test is whether her participation would be likely to contribute significantly to the proceedings. *Pebble Springs,*

[9] *Portland General Electric Co.* (Pebble Springs Nuclear Plant, Units 1 and 2), *supra; Public Service Co. of Oklahoma* (Black Fox Units 1 & 2), ALAB-397, 5 NRC 1143 (1977) *affirming in part LBP-77-17, 5 NRC 657 (1977); Tennessee Valley Authority* (Watts Bar Nuclear Plant, Units 1 & 2), ALAB-413, 5 NRC 1418 (1977).

[10] *Tennessee Valley Authority* (Watts Bar Nuclear Plant, Units 1 & 2), ALAB-413, 5 NRC 1418 (1977); *Detroit Edison Co.* (Enrico Fermi Atomic Power Plant, Unit No. 2), ALAB-475, 7 NRC 752, 758 n. 19 (1978).

supra, CLI-76-27, 4 NRC at 612, 617; *Nuclear Engineering Company* (Sheffield Waste Disposal Site), ALAB-473, 7 NRC 737 (May 3, 1978). Without a successful petition to intervene as of right, there is no automatic antitrust hearing under Section 105c when the Attorney General does not recommnd one and the Commission has not ordered one on its own. What we said in *Watts Bar* applies here: 'Certainly, before a hearing is triggered at the instance of one who has not alleged any cognizable personal interest in the operation of the facility, there should be cause to believe that some discernible public interest will be served by the hearing. If the petitioner is unequipped to offer anything of importance bearing upon [the subject matter], it is hard to see what public interest conceivably might be furthered by nonetheless commencing a hearing at his or her behest.' *Tennessee Valley Authority* (Watts Bar, Units 1 & 2), ALAB-413, 5 NRC 1418, 1422 (1977). We agree with the Licensing Board that petitioner lacks the background and training to prosecute a complex antitrust proceeding." (7 NRC 752, 758, fn. 19).

ORDER

For all the foregoing reasons and based upon a consideration of this entire record in this matter, it is, this 13th day of January, 1981.

ORDERED

1. That the request for an antitrust hearing filed by Harvard G. Ayers is denied.

2. Leave is granted to Mr. Ayers to file an amended petition for leave to intervene and request for an antitrust hearing which complies with the requirements described above, provided that an amended petition is lodged in the hands of the Licensing Board on or before January 30, 1981. Such an amended petition, if filed and served by Mr. Ayers, will be given expedited consideration by the Licensing Board.

THE ATOMIC SAFETY AND
LICENSING BOARD

Marshall E. Miller
ADMINISTRATIVE JUDGE

B. Paul Cotter, Jr.
ADMINISTRATIVE JUDGE

Robert M. Lazo
ADMINISTRATIVE JUDGE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Charles Bechhoefer, Chairman
Dr. Frank F. Hooper
Glenn O. Bright

In the Matter of **Docket No. 50-358 OL (Operating License Proceeding)**

CINCINNATI GAS & ELECTRIC, *et al.*
(William H. Zimmer Nuclear Station) **January 23, 1981**

The Licensing Board grants the applicants' motion for summary disposition of an intervenor's contention concerning the lack of a plan for training of the populace in communities through which radioactive materials will be shipped to cope with transportation accidents involving those shipments.

RULES OF PRACTICE: SUMMARY DISPOSITION

In an operating license proceeding, where significant health and safety or environmental issues are involved, a licensing board should grant a motion for summary disposition only if it is convinced from the material filed that the public health and safety or the environment will be satisfactorily protected. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 753-54 (1977); 10 CFR 2.760a.

OPERATING LICENSE: CRITERIA

The provisions of 10 CFR §73.37 require licensees to prepare a plan for the physical protection of spent fuel shipments against sabotage. There is no requirement that such a plan be reviewed prior to (and as a condition of) the grant of an operating license.

OPERATING LICENSES: HEALTH AND SAFETY ISSUES

In general, insofar as public health and safety issues are concerned, an applicant which demonstrates that it has complied with applicable regulations would be granted an operating license. Only in unusual circumstances, where possibly a demonstrable threat to the public health and safety had been shown to exist, could a licensing board consider and impose, if necessary, corrective measures additional to those prescribed or at least comprehended by the rules.

MEMORANDUM AND ORDER

(Granting Motion for Summary Disposition of Contention 5)

Contention 5, sponsored by Dr. David Fankhauser, an intervenor in this operating license proceeding, asserts that there are "no plans to provide knowledge and training of the populace in communities through which radioactive materials will be transported sufficient to allow them [*i.e.,* the communities] to be able to cope with transportation accidents." The Applicants (Cincinnati Gas & Electric Co., *et al.*) have moved for summary disposition of this contention. Upon consideration of the filings of various parties to this proceeding, as outlined below, we conclude that there is no requirement that an applicant or licensee provide knowledge or training to the populace in communities through which irradiated materials will be shipped; that there also is no obvious reason why a plan for the provision of such knowledge or training need be required prior to the grant of an operating license; and, accordingly, that the Applicants' motion should be granted.

A. Background

The Applicants' original motion for summary disposition of Contention 5 was filed on April 6, 1979. It was essentially founded on three premises: first, that questions related to the safety aspects of fuel transportation are outside the scope of matters before this Board; second, that safety in the transportation of radioactive material is provided primarily by the use of containers designed and constructed in accordance with 10 CFR Part 71[1]

[1]"Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions."

to withstand severe transportation accidents without leakage, thus minimizing the danger or threat from radiation and making the likelihood of a release of any radioactive material in a transportation accident so small as to be considered negligible; and, finally, that in any event, it would be impracticable for an applicant to provide the suggested training inasmuch as spent fuel transportation, which is carried out under applicable NRC, Department of Transportation, and state regulation, may encompass areas which are presently not ascertainable and which may be far removed from the plant site, and any releases which might occur would be highly localized and subject to adequate control through local emergency forces.

In his May 1, 1979 response, Dr. Fankhauser stated merely that, by their own admission, the Applicants had no plans for or knowledge of the shipping of waste material and, in addition, that safety in transportation is "partially dependant" upon transportation routes which as of that time had not been chosen. The Staff asked the Board to defer ruling on the Applicants' motion pending the consideration of new standards in the wake of the then-recent Three Mile Island (TMI) accident. No other party responded to the Applicants' motion (insofar as it dealt wih Contention 5).

We discussed the Applicants' motion for summary disposition of Contention 5 with the parties at the prehearing conference on May 23, 1979 (Tr. 434-41). We determined that, because the Commission was in the process of developing new regulations dealing with the transportation of radioactive material, we would defer action on the motion (Tr. 460). Thereafter, on June 15, 1979, the Commission published a proposed interim rule, to become effective on July 16, 1979. 44 Fed. Reg. 34466 (June 15, 1979). During the hearing on June 26, 1979, we invited the Applicants either to reconsider or to supplement their summary disposition motion in light of this rule (Tr. 1437-38). The Applicants did so by filing a "Renewed Motion" on July 25, 1979.

In their renewed motion, the Applicants asserted that, although the new rule covered shipments of irradiated reactor fuel, it focused on the prevention of sabotage of such shipments. The Applicants interpreted Contention 5 as not encompassing sabotage. Although imposing additional requirements for spent fuel shipments, the rule, according to the Applicants, made no reference to providing knowledge and training to the populace in communities through which irradiated fuel will be transported. Further, they noted that the coverage of the rule was limited to irradiated fuel shipments and did not extend to shipments of all types of radioactive material. The Applicants therefore claimed that their motion should be granted for the reasons they originally had advanced.

Dr. Fankhauser's response, dated August 1, 1979, took the position that the new rule required the Applicants to make plans for the routing of spent fuel, that the Applicants had not made any such plans, and that the renewed motion should be denied as a result of the lack of compliance with the new rule. Dr. Fankhauser added that his contention encompasses (although is not limited to) a concern with the threat of sabotage. No other party responded to the renewed motion.

In our Memorandum and Order Denying Motion to Delay Delivery of Fuel To The Site, LBP-79-24, 10 NRC 226, 232 (August 15, 1979), we held that, "as matter of law, there are no requirements for training of the populace in the communities through which [unirradiated] fuel will be shipped." By virtue of that ruling, the thrust of Contention 5 was for all intents and purposes confined to shipments of irradiated fuel.[2] Some time later, in our Memorandum and Order dated July 14, 1980, we announced our tentative conclusion that, under the proposed interim rule, "there * * * is no requirement or even warrant for providing knowledge or training of the general populace in communities through which spent (irradiated) fuel is to be transported." We noted that we had deferred ruling on the Applicants' summary disposition motion because of the interim nature of the proposed rule and the expectation of its further modification. We also pointed out that the Commission had adopted a "final" interim rule, 45 Fed. Reg. 37399 (June 3, 1980), together with interim guidance on the rule's implementation (NUREG-0561, Revision 1). This "final" interim rule became effective on July 3, 1980. By our Memorandum and Order of July 14, 1980, we invited all parties to submit additional comments on the Applicants' motion, taking into account the new rule (as well as several matter which we wished to have addressed).

Responses to our invitation were filed by the Applicants, the NRC Staff, Dr. Fankhauser, intervenor Zimmer Area Citizens-Zimmer Area Citizens of Kentucky (ZAC-ZACK), intervenor Miami Valley Power Project (MVPP), the City of Mentor, Kentucky, and the Commonwealth of Kentucky.[3] The Applicants reiterated their argument that consideration of the safety aspects of spent fuel shipments is beyond our jurisdiction. They also claimed that there is no requirement for a spent fuel shipment plan as a prerequisite for an operating license. The Staff joined them in this latter argument. The other parties all expressed the view that measures for the security of spent

[2]Neither the contention itself nor any of Dr. Fankhauser's papers filed or statements made concerning the contention evince any interest in radioactive material other than irradiated or unirradiated fuel.

[3]Responses of the NRC Staff and Dr. Fankhauser were dated August 1, 1980. ZAC-ZACK's comments were filed August 7, 1980. MVPP's comments were filed August 8, 1980. The Applicants and the City of Mentor responded on August 11, 1980. Kentucky responded on September 4, 1980. Dr. Fankhauser filed a response to comments of other parties on August 26, 1980.

fuel shipments (including training of the populace along routes of shipment) should be considered in this proceeding.

B. Discussion

1. At the outset, we must reject the Applicants' argument that we do not have jurisdiction to consider whether the spent fuel shipment measures proposed by Dr. Fankhauser should be applied in this proceeding.[4] In their original motion, the Applicants pointed to the circumstance that the primary safety rules governing shipment of radioactive material appear in 10 CFR Part 71 and the regulations of agencies such as the Department of Transportation, and accordingly are not embraced by the requirements governing the grant of operating licenses, which appear in 10 CFR Part 50.[5] As for the Commission's new security plan requirements, the Applicants advance much the same argument: the requirements appear in Part 73 and hence are not part of the operating license requirements of Part 50.

As a legal matter, the Applicants are correct in their claim that requirements of Parts 71 or 73 are not automatically subject to litigation in an operating license proceeding. But, as should have been apparent from the question posed by our Memorandum and Order of July 14, 1980, certain requirements of Part 73 have been incorporated into the operating license requirements of Part 50. See 10 CFR § 50.34(c). Although we may not have authority to impose on an applicant requirements (if any) of Parts 71 and 73 not incorporated into Part 50, we clearly have authority to consider which requirements are incorporated into Part 50 and whether an applicant has satisfied those requirements. *Cf. Duke Power Co.* (Perkins Nuclear Station, Units 1, 2 and 3), ALAB-591, 11 NRC 741 (1980). For that reason, we conclude that we have jurisdiction to consider whether there are any operating license requirements which comprehend the matters raised by Contention 5 and, if so, whether those requirements have been satisfied.

2. Under the Commission's Rules of Practice, a motion for summary disposition should be granted if the licensing board determines, with respect to the issue in question, that "there is no genuine issue as to any material fact and * * * the moving party is entitled to a decision as a matter of law." 10 CFR § 2.749(d). However, in an operating license proceeding such as this one, where significant health and safety or environmental issues are

[4]The Applicants correctly pointed out that the provisions of 10 CFR § 2.717(b) upon which we premised our jurisdiction to consider new fuel shipments (see LBP-79-24, 10 NRC 226, 228-230 (1979)) do not provide us authority to consider spent fuel shipments at this time. We are not relying on those provisions here.

[5]The Applicants concede that the environmental impacts of transportation of radioactive material may be considered under 10 CFR Part 51; but they claim that Contention 5 focuses on safety rather than environmental considerations and that it raises no questions governed by the requirements of Part 51. We agree.

involved, a licensing board should only grant such a motion if it is convinced from the material filed that the public health and safety or the environment (as applicable) will be satisfactorily protected. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 753-54 (1977); 10 CFR § 2.760a.

For the purposes of this discussion, we will read Contention 5 in the light most favorable to its proponent (*see Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), LBP-74-36, 7 AEC 877, 879 (1974)). Even though it is not that clear on its face, we will assume that Contention 5 encompasses the protection of spent fuel shipments from sabotage as well as from transportation accidents. See Dr. Fankhauser's filings dated August 1, 1979 and August 1, 1980. Even when read in that light, it is clear that there is no factual disagreement with respect to any material fact. Dr. Fankhauser contends that there is no plan for the shipment of spent fuel, and all parties agree. The only questions extant are legal in nature: whether there is any requirement for such a plan and, if so, whether a plan would have to include the training features sought by Dr. Fankhauser.[6] We turn now to those questions.

3. We interpret the recently amended provisions of 10 CFR Part 73 as requiring licensees to prepare a plan for the physical protection of spent fuel shipments against sabotage. 10 CFR § 73.37. There is no requirement, however, that such a plan be submitted and reviewed prior to (and as a condition of) the grant of an operating license. Indeed, the physical protection plan for spent fuel shipments, by virtue of the express terms of Part 73, need only be submitted to NRC 7 days prior to a planned spent fuel shipment. 10 CFR § 73.72 (incorporated into 10 CFR § 73.37(b)(1)).[7] Such shipments would not take place until long after issuance of an operating license—at least eight years, according to both the Applicants and Staff.[8]

4. The absence of any requirement for a plan for the shipment of spent fuel prior to the issuance of an operating license is dipositive of Contention 5. We might add, however, that, as the Applicants and Staff point out, the current lack of any facilities for the storage or reprocessing of spent fuel would make any near-term evaluation by NRC of prospective routes—as

[6]We note that, if there were a legal requirement for the type of plan envisaged by Contention 5 (as we are here interpreting it), Dr. Fankhauser might well be entitled to summary disposition of the contention in his favor.

[7]The Staff interprets § 73.72 as requiring notification 10 days in advance of a shipment, rather than 7. We are unaware of the source of this interpretation; but, for purposes of this discussion, the difference is not material.

[8]Although these assertions are not under affidavit, we take official notice that spent fuel will not be created—and hence cannot be shipped—until after issuance of an operating license and operation of the reactor.

provided by 10 CFR § 73.37(b)(7)—speculative at best. Without identification of specific routes, it would be impossible to determine where the training sought by Dr. Fankhauser should be carried out—even assuming we found that such training were warranted.[9] Moreover, with respect to requirements of 10 CFR § 73.37 other than concerning shipment routes, the extended period before which shipments could take place is a persuasive reason for the Applicants' not being required to develop a plan at this time, for any current review of that plan—involving such matters as the qualification of a shipper's employees—would also certainly have to be redone. *Cf. Potomac Electric Power Co.* (Douglas Point Nuclear Generating Station, Units 1 and 2), ALAB-277, 1 NRC 539, 544-47 (1975). For that reason, we find little warrant for a review at this time of a proposed shipment security plan.

It should be noted that Dr. Fankhauser (as well as ZAC-ZACK, MVPP, the City of Mentor, and the City of Cincinnati) assert that Contention 5 includes protection from transportation accidents generally and is not limited to protection from sabotage. ZAC-ZACK would read 10 CFR § 73.37 as including this subject, whereas Dr. Fankhauser, the City of Mentor, and the City of Cincinnati rely on generalized "public health and safety" findings required under 10 CFR § 50.40 and § 50.57 as authority to consider this matter.

In issuing its 1980 amendments to the final interim rule, the Commission made it very clear that 10 CFR § 73.37 is limited to a plan for the prevention of sabotage in spent fuel shipments. The Statement of Considerations explicitly indicates that the potentially serious consequences analyzed in the report upon which the revised 10 CFR § 73.37 is based (Sandia Laboratories Report SAND-77-1927, May, 1978) could occur *only* in the event of sabotage in or near a heavily populated area and *only* if the sabotage were to be carried out "through the skillful use of explosives." 45 Fed. Reg. 37399, 37402 (June 3, 1980).

Insofar as public health and safety issues are concerned, in normal circumstances an applicant which demonstrates that it has complied with applicable regulations would be granted an operating license. Only in unusual circumstances, where possibly a demonstrable threat to the public health and safety had been shown to exist, could a licensing board consider and impose, if necessary, corrective measures additional to those prescribed

[9]The specific routes that Dr. Fankhauser and the City of Mentor suggest be examined do not include a destination for the spent fuel shipments but merely encompass various egress routes from the site area.
Needless to say, given our rationale for dismissing Contention 5, we express no opinion as to whether, assuming there were a requirement for a plan, the training sought by Dr. Fankhauser should be included in such plan.

or at least comprehended by the rules. See *Maine Yankee Atomic Power Co.* (Maine Yankee Atomic Power Station), ALAB-161, 6 AEC 1003, 1004-1010 (1973), *remanded on other grounds,* CLI-74-2, 7 AEC 2, 3-5 (1974). As indicated above, the Commission already has determined that transportation accidents generally do not pose a significant risk to the public health and safety sufficient to warrant the consideration of protective measures beyond those prescribed in 10 CFR Parts 71 and 73. And nothing provided by Dr. Fankhauser or the other intervenors has convinced us that there is any unusual circumstance which suggests that the protection of spent fuel transportation against either sabotage or accidents need be considered in this proceeding. Thus, we decline to consider Contention 5 in the context of the generalized findings required by 10 CFR §§ 50.40 and 50.57.

5. Our holding here will necessarily put the consideration of the adequacy of a plan for the transportation of spent fuel submitted under 10 CFR § 73.37 beyond the purview of this operating license proceeding. In our Memorandum and Order of July 14, 1980, we asked the parties whether there is any other procedure by which compliance with Part 73 can be questioned by a member of the public prior to the occurrence of a shipment. Taking into account the limited, 7-day period for review of a proposed plan, the answer is obviously negative. The Applicants and Staff suggest a request for a show-cause order under 10 CFR § 2.206. Although we agree that such mechanism is the only one available, it is obvious that, at best, that route can provide only after-the-fact review. We suggest that further review, affording the opportunity for public participation, might well be warranted.[10] But the decision as to that matter is not in our hands. It has already been made by the Commission and can only be changed by the Commission.

[10]Indeed, the 7-day review period for each shipment seems inadequate, even for Staff review; it would seem that a review of at least an initial shipment would require a longer period if the review is to be completed prior to shipment. Moreover, review of such matters as the adequacy of the training of escorts or of a licensee's communications center (see 10 CFR §§ 73.37(b)(4) and (10)) could likely be effectively undertaken well in advance of the initial shipment. However, the 7-day period is currently authorized by 10 CFR § 73.72 for notification of both initial and subsequent shipments and cannot be modified by this Board, even were we to favor such modification.

That public participation might prove useful is suggested by the recent decision in *Duke Power Co.* (Oconee-McGuire), LBP-80-28, 12 NRC 459 (Oct. 31, 1980).

C. Order

For the foregoing reasons, it is, this 23rd day of January, 1981.

ORDERED

That the Applicants' motion for summary disposition of Contention 5 be *granted.*

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Cite as 13 NRC 45 (1981) **DD-81-1**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of **Docket Nos. 50-413**
50-414
(10 CFR 2.206)

DUKE POWER COMPANY
(Catawba Nuclear Station,
Units 1 and 2) **January 9, 1981**

The Director of Nuclear Reactor Regulation denies a petition filed under 10 CFR 2.206 of the Commission's regulations requesting reopening of the safety phases of the licensing proceeding for Duke Power Company, Catawba Nuclear Facility.

RULES OF PRACTICE: 2.206 PROCEEDINGS

A petition under 10 CFR 2.206 must set forth facts that establish a basis for taking the proposed action in addition to specifying the relief requested.

PRIOR AGENCY DETERMINATIONS:

Absent a significant showing calling into question prior agency determinations on issues, the Staff will not take action under 10 CFR 2.206 to institute a proceeding to further examine these issues.

NEPA: SEVERE ACCIDENT CONSIDERATIONS

RULES OF PRACTICE: REOPENING OF PROCEEDINGS

As provided in the Commission's June 1980 "Statement of Interim Policy", the Stafff will not take action to reopen past NEPA reviews in response to a petition under 10 CFR 2.206 in the absence of some "Special circumstances".

NEPA: Need for Power

Based upon an examination of reserve margin, the Catawba facilities continued to be needed.

TECHNICAL ISSUES DISCUSSED:

Stud bolts; ice condenser containments.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By petition dated January 28, 1979, Jesse L. Riley, President, Carolina Environmental Study Group (CESG), petitioned the Commission to reopen the safety phases of the licensing proceedings for Duke Power Company's Catawba Nuclear Station and McGuire Nuclear Station.

On March 7, 1979, the Director of the Office of Nuclear Reactor Regulation advised CESG that its request to reopen the McGuire proceedings had been referred to the Atomic Safety and Licensing Board since the matter of issuance of an operating license for the McGuire facility was currently pending before that Board. However, because the Catawba case was not currently pending before any Licensing or Appeal Board, the request to reopen the Catawba proceedings was to be treated as a request under 10 CFR § 2.206 of the Commission's regulations. Notice of the consideration of the Catawba request under 10 CFR § 2.206 was published in the *Federal Register.* 44 Fed Reg 14654 (March 13, 1979).

Before examining the specific issues raised by CESG in this petition, it is appropriate to review the criteria for evaluation of requests for action under 10 CFR 2.206.

A petition shall specify the action requested and set forth the facts that constitute the basis for the request.[1] The factual basis of the petition should identify new information regarding the issues under consideration. In order to have a hearing reopened on the basis of new information, as CESG seeks to do, the Appeal Board has held that the new information must identify a significant unresolved safety issue or a major change in facts material to the resolution of major environmental issues.[2] Although the Director, in

[1] 10 CFR §2.206(a).

[2] *Vermont Yankee Nuclear Power Corporation* (Vermont Yankee Nuclear Power Station), ALAB-124, 6 AEC 358 (1973); *Commonwealth Edison Company* (La Salle, Units 1 and 2), ALAB-153, 6 AEC 821 (1973). The Director of NRR had previously applied this standard in denying another petition under 10 CFR 2.206 which requested suspension of construction

considering a request for action under 10 CFR 2.206, is not bound by the Appeal Board's standard for reopening a licensing proceeding on the basis of new information, this standard is persuasive in considering requests under 10 CFR 2.260 because, as the Commission has indicated on another occasion, "[P]arties must be prevented from using 10 CFR 2.206 procedures as a vehicle for reconsideration of issues previously decided..." *Consolidated Edison Company* (Indian Point Units 1-3), CLI-75-8, 2 NRC 173, 177 (1975).

CESG's petition provides no explanation, by reference to the record or otherwise, why the matters it identifies support reopening of the record under this standard. This failure would justify denial of the petition at the outset because the petitioner has not, as required by 10 CFR 2.206, specified the facts that constitute the basis for the request.[3] However, the staff has conducted its own review of CESG's petition to reopen and has found no good cause to reopen the record at this time. Accordingly, the petition to reopen the Catawba safety hearing must be denied. The staff's analysis follows.

In its petition, CESG has asserted a number of issues as the basis for its request to reopen the safety hearings[4] subsequent to the issuance of construction permits[5]. These issues are:

(1) need for the Catawba facility and the effect a reduced level of need for that facility would have on the cost/benefit balance, especially the consideration of risk,

permits pending reconsideration of the need for power issue after the proceeding on issuance of construction permits for the facility had been closed. *Georgia Power Company* (Alvin W. Vogtle Nuclear Plant, Units 1 and 2), DD-79-4, 9 NRC 582 (April 13, 1979) (Docket Nos. 50-424 and 50-425).

[3]*See also* the Director's denial under 10 CFR 2.206 in *Duke Power Company* (Oconee Nuclear Station, Units 1, 2 and 3), DD-79-6, 9 NRC 661 (May 24, 1979) (Docket Nos. 50-269, 50-270, and 50-287).

[4]The Director does not have the power to direct a Licensing Board or Appeal Board to conduct further proceedings on the matters which CESG raises. The Director could recommend to the Commission that the hearings be reopened or the Director could issue an order based on the matters raised by CESG under which interested persons may have a right to request a hearing.

[5]Construction permits for Catawba Nuclear Station were issued on August 7, 1975. The licensee tendered is application for an operating license on March 21, 1979. The application has not yet been docketed by the NRC staff. After the application is docketed, a notice of opportunity for hearing will be published in the *Federal Register*. See 10 CFR 2.105. At that time, interested persons may seek a hearing on the proposed issuance of an operating license. It should be noted that CESG has participated as a party in the Catawba construction permit proceeding and both the McGuire construction permit and operating license proceedings.

(2) inadequate treatment at the construction permit proceeding of Class 9 accidents, stud bolts, and the ice-condenser pressure suppression containment,
(3) the degree to which the construction permit safety evaluation of Catawba was "infected" by deficiencies which CESG claims are present in the Reactor Safety Study.

I. NEED FOR POWER

CESG contends that there is a diminshed need for the Catawba facility thus affecting the cost/benefit balance struck at the construction permit stage and requiring a renewed examination of the risk involved in licensing the units. The linchpin of CESG's contention in this area is the claim of diminished need for the Catawba units. Given a continuing need for these units, the original cost/benefit balance struck at the construction permit stage remains valid. Other than its claim of diminished need, CESG offers nothing additional in its petition which could call the original cost/benefit balance into question.

The need for power issue was thoroughly explored in the Catawba construction permit proceeding. CESG was an active participant. In the construction permit proceeding, the Licensing Board,[6] as well as the Appeal Board,[7] determined that the need for power evaluations warranted the construction of the facility. An examination of these proceedings clearly exhibits that the need-for-power issue had been exhaustively treated at that time.

The question for consideration then is whether CESG has identified such new information as would clearly mandate a change in result.

The staff has analyzed the information presented in CESG's petition and has found that this information does not identify a major change in facts which would alter the need for power determination as originally analyzed in the construction permit proceedings for the Catawba facility.

CESG claims that, given Duke's current high reserves, there is virtually a certainty that the Catawba Units 1 and 2 will not be required at any time in the foreseeable future.

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[6]Initial Decision, *Duke Power Company* (Catawba Nuclear Station, Units 1 and 2) LBP-75-34, 1 NRC 626, 656-666 (1975).
[7]Partial Decision, *Duke Power Company* (Catawba Nuclear Station, Units 1 and 2) ALAB-355, 4 NRC 397, 404-414 (1976).

The staff has examined the capacities and demand requirements of the Duke system and continues to be of the view that system reserves justify the addition of the Catawba units in accordance with Duke's latest capacity expansion plan. Current scheduling calls for Catawba Unit 1 to be available by the summer of 1984 and Unit 2 to be available by the winter of 1985/86.

Table 1 presents Duke's latest forecast of peak demand and capacity plans for the winter of 1984/85 through the winter of 1989/90. Reserve margins as a percentage of peak demand are reported for two cases reflecting capacity estimates with and without the Catawba units added as scheduled. Based on these projections, reserve margins on the Duke system during winter peak demand will not be adequate to insure reliability of service unless the proposed Catawba units are added in this time frame.

The reserve margins calculated in Table 1 show that even with the scheduled addition of the Catawba units, reserve margins are expected to range between 7.3% and 26.6%. Without their addition, reserves are estimated to become unacceptably low as of the winter of 1984/85 (14.3% reserves) and become progressively worse through the forecast period. The Duke Power Company has identified reserves ranging from 17% to 25% as necessary to provide minimum acceptable reliability. The Department of Energy has indicated that reserve margins in the 15% to 25% range characterizes systems that are reasonably reliable. Based on these reserve margin standards, the staff concludes that the Catawba units are needed in order for the Duke system to maintain reliable service.

This conclusion that the Catawba units are needed is largely predicated on Duke's expectation that peak demand will grow at an average annual growth rate of about 4.5% over the forecast period. The staff views this growth rate as a reasonable estimate of future growth as the staff's own independent forecast of growth in electrical energy demand for the State of North Carolina between 1976 and 1990 is 4.4% per year. This estimate is based on a state level econometric forecasting model of electricity demand developed at the Oak Ridge National Laboratory.

TABLE 1

Peak Load Demand, Capacity, and Reserve Margins With and Without the Catawba Nuclear Station — Winter 1984/85 through Winter 1989/90

YEAR	PEAK LOAD[a]	CAPACITY WITH CATAWBA[b,c]	CAPACITY WITHOUT CATAWBA[b]	RESERVE MARGINS %	
	MWe	MWe	MWe	WITH CATAWBA as % of Peak Load Demand	WITHOUT CATAWBA as % of Peak Load Demand
Winter 1984-85	12692	15646	14501	23.3	14.3
Winter 1985-86	13157	16656	14366	26.6	9.2
Winter 1986-87	13631	16563	14273	21.5	4.7
Winter 1987-88	14092	16563	14273	17.5	1.3
Winter 1988-89	14631	16388	14098	12.0	3.6
Winter 1989-90	15179	16280	13990	7.3	7.8

[a]Peak load estimates based on Duke Power Company's long term forecast of June 1980.

[b]All capacity estimates assume MWe net firm purchases and the following retirements.
135 MWe effective by the winter of 1985-86
35 MWe effective by the winter of 1988-87
85 MWe effective by the summer of 1988
90 MWe effective by the winter of 1988-89
108 MWe effective by the winter of 1989-90

[c]Only planned additions over this timeframe are McGuire and Catawba Nuclear units:
McGuire Unit 1 - available for 1981 peak
McGuire Unit 2 - available for 1982 peak
Catawba 1 - available for 1984 peak
Catawba 2 - available for 1985 peak

Source: Duke Power Company, submittal of S. B. Nager to S. Feld (NRC), July 29, 1980

The staff notes that the Duke Power Company has initiated a load management program aimed at promoting the increased use of interruptible contracts on the part of Duke's larger customers. Customers interested in this program will be offered more attractive rates with the understanding that service may be interrupted during periods of peak demand. Because of the nature of this contracted service, Duke will not have to maintain peak reserves to support this interruptible load. Thus, that portion of the peak load represented by interruptible contracts can be added to reserves in assessing reliability. During the winter peak, the interruptible load is projected to grow from 27 MWe in the winter of 1980-81 to 389 MWe by the winter of 1989-90. The effect of this program on reserve margins would be to increase winter reserves.

However, its impact is expected to be minimal throughout this forecast period with maximum impact occurring in the winters of 1987-88 through 1989-90 when reserves would be effectively raised by about 2.5 percentage points. Correcting for this adjustment would not alter the underlying conclusions reached above.

Given a continuing need for the Catawba facility, the cost/benefit balance struck at the construction permit proceeding remains valid. While CESG implies that the cost/balance may be affected by an upward revision of risk[8], none of the factors identified by CESG as contributors to additional risk are supportable as is discussed in the next portion of this decision. Consequently, there is no basis for CESG's claim that the cost/benefit balance originally struck is now invalid.

II. ALLEGED INADEQUATE TREATMENT OF CERTAIN CONSTRUCTION PERMIT ISSUES

1. Stud Bolts

CESG raises the issue of stud bolts, which are used to mechanically secure the reactor top closure to the reactor vessel flange, as a factor leading to an increased level of risk at the Catawba facility. This issue has been litigated by CESG in proceedings before this agency with respect to both the Catawba and McGuire facilities.

[8]CESG Petition, page 2, paragraph 4.

CESG raised the stud bolt matter in both the Catawba and McGuire construction permit proceedings and, in each instance, the Licensing Board found against CESG.[9] Appeal Board reviews in each instance supported the findings of the Licensing Board.[10]

At the construction permit proceeding, CESG contended that 18% of the stud bolt stock specimens in an infinite population would fall below the acceptance value of 130 ksi, set by the ASME Code.

During the Licensing Board hearings on Catawba held in Rock Hill, South Carolina, in April 1974, this matter was extensively discussed. At that time, the discussion pointed out that the data used to establish Code allowables could be considered to have come from an essentially truncated distribution rather than a normal distribution. However, statistics, applied in the normal sense of data collected from a small sample of an infinite population, really are not applicable in this case for the reasons which follow.

Stresses are not set by statistical means. Data are collected and normalized by dividing the value of the property for each set at elevated temperatures by the room temperature value of that set. Then using linear regression techniques, a ratio trend curve is established. In using the ratio trend curves, values higher than 1.0 are reduced to 1.0. The ratio trend curve may be further modified by the responsible ASME committee after reviewing and adjusting the data as considered necessary before being accepted as the "trend curve," or "characteristic variations of property with temperature" for the material.

The ASME Code includes among its criteria for establishing allowable stresses not only fractions of yield and tensile strengths at elevated temperatures but also fractions of the specified minimum values of these properties at room temperature. In effect, allowable stresses become anchored to the requirements of the purchase specification. In turn, the yield and tensile strengths at temperature as characterized by the trend curves may be developed without specific concern for their exact positions relative to the strength scale since they are adjusted to specified minimum strengths.

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[9]*Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2) LBP-73-7, 6 AEC 92, 106-108 (1973); *Duke Power Company* (Catawba Nuclear Station, Units 1 and 2) LBP-75-34, 1 NRC 626, 642-646 (1975).

[10]*Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2 ALAB-128, 6 AEC 399, 401-404 (1973); *Duke Power Company* (Catawba Nuclear Station, Units 1 and 2) ALAB-355, 4 NRC 397, 414 (1976).

In contrast, many foreign codes require a statistical assessment of minimum strength at temperature since no room temperature specified minimum value is used to anchor the data.

Another way in which the data could be developed is the use of an unnormalized unanchored along with the variance of the data to form the basis for a statistical definition of a minimum trend curve in terms of an appropriate confidence level. Perhaps this is what CESG may be alluding to. However, the Code has chosen not to use statistical means to establish allowable stresses but instead establishes these stresses by multiplying the value selected from the ratio trend curve by the specified minimum room temperature property value and then by the appropriate stress basis factors selected from Appendix III of Section III of the Code. The specified minimum strength value in the case of the stud bolting is as established in Table 2 of the material specification, SA-540. A test coupon is removed from each end of bars selected to represent each heat of a given size for each tempering change or each 10,000 pounds whichever is less. Ten thousand pounds represents approximately 15 stud bolts of the size used on McGuire. The specified minimum property values must be met or the lot is rejected. Therefore, the conclusion that specified minimum values of yield strength at room temperature will not be below 130,000 psi is valid, and the statistical inference implied by CESG that 18% of the specimens would be below that figure is invalid.

CESG offers nothing in its petition to place into question these prior determinations in agency proceedings. Therefore, given the consideration this issue had already received and the absence of any new information on this subject in CESG's petition, and in light of the above discussion, there is no justification for CESG's claim that this issue increases the level of risk associated with the Catawba facility.

2. Ice Condenser Pressure Suppression Containment

CESG also raises the issue of ice condenser pressure suppression containment as a factor leading to an increased level of risk at the Catawba facility.

CESG has raised the ice condenser pressure suppression containment issue in both the construction permit and operating license proceedings for the McGuire facility. The containments for the McGuire and Catawba Units are virtually identical. In the McGuire construction permit proceeding, the matter was found against CESG by the Licensing Board[11]. The Appeal Board confirmed this finding[12]

The only information offered by CESG in its petition on this issue refers to certain internal documents of the Commission dated in 1972 alledgedly ...."expressing reservations about pressure suppression containments..."[13]

In 1972, Dr. S. H. Hanauer (then Technical Advisor to the NRC's Executive Director for Operations) wrote a memorandum that raised several questions on the viability of pressure suppression containment concepts. As the memo expressed reservation with respect to pressure suppression containments, an evaluation addressing each of the points raised by the memo was undertaken.

In view of the extensive testing and analyses sponsored by the owners of pressure suppression containments and the in-depth reviews by the Advisory Committee on Reactor Safeguards, the staff issued a report on this matter entitled, "A Technical Update on Pressure Suppression Type Containments in use in U.S. Light Water Reactor Nuclear Power Plants," NUREG-0474, dated July 1978. Each of the technical concerns identified by Dr. Hanauer in his 1972 memorandum is discussed in detail in the report. It was concluded that pressure suppression types of containments were conceptually acceptable. Consequently, the staff's earlier findings were confirmed.

Since the filing of CESG's petition, a new concern has developed with respect to certain types of containment systems, to which the staff has responded. The accident at TMI-2 indicated a need to consider the possibility of hydrogen generation well in excess of the amounts considered in 10 CFR 50.44 of the Commission's regulations. The staff has undertaken a study of the potential of excess hydrogen generation, the effects such concentrations of hydrogen would have on the various types of plants, and the effectiveness of various mitigation systems in protecting the plant against such situations. The results of our studies to date are presented in the SECY 80-107 series of documents:

1. SECY 80-107, Proposed Interim Hydrogen Control Requirements for Small Containments, dated February 22, 1980.

[11]*Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2) LBP-73-7, 6 AEC 92, 101-104 (1973).
[12]*Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2) ALAB-128, 6 AEC 399, 401-404 (1973).
[13]CESG petition, p. 3.

2. SECY 80-107A, Additional Information Re: Proposed Interim Hydrogen Control Requirements, dated April 22, 1980.

3. SECY 80-107B, Additional Informaion Re: Proposed Interim Control Requirements, dated June 20 1980.

When the Commission approved the licensing of the Sequoyah plant for full power operations, certain additional requirements for hydrogen control were imposed as license conditions. These include requiring an acceptable interim system for hydrogen control by January 31, 1981, and an acceptable final system by Jauary 31, 1982. Based on Commission guidance, we are proceeding with plans to implement similar requirements for all other operating ice condenser plants on a case-by-case basis.

In addition, a two-step rule-making process is currently underway which proposed that an interim rule be put in place expeditiously for the near term, and that a final rule be developed for the longer term. "Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations," 45 Fed. Reg. 65466 (October 2, 1980). With respect to ice consdenser containments, the proposed interim rule will require owners to perform certain extensive analyses of accident scenarios involving hydrogen releases and furnish the staff with a proposed approach for mitigating these hydrogen releases. Upon evaluation of these interim measures, a final rule will be developed for longer term requirements. The catawba facility will be required to meet the regulatory requirements in this area.

In summary, CESG again offers nothing in its petition to place into question prior agency determinations in this area. Given the consideration this issue has received, and current measures being undertaken to address new issues, there is no justification for CESG's claim that this issue increases the level of risk associated with the Catawba facility.

8. Class 9 Accidents

CESG also contends that the risk level for the Catawba facility is affected by the failure to consider Class 9 accidents at the construction permit proceeding. The Commission's current policy differs somewhat from the policy applied at the Catawba operating license proceeding. The term "Class 9 accident" was first used in a Commission rulemaking proposed in December 1971. "Consideration of Accidents in Implementation of the National Environmental Policy Act of 1969," 36 Fed. Reg. 22851 (1971). An Annex to Appendix D of 10 CFR Part

50 was proposed to establish the manner in which various categories of accidents should be taken into account in the environmental review for a nuclear power plant. The Commission has since withdrawn the proposed Annex and has replaced it with new interim guidance for the treatment of accident risk considerations in NEPA reviews.[14] In following the Commission's interim guidance, consideration of serious accidents is now planned as part of the Staff's review of a licensee's application for an operating license. It is useful, however, to briefly review the withdrawn Annex and other events leading to the Commission's new interim policy.

In the proposed Annex, nine classes of accidents were created based on their range of severity. Each class of accidents, except Classes 1 and 9, was required to be investigated in environmental reports and statements. Class 1 accidents were exempt because of their trivial consequences. In dealing with Class 9 accidents, the Annex stated:

> "the occurrences in Class 9 involve sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low. Defense in depth (multiple physical barriers), quality assurance for design, manufacture, and operation, continued surveillance and testing, and conservative design are all applied to provide and maintain the required high degree of assurance that potential accidents in this class are, and will remain, sufficiently remote in probability that the environmental risk is extremely low." 36 Fed. Reg. 22862 (1971).

Even though the Annex was never formally adopted by the Commission, the Annex was used as an "interim guidance." The proposed Annex was used consistently from 1971 to 1979, i.e., Class 9 accidents were not considered in environmental statements.

In September 1979, the Commission announced that it would complete the rulemaking started by the Annex and review the policy regarding accident considerations[15]. On May 16, 1980, the Commission withdrew

[14]"Nuclear Power Plant Accident Considerations under the National Environmental Policy Act of 1969;" 45 Fed. Reg. 40101 (June 13, 1980).

[15]In *Offshore Power Systems* (Floating Nuclear Power Plants), CLI-79-9 10 NRC 257 (1979). The Commission determined that consideration of a Class 9 accident in the environmental review for floating nuclear power plants was appropriate. 10 NRC at 260-61. The Commission did not use the proceeding to resolve the generic issue of consideration of Class 9 accidents at

the Annex and issued a statement of interim policy. This new interim policy was published in the *Federal Register*, on June 13, 1980. The new policy requires environmental impact statements for ongoing and future NEPA reviews to consider a broader spectrum of accidents, including severe accidents that were once designated as "Class 9." The Commission gave the following guidance for considering environmental risks, or impacts, attributable to accidents at a facility:

> "In the analysis and discussion of such risks, approximately equal attention shall be given to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases...
>
> "Events or accident sequences that lead to releases shall include but not be limited to those that can reasonably be expected to occur. In-plant accident sequences that can lead to a spectrum of releases shall be discussed and shall include sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core." 45 Fed. Reg. at 40103."

When addressing the new interim policy concerning plants for which Final Environmental Statements have been issued, the Commission stated that:

> "It is expected that these revised treatments will lead to conclusions regarding the environmental risks of accidents similar to those that would be reached by a continuation of current practices, particularly for cases involving special circumstances where Class 9 risks have been considered by the staff... Thus, this change in policy is not to be construed as any lack of confidence in conclusions regarding the environmental risks of accidents expressed in any previously issued Statements, nor, absent a showing of similar special circumstances, as a basis for opening, reopening, or expanding any previous or ongoing proceeding."[5]
>
> "However, it is also the intent of the Commission that the staff take steps to identify additional cases that might warrant early consider-

land-based reactors, but noted that "such a generic action is more properly and effectively done through rulemaking proceedings in which all interested persons may participate." Id. at 262. *See also Public Service Co. of Okla.* (Black Fox Station, Units 1 and 2) CLI-80-8, 11 NRC 433, 434-435 (1980).

[5]Commissioners Gilinsky and Bradford disagree with the inclusion of the preceding two sentences. They feel that they are absolutely inconsistent with an evenhanded reappraisal of the former, erroneous position on Class 9 accidents." 45 Fed. Reg. at 40103.

ation of either additional features or other actions to prevent or to mitigate the consequences of serious accidents. Cases for such consideration are those for which a Final Environmental Statement has already been issued at the Construction Permit stage but for which the Operating License review stage has not yet been reached. In carrying out this directive, the staff should consider relevant site features, including population density, associated with accident risk in comparison to such features at presently operating plants. Staff should also consider the likelihood that substantive changes in plant design features which may compensate further for adverse site features may be more easily incorporated in plants when construction has not yet progressed very far."

The circumstances identified by the staff as "special" fall into three categories: (1) high population density around the proposed site; (2) a novel reactor design (a type of power reactor other than a light water reactor); or (3) a combination of a unique design and a unique siting mode. Another exeptional case noted by the Commission that might warrant consideration is the proximity of a plant to a "man-made or natural hazard."[16]

As discussed in Section 1 of the Catawba Safety Evaluation Report,[17] the nuclear steam supply system for each unit will consist of a four-loop Westinghouse pressurized water reactors. The principal features of the Catawba plants are similar to those previously approved for other nuclear power plants now under construction or in operation, especially the McGuire Nuclear Station Units 1 and 2 (Docket Nos. 50-369 and 50-370) and the Donald C. Cook Nuclear Plant Units 1 and 2 (Docket Nos. 50-315 and 50-316). Therefore, Catawba is not a novel reactor design.

The staff has developed population density guidelines, which are given in Regulatory Guide 4.7, for determining when the population surrounding a proposed new site is sufficiently high to require that special attention be given to the consideration of alternative sites with lower population densities.

The following table shows the cumulative population and population density out to a radius of 30 miles around the Catawba site for the years, 1970, 1985, and 2019. The 1970 population was based on census

[16]*Black Fox,* CLI-80-8, *supra.*
[17]Safety Evaluation Report of the Catawba Nuclear Station Units 1 and 2, dated October 12, 1973, Supplement No. 1 dated January 21, 1974.

data, while the 1985 and 2019 projections were developed by the applicant.

POPULATION DISTRIBUTION - CATAWBA SITE

RADIUS	CUMULATIVE POPULATION			POPULATION DENSITY, PERSONS/MI²		
Miles	**1970**	**1985**	**2019**	**1970**	**1985**	**2019**
0 - 5	5626	13196	19007	72	168	242
0 - 10	65227	85433	130262	208	272	415
0 - 20	442339	601884	961164	352	479	765
0 - 30	705691	943730	1500059	250	334	531

As shown in the table above, the cumulative population density surrounding the Catawba site is estimated to be less than the 500 persons per square mile density guideline of R.G. 4.7 which applies at the startup date (1983) out to a distance of 30 miles. The projected growth rate for the area surrounding the site indicates that the population density will stay well within the 1,000 persons per square mile guideline over the lifetime of the plant.

While the population density within 20 miles for the year 1985 is projected to be close to the value of 500 persons per square mile, it approaches this value only at one location, and at a significant distance from the site. A large fraction of the population at this distance is due to the City of Charlotte, N.C., and its environs, located about 17 miles from the Catawba site. Staff studies of accident risk leads the staff to conclude that the risk is higher for persons relatively close to the site, and generally decreases with distance. In particular, the staff has found that the most severe consequences of very large accidents, namely, acute fatalities, would be generally limited to distances of about 5 miles or less. Also, the Commission's recently revised regulations in regard to Emergency Planning, 10 CFR Part 50, Appendix E, require emergency planning zones for the plume exposure pathway out to about 10 miles from the reactor site. Beyond these distances the consequences are expected to diminish significantly.

Based upon the foregoing findings and considerations, the staff concludes that the population data for the Catawba site do not reflect a sufficiently unique circumstance to warrant considerations of Class 9 accident consequences at this time.

With regard to the third category of special circumstances, the staff has identified a potential special circumstance involving the dewatering system used at Catawba. In the event of a serious accident the dewatering system, as presently designed, could potentially result in undesireable releases to the surrounding environment. The staff will examine this issue during its operating license review to determine its seriousness and potential impact.

In the staff's view, this issue can be resolved prior to issuance of an operating license, and therefore would not form a bases upon which to reopen the construction permit proceedings. This finding is based on the fact that the range of solutions, including additional structural design considerations or storage capabilities, are of such nature that they do not preclude continued construction, and could result in modifications which can be made in later stages of construction.

In addition to the staff's review, measures have been taken or are under consideration by the Commission and the staff to prepare to meet the possible consequences of a serious accident at a reactor site including:

- A rule was issued, 45 Fed. Reg. 162 (Aug. 19, 1980), which significantly revises requirements in 10 CFR Parts 50 and 70 for emergency planning at nuclear power plants.
- Recommendations of the Siting Policy Task Force (*see* NUREG-0625, Aug. 1979) with respect to possible changes in the reactor siting policy and criteria set forth in 10 CFR Part 100. One goal of the recommendations is to consider in siting the risk associated with accidents beyond the design basis (*i.e.,* Class 9) by establishing population density and distribution criteria.
- Proposed "Action Plans" (*see* NUREG-0660, Vol. 1, May 1980) for implementing recommendations made by bodies that have investigated the Three Mile Island accident. Among other matters these plans incorporate recommendations for rulemaking related to degraded core cooling and core melt accidents. In addition, certain of these recommendations have been adopted as requirements as described in NUREG-0737 "Clarification of TMI Action Plan Requirements" (October 1980).
- Imposition of additional requirements on operating reactors, e.g., the short-term "lessons-learned" recommendations. *See* "TMI-2 Lessons Learned Task Force Status Report and Short-term Recommendations," NUREG-0578 (1979), and Orders published in 45 Fed. Reg. 2427-2455 (Jan. 11, 1980).

Given the staff's forthcoming review of serious accidents, there is no basis to support CESG's claim that failure to consider Class 9 accidents at the construction permit stage elevates the level of risk associated with the Catawba facility. This conclusion is further assured by the additional measures noted by the Commission in its new statement of interim policy on accident considerations.

III. THE REACTOR SAFETY STUDY

The final issue raised by CESG deals with the Reactor Safety Study (RSS).[18] CESG asserts that the McGuire and Catawba "safety evaluation were infected with the same tendencies that the Commission has found in the Rasmussen Report." However, aside from averring that the same personnel were involved, CESG provides no specifics in support of such claim. Indeed, CESG acknowledges that the RSS was not relied upon in either the Catawba or McGuire safety evaluation.[19]

This question was extensively dealt with by the Atomic Safety and Licensing Board in the McGuire operating license proceeding.[20] The Licensing Board there correctly concluded that ". . .there is no nexus between the Rasmussen Report and CESG's claim of special circumstances warranting reopening the record in this proceeding."[21] The same conclusion applies to the Catawba facility.

The Licensing Board in McGuire also reviewed the results of an NRC staff survey of the uses which the Staff had made of the RSS. The Board concluded that the Commission's withdrawal of its approval of the Executive Summary of the RSS was not a basis for reopening the record of that proceeding.[22] That conclusion applies with like force to the Catawba facility.

Accordingly, the review of the Rasmussen Report cannot serve as a basis for reopening this proceeding.

Conclusion

For the reasons set forth above, the petition of the Carolina Environmental Study Group to reopen the safety phases of licensing proceedings for Duke Power's Catawba Nuclear Station is hereby denied.

A copy of this Decision will be placed in the Commission's Public Document Room, 1717 H Street, NW, Washington, DC 20555, and at the local public document room for the Catawba Nuclear Station at the York County Library, 325 South Oak Avenue, Rock Hill, South Carolina 29730. A copy of this Decision will also be filed with the Secretary of the

[18]Reactor Safety Study, WASH-1400, October 1975.
[19]CESG Petition, page 3, paragraph 5.
[20]Memorandum and Order Ruling on Motions to Reopen Record, *Duke Power Co.* (McGuire Nuclear Station, Units 1 and 2), unpublished (April 10, 1979). A copy of this Order is attached.
[21]Id., p. 3.
[22]Id., pp. 2-9.

Commission for review by the Commission in accordance with 10 CFR 2.206(c) of the Commission's regulations. As provided in 10 CFR 2.206(c), this Decision will constitute the final action of the Commission twenty (20) days after the date of issuance, unless the Commission on its own motion institutes a review of this Decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland
this 9th day of Jan., 1980

ADDENDUM TO DIRECTOR'S DECISION UNDER 10 CFR 2.206

On January 9, 1981, the Director of Nuclear Reactor Regulation issued a Director's Decision pursuant to 10 CFR 2.206 which denied a petition of the Carolina Environmental Study Group submitted on January 28, 1979. That petition requested that the Commission re-examine certain issues related to the Catawba Nuclear Station of the Duke Power Company.

A portion of the Staff's review was unintentionally omitted from the decision. Specifically, with respect to the Staff's examination of special circumstances which might warrant consideration of serious accidents[1], the Staff considered "man-made or natural hazards".

The Staff analyzed the site characteristics and other nearby features to assess the potential for impairment of safety-related portions of station facilities due to natural or man-made hazards occuring nearby. The Safety Evaluation Report[2] issued at the Construction Permit stage, states the staff conclusion that there are no industrial, transportation, or military facilities in the area of the site which have potential to adversely affect plant safety systems. The staff review specifically ensured that station design is adequate to accommodate other natural characteristics of the site environs. The staff review has not identified any unusual circumstances with respect to natural or man-made hazards that would warrant reopening or expanding proceedings on Catawba.

A copy of this Addendum will be placed in the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. 20555, and at the local public document room for the Catawba Nuclear Station at the York County Library, 325 South Oak Avenue, Rock Hill, South Carolina 29730. A copy of this Addendum will also be filed with the Secretary of the Commission for review by the Commission in accordance with 10 CFR

[1]Director's Decision, p. 20.
[2]Safety Evaluation of the Catawba Nuclear Station Units 1 & 2, October 12, 1973, pp. 2-8, 2-9.

2.206(c) of the Commission's regulations. As provided in 10 CFR 2.206(c), this Addendum will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission on its own motion institutes a review within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland
this 6th day of February, 1981

ISSUANCE OF ADDEDUM TO DIRECTOR'S DECISION UNDER 10 CFR 2.206

On March 13, 1979, notice was published in the *Federal Register* (44 FR 14654) that, by petition dated January 28, 1979, Mr. Jesse L. Riley, President, Carolina Environmental Study Group (CESG), had requested that the Commission reopen safety phases of the licensing proceedings for Duke Power Company's Catawba Nuclear Station and McGuire Nuclear Station. CESG has asserted several issues as the basis for its request. On March 7, 1979, the Director of the Office of Nuclear Reactor Regulation advised CESG that its request to reopen the McGuire proceedings had been referred to the Atomic Safety and Licensing Board since the matter of issuance of operating licenses for the McGuire facility was currently pending before that Board. The Catawba case was not currently pending before any Licensing or Appeal Board. Consequently, the CESG's request with respect to Catawba was treated as a petition under 10 CFR 2.206 of the Commission's regulations to reopen the safety hearing for the Catawba Nuclear Station.

The Director of Nuclear Reactor Regulation denied the petition in the Director's Decision 81-1 issued on Jauary 9, 1981. However, a portion of the Staff's analyses of the CESG petition was unintentionally omitted from that Decision. Consequently, the Staff has prepared an Addendum to supplement the Director's Decision.

Copies of the addendum are available for inspection in the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Local Public Document Room for Catawba, located at the York County Library, 325 South Oak Avenue, Rock Hill, South Carolina 29730. A copy of the addendum will also be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

As provided in 10 CFR 2.206(c), the Addendum will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission, on its own motion, institutes a review within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland
this 6th day of February 1981

Cite as 13 NRC 67 (1981) CLI-81-2

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

John F. Ahearne
Victor Gilinsky
Joseph M. Hendrie
Peter A. Bradford

In the Matters of

GENERAL ELECTRIC COMPANY (Exports to Taiwan)	**Application No. XR-135** **Docket No. 11001075** **Application No. XSNM-01662** **Docket No. 11001076**
WESTINGHOUSE ELECTRIC COMPANY (Exports to Taiwan)	**Application No. XR-136** **Docket No. 11002058** **Application No. XSNM-01719** **Docket No. 11002175**
COMBUSTION ENGINEERING, INC. (Exports to Taiwan)	**Application No. XR-137** **Docket No. 11002252** **Application No. XSNM-01753** **Docket No. 11002253** **Application No. XSNM-01754** **Docket No. 11002254**

February 13, 1981

The Commission denies a petition to intervene and request for a hearing on license applications for the export to Taiwan of two nuclear reactors and associated nuclear fuel, determines that the applications meet all applicable export licensing criteria, and directs that the licenses be issued with the condition that they become effective only upon award to the licensee of a contract by the Taiwan Power Company.

EXPORT LICENSES: SCOPE OF REVIEW

In making export licensing decisions, the Commission will not consider: seismic risks posed by the reactor site; volcanic risks posed by the site; risks posed by a high population density around the site; and dangers to the health and safety of individuals residing in the recipient nation.

EXPORT LICENSES: SCOPE OF REVIEW

In making reactor export licensing decisions, the Commission will consider the likely environmental impact on the global commons of the proposed reactor exports.

EXPORT LICENSES: SCOPE OF REVIEW (NUCLEAR NON-PROLIFERATION ACT)

Where licenses authorizing exports of power reactors and low enriched uranium to a specific country have previously been issued by the Commission subsequent to the enactment of the Nuclear Non-Proliferation Act, the Commission may issue licenses for similar exports to the same country upon determining that there are no material changed circumstances associated with the applications under consideration from those existing at the time of issuance of the earlier license. Atomic Energy Act, Section 126a.(2)(B); 10 CFR 110.44(a) (2). In determining whether material changed circumstances have occurred since issuance of a previous license, the Commission examines whether any events have adversely affected the adequacy of the assurances given by the recipient nation on the matters covered by sections 127 and 128 of the Atomic Energy Act, and whether there have been any significant changes in the recipient nation's non-proliferation policies.

MEMORANDUM AND ORDER

I. Background

The Taiwan Power Company seeks to purchase two nuclear power reactors which will be sited in Yenliao, about 50 kilometers east of Taipei. In soliciting bids, the utility stated that the contract would not be awarded to a supplier unless that company had already received authorization from its government to export the reactors, the initial fuel core, and one full reload. The General Electric Company, the Westinghouse Electric Corporation, and Combustion Engineering, Inc., have each filed applications with the Commission seeking authorization to export to Taiwan two nuclear reactors, the initial fuel cores, and one full reload.[1]

The Center for Development Policy has filed petitions seeking leave to intervene and requesting a public hearing on the applications. Petitioner requests that the NRC hold public hearings on the following issues:

1. The nature and magnitude of the seismic risks and dangers posed by the reactors' site and the effects on the global commons;

2. The nature and magnitude of the volcanic risk and dangers posed by the reactors' site and the effects on the global commons;

3. The nature and magnitude of the risks and dangers posed by the high population density around the reactors' site;

4. The risk to the common defense and security of the United States due to the lack of legally binding non-proliferation agreements;

5. Dangers to the health and safety of Taiwanese citizens;

6. The likely environmental impact on the global commons of the proposed reactors and disposition of spent fuel; and

7. Generic safety questions posed by all nuclear power plants.

Petitioner requests the Commission to defer action on the pending export applications until the Commission: (a) has made publicly available all pertinent data within its possession that relates to the issues raised; and (b) has held an adjudicatory hearing on these issues, commencing no earlier than 90 days after the requested information had been made available to the public. Petitioner also asks that the Commission's staff assist it to analyze and evaluate the information provided. In addition, petitioner urges the Commission to request the United States Geological Survey to prepare a volcanic and seismic assessment of risks the proposed reactors would pose to the environment of the global commons, and also to develop guidelines which would govern the siting of reactors near volcanoes. Finally, petitioner requests the Commission to prepare a revised environmental impact statement assessing the impact U.S. nuclear exports have on the global environment. The statement would update the "Final Environmental Statement on U.S. Nuclear Power Export Activities" published by the

[1]General Electric filed its applications for authorization to export two boiling water reactors and the fuel on March 7, 1980. Westinghouse filed its application covering two pressurized water reactors on April 23, 1980. It submitted its fuel application on August 14, 1980. Combustion Engineering filed its applications for two pressurized water reactors and the fuel on October 9, 1980. The Commission published notice of the receipt of the reactor applications in the Federal Registers of April 15, 1980 (45 Fed. Reg. 25560), June 24, 1980 (45 Fed. Reg. 42431), and November 18, 1980 (45 Fed. Reg. 76306).

Energy Research and Development Administration in April, 1976 (ERDA-1542).

The NRC staff, the State Department (speaking on behalf of the Executive Branch) and the applicants filed responses with the Commission recommending that the petitions be denied.

The NRC staff[2] and the Executive Branch[3] have also submitted documents to the Commission in which they conclude that the license applications meet all the applicable export licensing criteria and recommend issuance of the licenses. However, both recommend that the licenses be made effective only upon the award of a contract by the Taiwan Power Company to one of the United States applicants for constructing the reactor. The Executive Branch submission included a "Concise Environmental Review of the Fourth Nuclear Power Plant, Units 7 and 8 on Taiwan"; the staff submission included an "Office of Nuclear Reactor Regulation Staff Evaluation of the Potential Radiological Impact on the Global Commons of the Export of Taiwan Nuclear Units 7 and 8."

II. The Hearing Request

(a) Hearings as matter of right

The Center for Development Policy (CDP) is a project of the International Center, a District of Columbia nonprofit corporation. The functions of CDP are to monitor the flow of resources to developing nations, conduct research and analysis of development policies and their implementation, and disseminate the results to the public and public officials.

The interests petitioner assert and the issues they raise are virtually identical to those contained in CDP's earlier petition challenging the export of two nuclear reactors to South Korea. The Commission denied that intervention request, determining that petitioner had failed to establish that it was entitled to a hearing as matter of right under Section 189(a) of the Atomic Energy Act of 1954, as amended. *In the Matter of Westinghouse Electric Corp.* (Export to South Korea), CLI-80-30, 12 NRC 253 (1980). For the reasons set forth in that opinion, the Commission has determined that petitioner is not entitled to a hearing as a matter of right in the present case.

(b) Hearing as a matter of discretion

Even though petitioner is not entitled to a hearing as a matter of right, the Commission can order a public hearing if it determines that a hearing would be in the public interest and would assist the Commission in making

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[2]Memorandum to the Commission from William J. Dircks, Executive Director for Operations, NRC, dated January 14, 1981, SECY-81-34.

[3]Memorandum for James R. Shea, Director, Office of International Programs, NRC, from Louis V. Nosenzo, Deputy Assistant Secretary of State, dated December 12, 1980.

the statutory determinations required by the Atomic Energy Act. 10 CFR 110.84(a). The Commission is unable to make such a determination here.

Four of the issues raised by petitioners pertain to matters which the Commission has stated it will not consider in making its export licensing decisions. These are: (1) the seismic risks posed by the reactors' site; (2) the volcanic risk posed by the reactors' site; (3) the risks posed by the high population density around the reactors' site; and (4) dangers to the health and safety of Taiwanese citizens. The export licensing process is also an inappropriate forum to consider a fifth issue, the generic safety questions raised by nuclear power plants. *In the Matter of Westinghouse Electric Company* (Exports to the Philippines), CLI-80-14, 11 NRC 631 (1980); *Westinghouse Electric Company* (Exports to the Philippines, CLI-80-15, 11 NRC 672 (1980); *Westinghouse Electric Company,* CLI-80-30, 12 NRC 253 (1980).

The remaining issues raised by petitioner — whether U.S. Agreements with Taiwan on non-proliferation matters are binding in light of the United States' China policy and the likely environmental impact on the global commons of the proposed reactor exports and disposition of spent fuel — fall within the Commission's jurisdiction and may be considered by the Commission in its export licensing determinations. The issue petitioner raises regarding the status of U.S. nuclear agreements was explicitly addressed by the Congress in the Taiwan Relations Act of 1979, 22 U.S.C. Sec. 3301, *et seq.* In that Act Congress made it clear that agreements entered into between the United States and the governing authorities on Taiwan possess a legally binding character and that severence of United States diplomatic relations with Taiwan does not provide a basis for denying nuclear export applications. See 42 U.S.C. 3303. Therefore, this issue need not be considered further by the Commission.

With respect to the impact of the proposed reactor exports upon the global commons, both the Executive Branch and the NRC staff have addressed the issue in their submissions to the Commission. As noted previously, the NRC staff prepared an analysis of the radiological impacts of the proposed exports upon the global commons and the Executive Branch in its Concise Environmental Review has also addressed impacts of the reactor exports upon the global commons. There is no indication in its pleadings that petitioner possesses expertise on assessing impacts on the global commons or that they have information not presently available to the Commission on the matter. Instead petitioner has requested that the Commission provide it with pertinent information and then assist it to analyze and evaluate the data. We have no reason to believe that such an effort would result in the development of significant new insights or a more comprehensive analysis of the issues than that already submitted by the

NRC staff and the Executive Branch. We also find no reason to seek additional analyses from the U.S. Geological Survey or to request the Executive Branch to update ERDA-1542 before acting on the merits of the pending applications.

In the absence of evidence that a hearing would generate significant new analyses, a public hearing would be inconsistent with one of the major purposes of the Nuclear Non-Proliferation Act of 1978 — that United States agencies enhance the nation's reputation as a reliable supplier of nuclear materials to nations which adhere to our non-proliferation standards by acting upon export license applications in a timely fashion.[4] A hearing would delay the Commission's decision on the applications for several months. Therefore, we conclude that a public hearing would not be in the public interest or assist the Commission in making its statutory determinations.

III. The Taiwan Export License Applications

The Commission has issued export licenses authorizing exports of power reactors and low enriched uranium to Taiwan since enactment of the Nuclear Non-Proliferation Act, specifically XR-113 (two power reactors) and XSNM-01341 (low enriched uranium). These licenses were issued on June 8, 1979. Under these circumstances, Section 126a.(2)(b) of the Atomic Energy Act and 10 CFR 110.44(a)(2) of the Commission's implementing regulations authorize the Commission to issue licenses to Taiwan by determining that there are no material changed circumstances associated with the applications under consideration here from those existing at the time of issuance of the earlier licenses. In determining whether material changed circumstances have occurred since issuance of a previous license, the Commission examines whether any events have adversely affected the adequacy of the assurances given by the recipient nation on the matters covered by sections 127 and 128 of the Atomic Energy Act, and whether there have been any significant changes in the recipient nation's non-proliferation policies.

The Commission has reviewed the submissions of the Executive Branch, the NRC staff and the petitioner and has concluded that there have been no material changed circumstances regarding the adequacy of the assurances given to the United States by Taiwan and that there have been no changes in Taiwan's non-proliferation policies that would cause it to alter its previous determinations. We therefore direct the Assistant Director for Export-Import and International Safeguards, Office of International

[4]See Section 2(b) of the Nuclear Non-Proliferation Act, 22 U.S.C. 3201(b).

Programs, to issue license XR-135 and XSNM-01662 to the General Electric Company, licenses XR-136 and XSNM-01719 to the Westinghouse Electric Corporation, and XR-137, XSNM-01753, and XSNM-01754 to Combustion Engineering, Inc. Each of the power reactor licenses shall expire on January 31, 2002. This will permit shipment of warranty replacements and other items needed to assure safe and efficient operation of the reactors. Each of the fuel applications shall expire on January 31, 1995. Each of the seven licenses shall also be conditioned to provide that they shall become effective, only upon award to the licensee of the contract for Taiwan Power Company Units 7 and 8.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 13th day of February, 1981.

SEPARATE OPINION OF COMMISSIONER BRADFORD

I concur in the denial of the intervention petition, but dissent from the issuance of the licenses. For reasons set forth in the recent Philippine case[1] and reiterated in the Korean case,[2] I believe a more extensive review is required. I know of no reason why these applications would not pass such a review, but none has been done.

I should also note that a review of the type I have advocated would not necessarily preclude the issuance of a license when, as here, the contract has not been awarded. However, some additional information on the type(s) of reactor under consideration would be necessary.

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[1] *Westinghouse Electric Corp.*, CLI-80-14, 11 NRC 631, 666 (1980).
[2] *Westinghouse Electric Corp.*, CLI-80-30, 12 NRC 253, 263 (1980).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-466**

HOUSTON LIGHTING & POWER COMPANY
(Allens Creek Nuclear Generating Station, Unit No. 1) **February 2, 1981**

The Appeal Board affirms the Licensing Board's grant of applicant's motion for summary disposition of a contention advocating the consideration of a biomass farm as an alternative to the construction of the Allens Creek facility.

RULES OF PRACTICE: MOTION FOR SUMMARY DISPOSITION

The Rules of Practice authorize any party to a proceeding to move, with or without supporting affidavits, for a decision in that party's favor as to all or any part of the matters involved in the proceeding. Such motion must be accompanied by "a separate, short and concise statement of the material facts as to which the moving party contends that there is no genuine issue to be heard." 10 CFR 2.749(a).

RULES OF PRACTICE: MOTION FOR SUMMARY DISPOSITION (RESPONSE)

The Rules of Practice require a party responding in opposition to a motion for summary disposition to include in its response "a separate, short and concise statement of the material facts as to which it is contended that there exists a genuine issue to be heard." 10 CFR 2.749(a). Where a motion for summary disposition is properly supported, the response may not rest upon mere allegations or denials, but rather "must set forth specific facts showing that there is a genuine issue of fact." *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451, 453 (1980); 10 CFR 2.749(b).

ENVIRONMENTAL IMPACT STATEMENT: CONSIDERATION OF ALTERNATIVES

Environmental impact statements need not discuss the environmental effects of alternatives which are deemed only remote and speculative possibilities. *Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 NRC 14, 38 (1979).

APPEARANCES

Mr. F. H. Potthoff, III, Houston, Texas, appellant *pro se.*

Messrs. Jack R. Newman, Robert H. Culp and **David B. Raskin,** Washington, D. C., and **Messrs. J. Gregory Copeland, C. Thomas Biddle** and **Darrell Hancock,** Houston, Texas, for the applicant, Houston Lighting & Power Company.

Mr. Richard L. Black for the Nuclear Regulatory Commission staff.

DECISION

Last April, by a divided vote, we reversed an order of the Licensing Board which had denied the petition of F. H. Potthoff, III, for leave to intervene in this construction permit proceeding involving the proposed Allens Creek facility. ALAB-590, 11 NRC 542. The reversal was founded on our conclusion, contrary to that of the Board below, that the petition contained one litigable contention. That contention (identified as No. VI) asserted in substance that the construction and operation of a marine biomass farm is both a viable and an environmentally superior alternative to Allens Creek (and therefore should have been favorably considered in the Final Environmental Statement prepared by the NRC staff).

Although "determin[ing] that Mr. Potthoff must be admitted to the proceeding on the strength of his contention VI", we took pains to note that this did not "carry with it any implication that we view the contention to be meritorious". 11 NRC at 549. Further, we stressed that it did not perforce follow from our decision that the contention would have to be taken up at an evidentiary hearing. In this connection, we called attention to the availability of the summary disposition procedures set forth in the Commission's Rules of Practice. 10 CFR 2.749. Those procedures "provide in reality as well as in theory, an efficacious means of avoiding unnecessary and possibly time-consuming hearings on demonstrably insubstantial issues". 11 NRC at 550.

In compliance with ALAB-590, the Licensing Board issued an order on April 30, 1980 in which it granted Mr. Potthoff's intervention petition and accepted his contention VI. Thereafter, on August 4, the applicant moved for summary disposition of the contention in its favor. Following receipt of Mr. Potthoff's response in opposition to it, the Licensing Board granted the motion in an unpublished order entered on November 13.[1]

Mr. Potthoff appeals from this result.[2] The appeal is opposed by both the applicant and the NRC staff. We affirm.

I

Section 2.749(a) of the Rules of Practice authorizes "[a]ny party to a proceeding" to "move, with or without supporting affidavits, for a decision by the [Licensing Board] in that party's favor as to all or any part of the matters involved in the proceeding". The motion must be accompanied by "a separate, short and concise statement of the material facts as to which the moving party contends that there is no genuine issue to be heard".

[1]The NRC staff had filed a response in support of the motion on October 2, to which was appended the affidavit of an environmental scientist associated with the Oak Ridge National Laboratory. The Licensing Board expressly declined to consider that response for the assigned reason that, at the time it was submitted, 10 CFR 2.749 (a) "provided only for the submission of an answer opposing a motion for summary disposition". Order p. 21, fn. 10.
Effective October 17, 1980 — several weeks before the Licensing Board acted on the motion — the Commission revised Section 2.749 (a) to authorize in terms responses "supporting or opposing" motions for summary disposition". 45 *Fed. Reg.* 68919 (October 17, 1980). In doing so, the Commission noted that "it has been a long-standing practice of the [NRC] staff to file an answer to the motions of other parties for summary disposition — in support or in opposition, as appropriate". Accordingly, the change in language was characterized as simply a "clarification" of the summary disposition rule. *Ibid.*
In these circumstances, it is doubtful at best that the Board was justified in declining to consider the staff response here. As will later be seen, however, it is not necessary to decide the point in the disposition of the appeal before us.

[2]Had other contentions of Mr. Potthoff been admitted to the proceeding, the proscription against appeals from interlocutory orders (10 CFR 2.730 (f)) would have come into play. In other words, he would have had to await the rendition of the Licensing Board's initial decision before complaining to us of the summary disposition of contention VI. Because, however, that contention provided the sole footing for his being allowed intervention the consequence of the summary disposition of it was Mr. Potthoff's dismissal from the proceeding. See November 13 order, at p. 25 This being so, there is the requisite degree of finality to permit an appeal at this juncture. See *Toledo Edison Co.* (Davis-Besse Nuclear Power Station), ALAB-300, 2 NRC 752, 758 (1975). None of the parties to the appeal contends otherwise.

The Section is equally explicit respecting the obligation of a party which opposes the motion. Its response must include, *inter alia*, "a separate, short and concise statement of the material facts as to which it is contended that there exists a genuine issue to be heard". (In this connection, the material facts asserted by the movant "will be deemed to be admitted" unless the opposing party controverts them.) Further, by virtue of Section 2.749 (b), "if the motion is properly supported, the opposition may not rest upon 'mere allegations or denials'; rather, the answer 'must set forth specific facts showing that there is a genuine issue of fact' ". *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451, 453 (1980).

A. In its required statement of material facts as to which no genuine issue existed, the applicant asserted essentially the following: The Allens Creek nuclear facility is scheduled to commence operation in 1988 and to provide 1,200 MW of base load electrical power. Without allowance being made for preprocessing activities,[3] it would take a marine biomass farm covering an area in excess of 900 square miles (576,000 acres) to produce an equivalent amount of energy. No farm of that size is now in existence or known to be under development. Moreover, even should future research and development of marine biomass production and conversion technologies demonstrate the commercial feasibility of establishing a farm of the size needed to replace the energy output of the Allens Creek facility, such a farm could not be available by 1988.

In addition, according to the applicant's statement, the creation of a marine biomass farm of the requisite dimensions would entail obtaining exclusive control over 15 to 20 percent of the usable acreage of the Gulf of Mexico from the mouth of the Mississippi River to Mexico. There are no present legal means by which the applicant could obtain title to or exclusive use of that territory. Finally, the environmental impacts associated with so massive a farm would be "numerous and potentially extremely significant" and would exceed those arising from construction and operation of Allens Creek.

These several representations were addressed in the affidavit of Dr. Herbert H. Woodson,[4] which also accompanied the summary disposition motion. As noted in ALAB-590, in contention VI Mr. Potthoff relied upon the Federal Energy Administration's "Project Independence Report",

[3]See fn. 5, *infra*.

[4]Dr. Woodson, formerly a Professor of Electrical Engineering first at the Massachusetts Institute of Technology and then at the University of Texas, has been since 1974 the Director of the Center for Energy Studies at the latter institution. That center is described in his curriculum vitae as "an interdisciplinary research organization that carries on a diverse array of energy-related projects".

issued in November 1974, for the proposition that a marine biomass farm is a viable alternative to Allens Creek. 11 NRC at 544, 547. Dr. Woodson states that he examined that report and found that it disclosed the need for further research to establish the commercial viability of biomass conversion. His affidavit states further that the research data which had become available since 1974 persuaded him that

> marine biomass is not now a commercially viable energy source for the production of electricity. Substantial research and development must be undertaken, and technological problems overcome, before this energy source can be considered a viable alternative. That it can become viable as an alternate energy source is not a certainty; and, in fact, its practical viability is highly doubtful.

Affidavit, p. 1. In this regard, the affidavit quotes portions of two reports which had been issued by the Electric Power Research Institute in 1979 on the subject of marine biomass resources and conversion. *Id.* at pp. 2-3.

Dr. Woodson proceeds to detail the basis for the applicant's assertion respecting the required size of any marine biomass farm which might serve to produce the energy equivalent of Allens Creek.[5] He then takes note of a biological test farm which was established off the California coast in September 1978 and covers an area of almost 10,000 square feet — approximately one-fifth of an acre. The affidavit stresses the experimental nature of that farm and maintains that it cannot "be classified as a prototype for a practical marine biomass energy farm". *Id.* at pp. 5-6.

The Woodson affidavit next focuses upon the economic aspects of the production of energy through biomass conversion. It refers to certain analyses indicating that the unit (kwh) cost of electricity generated by a plant fueled with either substitute natural gas (derived from kelp) or ethanol (derived from algae) would greatly exceed that of nuclear-generated power. *Id.* at pp. 6-7.

The balance of the affidavit is directed to the reasons why Dr. Woodson believes (1) "it is clear that a commercial-scale marine biomass energy production system could not possibly be available until the year 2020 at the very earliest"; (2) no reasonable means exist whereby the applicant could obtain the use of the needed amount of sea space; and (3) potentially very significant environmental impacts would result from the creation of a

[5]As earlier seen, the applicant's statement of material fact adverted to a 576,000 acre farm. That figure did not take into account the energy requirements for such preprocessing activities as harvesting, transportation and drying. The Woodson affidavit avers (at p. 5) that, taking these activities into account, the estimated area needed increases to 960,000 acres (or 1,500 square miles).

massive marine biomass farm. *Id.* at pp. 7-11. On the last score, Dr. Woodson alludes both to the effect of covering approximately 1,000,000 acres of Gulf coastal waters with dense seaweed growing near the surface and to the serious problem which might be encountered in disposing of the residue after the energy is extracted. The root of that problem is that the residue ash would contain appreciable amounts of sulfur compounds, as well as other elements characterized by Dr. Woodson as "worrisome". In the affiant's view, the Environmental Protection Agency likely would classify the ash as hazardous, if not toxic, waste. *Id.* at p. 11.

B. In his response in opposition to the applicant's motion, and also on the basis of certain referenced studies, Mr. Potthoff asserts (at p. 4) that a marine biomass farm covering an area of 306 square miles (195,840 acres) would produce the energy needed to replace the Allens Creek facility.[6] This is said to represent five percent of the usable area in the Gulf of Mexico. That such a farm is a viable alternative to the nuclear plant is said to be demonstrated by the test farm to which Dr. Woodson referred. Describing it as a "prototype marine farm" with a 100-foot diameter, Mr. Potthoff maintains that it has provided "a friendly environment for tens of thousands of young kelp plants" and, further, has weathered severe storms. As he sees it, its design could be employed in the Gulf on a much larger scale. *Id.* at p. 3.

The response also asserts that a marine biomass farm would be environmentally preferable because it would not disturb either land or water resources. Although acknowledging that some pollutants would be generated, Mr. Potthoff insists that they could be removed through the use of "current pollution controls". *Id.* at p. 4.

C. After reviewing the content of the Woodson affidavit and Mr. Potthoff's response, the Licensing Board concluded that "it has been clearly established that a marine biomass farm is not now, nor, within the time frame of [Allens Creek], will it be a reasonable and feasible alternative" to the proposed nuclear plant. November 13 order, p. 25. In this connection, the Board noted (*id.* at p. 24, fn. 13) that the document cited by Mr. Potthoff in support of his reliance upon the results obtained from the California test farm contained the statement that "it must be remembered that this first test farm is in no way a prototype of what is perceived for large scale commerical farms".

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[6]In a later-filed supplement to his response, Mr. Potthoff noted that that figure assumed the use of kelp. Because the staff had submitted an affidavit in support of summary disposition which had indicated that kelp is a cold-water species which probably could not survive the warm environmental conditions prevailing in the Gulf of Mexico, Mr. Potthoff suggested the use instead of red algae, which he asserted would require a farm of comparable area.

II

As we observed two years ago in the *Hope Creek* proceeding:[7]

> The Supreme Court has embraced the doctrine, first enunciated in *Natural Resources Defense Council* v. *Morton*, 458 F.2d 827, 837-38 (D.C. Cir. 1972), that environmental impact statements need not discuss the environmental effects of alternatives which are "deemed only remote and speculative possibilities." *Vermont Yankee Nuclear Power Corp.* v. *Natural Resources Defense Council*, 435 U.S. 519, 551 (1978).

The question here thus is whether there is a genuine issue of material fact respecting the viability of the marine biomass alternative to Allens Creek which Mr. Potthoff insists should have been considered (as it concededly was not) in the staff's FES. If the Licensing Board rightly concluded that no such issue exists, it perforce follows that the alternative need not have been discussed in the FES and can properly be excluded from the hearings which are being held on the Allens Creek construction permit application.

Even when viewed in the light most favorable to him, it is quite apparent to us that Mr. Potthoff's response to the applicant's motion and supporting affidavit fell far short of countering the principal points made in those papers. Specifically, he offered little beyond naked assertions to buttress his claim that, contrary to the averments in the Woodson affidavit, it would be technologically, commercially and legally possible for the applicant to substitute a marine biomass farm for the proposed nuclear facility.

Insofar as technological feasibility is concerned, the Licensing Board was referred to nothing which might conceivably have suggested a reasonable likelihood that an operational marine biomass farm — whether 306 square miles in area or of the much larger size hypothesized by Dr. Woodson — could be in place by the end of this decade. It appears without contradiction that our country's research and development activities in this sphere of potential energy sources remains in a state of infancy. Indeed, the California "test farm" was the only concrete example of such an activity alluded to by either Dr. Woodson or Mr. Potthoff. Leaving aside the finding below that Mr. Potthoff's own references indicated that that farm is not intended to serve as a prototype for large-scale commercial farms (see p. 80, *supra*),[8] a long distance almost assuredly will have to be traversed before

[7] *Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 NRC 14, 38 (1979).

[8] In his appellate brief, Mr. Potthoff states (at p. 2) that "[i]f the [California] test farm does not prove the feasibility of such a large marine farm, then the experience of the Japanese does". He

the experience with a 100-foot diameter farm (covering approximately one-fifth of an *acre*) might bring about an operational commercial farm embracing several hundred square *miles*. [9]

Beyond these considerations, Mr. Potthoff's response was conspicuously silent on another, and equally crucial, aspect of his proposed alternative to the nuclear facility: the ability of the applicant to obtain control over a vast portion of the Gulf of Mexico and to foreclose any other use of it.[10] Once again, it matters not whether the marine biomass farm envisaged by Mr. Potthoff would involve 306 square miles of Gulf waters or, rather, might extend over an area several times that size. In either event, we are unaware of any mechanism — now available or likely to become available in the foreseeable future — whereby this or any other public utility would be able to appropriate for its own commerical purposes marine territory even approaching such dimensions.[11] The applicant having specifically

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then asserts that, in 1968, the Japanese grew 76,400 square miles (191,000 square kilometers) of seaweed on bamboo nets for use as a food "staple".

This point not having been raised below, it cannot be pressed on appeal. *Tennesse Valley Authority* (Hartsville Nuclear Plant, Units 1A, 2A, 1B, and 2B), ALAB-463, 7 NRC 341, 347-48 (1978). Beyond that, Mr. Potthoff's cursory description of the Japanese project is wholly insufficient to rebut the averments of the Woodson affidavit. In this regard, as the staff observes in its brief (at pp. 11-12), the successful cultivation of seaweed as a food source does not carry with it any implication respecting the use of marine biomass for large-scale energy production.

Mr. Potthoff also challenges for the first time on appeal Dr. Woodson's expert qualifications (Br. p. 2). We summarily reject that challenge both because it was not presented to the Licensing Board and because it is footless. While, as Mr. Potthoff stresses, Dr. Woodson is an electrical engineer, his service in recent years as the Director of the University of Texas Center for Energy Studies clearly provides a sufficient basis for his expression of an expert opinion on the matters addressed in his affidavit.

[9]On the appeal, Mr. Potthoff alludes (Br. p. 3) to a timetable contained in one of the references cited at p. 2 of the Woodson affidavit (*EPRI/GRI Workshop on Biomass Resources and Conversion*, WS 78-89, Electric Power Research Institute, Palo Alto, California, July 1979). According to that timetable (fig. 12-1 at p. 12-2), "concept validation" is to take place between 1973 and 1982; "system development" between 1982 and 1985; and "commercial prototype development" between 1985 and 1988. But, even if these estimates turn out to have been accurate, they do not advance Mr. Potthoff's claim (Br. p. 3) that a "marine biomass technology will be proven by 1988, the time [Allens Creek] is scheduled to come on line". We agree with the staff (Br. p. 13) that the term "commerical prototype" must be understood to refer to a demonstration model for the purpose of overcoming technological problems and, as such, cannot be taken as encompassing "a large-scale marine biomass energy system capable of replacing the energy to be produced by the proposed Allens Creek nuclear plant".

[10]In this regard, we can officially notice that the Gulf of Mexico is now laden with water transportation routes, oil wells, commerical fishing activities, *etc.* — none of which would be compatible with a marine biomass farm.

[11]As previously noted (fn. 8, *supra*), Mr. Potthoff has supplied very few details regarding the Japanese seaweed cultivation experience upon which he attempts to rely on this appeal. Common sense suggests, however, that that cultivation was not a single enterprise concentrated in one massive area but, instead, was carried out by many individuals or organizations in numerous and widely-dispersed areas along the coast of that country.

raised this point in its motion, it was incumbent upon Mr. Potthoff to confront it squarely as part of his overall demonstration that there is a triable issue respecting whether the biomass alternative is, in fact, a reasonable one.[12]

Affirmed.
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

Dr. Buck, concurring:

While I am in full accord with the result reached in this decision, I regret to have to say that the record before us has provided no cause for me to reconsider the views I expressed in dissent when this matter was last before us. See ALAB-590, 11 NRC 542, 553 (1980).

More specifically, as in my judgment was inevitable, the admission of Mr. Potthoff's contention VI to the proceeding as the result of ALAB-590 served no purpose other than to consume unnecessarily the time of the parties, the Board below and this Board. That time could have been much more profitably devoted to those issues in the proceeding truly deserving of serious consideration.

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[12]Indeed, by not controverting the applicant's statement in support of its motion (at p. 12) that "[t]here are currently no legal means of obtaining title to or exclusive use of the substantial amount of sea space required to build a marine biomass farm sufficient to replace" the Allens Creek facility, Mr. Potthoff may be deemed to have admitted that fact. See p. 78, *supra.* Because of our conclusion that Mr. Potthoff has not established the existence of a genuine issue regarding the technological and legal feasibility of his proposed alternative, it is unnecessary to consider that portion of the applicant's motion and the response thereto as was addressed to relative economic or environmental costs.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-466**

HOUSTON LIGHTING & POWER COMPANY
(Allens Creek Nuclear Generating Station, Unit No. 1) **February 3, 1981**

The Appeal Board denies intervenor's request for (1) directed certification of various licensing board rulings; (2) an order halting the progress of the hearing below pending the requested appellate review; and (3) a change in the composition of the Licensing Board.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

The practice of simultaneously seeking reconsideration by the Licensing Board of interlocutory rulings and appellate review of the same rulings is disfavored.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Appeal boards are disinclined to assume "the role of a day-to-day monitor" of the "numerous determinations" which must be made by licensing boards "respecting what evidence is permissible and in what procedural framework it may be adduced." *Toledo Edison Co.* (Davis-Besse Nuclear Power Station, Unit 1), ALAB-314, 3 NRC 98, 99 (1976).

RULES OF PRACTICE: DISQUALIFICATION

A motion to disqualify one or more members of a licensing board must be first presented to that board in strict conformity with 10 CFR 2.704(c); if the motion is denied, it will be routinely referred to an appeal board for review. 10 CFR 2.704(c).

Mr. James Morgan Scott, Jr., Sugar Land, Texas, for the intervenor Texas Public Interest Research Group.

MEMORANDUM AND ORDER

This construction permit proceeding is now in evidentiary hearing before the Licensing Board. On January 29, 1981, intervenor Texas Public Interest Research Group (TexPIRG) filed a motion with both that Board and this Board. The motion complains of a number of oral rulings and actions of the Licensing Board during the hearing week commencing on January 19 and ending on January 23, 1981. Insofar as addressed to the Licensing Board, in effect it asks for reconsideration of most, if not all, of those rulings and actions. The relief sought from us is (1) interlocutory appellate review of TexPIRG's grievances by way of directed certification[1] and (2) an order halting the progress of the hearing pending the outcome of that review. In addition, we are requested to direct a change in the composition of the Licensing Board.

1. We disapprove of the practice of simultaneously seeking Licensing Board reconsideration of interlocutory rulings and appellate review of the same rulings.

2. Should TexPIRG be dissatisfied with the Licensing Board's disposition of its motion for reconsideration, that party will than be free to file a petition for directed certification with this Board. In any such petition, TexPIRG must refer to the specific page or pages of the hearing transcript upon which each challenged ruling or action appears. Absent a precise record reference, the challenge will not be entertained by us.[2] Additionally,

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[1]See 10 CFR 2.718(i); *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 482-83 (1975).

[2]The motion now before us is devoid of record references; it instead invites us to "read the complete record of the hearing to date (one week) * * *". The Commission's Rules of Practice specifically require those appealing from initial decisions both (1) to "identify with particularity the portion of the decision (or earlier order or ruling)" which is being challenged; and (2) to "specify, inter alia, the precise portion of the record relied upon in support of [each] assertion of error". 10 CFR 2.762(a). Assuredly no less is to be expected of a party asking that we exercise our discretion to review licensing board rulings in advance of the rendition of the initial decision.

in determining the scope of the petition, TexPIRG would be well-advised to bear in mind our disinclination to assume "the role of a day-to-day monitor" of the "numerous determinations" which must be made by licensing boards "respecting what evidence is permissible and in what procedural framework it may be adduced". *Toledo Edison Co.* (Davis-Besse Nuclear Power Station, Unit 1), ALAB-314, 3 NRC 98, 99 (1976).

3. In pressing for the replacement of the entire Licensing Board assigned to the proceeding, TexPIRG asserts its doubt that "it can get a fair, impartial decision from [that] Board because of the obvious friction and tension between the Board and [its] attorney (no matter [whose] fault it is)". We need not pass now upon the substantiality of this assertion. A motion to remove (*i.e.,* disqualify) one or more members of a licensing board must be first presented to that board in strict conformity with the provisions of 10 CFR 2.704(c). If denied, the motion then is to be routinely referred to us for determination of "the sufficiency of the grounds alleged". *Ibid.*

Directed certification and allied relief *denied.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-466**

HOUSTON LIGHTING & POWER COMPANY
(Allens Creek Nuclear Generating Station, Unit No. 1) **February 4, 1981**

The Appeal Board denies an intervenor's request for directed certification of a ruling by the Licensing Board barring cross-examination on the testimony of a witness by another intervenor.

RULES OF PRACTICE: STANDING TO APPEAL

A party to a proceeding before a Licensing Board has no standing to press before an Appeal Board the grievances of other parties to the proceeding not represented by him. *Puget Sound Power and Light Co.* (Skagit Nuclear Power Projects, Units 1 and 2), ALAB-556, 10 NRC 30, 32-33 (1979); *Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-345, 4 NRC 212, 213 (1976).

RULES OF PRACTICE: STANDING TO APPEAL

A party to an NRC proceeding is not entitled to complain of a licensing board ruling unless and until that ruling has worked a concrete injury to his personal interests. *Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-252, 8 AEC 1175, 1177 (1975);

Toledo Edison Co. (Davis-Besse Nuclear Power Station), ALAB-157, 6 AEC 858, 859 (1973).

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Appeal boards will not normally invoke their discretionary directed certification authority for the purpose of monitoring the day-to-day conduct of licensing board evidentiary hearings.

APPEARANCES

Dr. David Marrack, Bellaire, Texas, intervenor *pro se.*

MEMORANDUM AND ORDER

David Marrack is one of a number of intervenors who are participants in this construction permit proceeding now in evidentiary hearing before the Licensing Board. Another participating intervenor is Wayne E. Rentfro.

During the course of the hearing session on January 23, 1981, Mr. Rentfro indicated a desire to pose two questions to an applicant's witness who was then testifying (Tr. 3841). Counsel for the applicant immediately interposed an objection on the ground that no part of the witness' testimony related to an issue within the scope of Mr. Rentfro's asserted interest in the proceeding (*ibid*). In this connection, counsel relied upon our holding some years ago in the *Prairie Island* proceeding[1] to the effect that:

> In both operating license and construction permit proceedings, an intervenor can and should be afforded the opportunity to cross-examine on those portions of a witness' testimony which relate to matters which have been placed into controversy by at least one of the parties to the proceeding — *so long as that intervenor has a discernible interest in the resolution of the particular matter.* [Emphasis supplied.]

In an accompanying footnote, we added:

> For this purpose, the extent of the intervenor's interest in the proceeding is to be ascertained on the basis of those relevant assertions in the intervention petition which were explicitly or implicitly accepted by the Licensing Board in connection with the grant of intervention.[2]

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[1]*Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-244, 8 AEC 857, 868 (1974), *reconsideration denied*, ALAB-252, 8 AEC 1175 (1975), *affirmed*, CLI-75-1, 1 NRC 1 (1975).
[2]8 AEC at 868, fn. 15.

After entertaining responses to the objection, the Licensing Board sustained it on the ground assigned by applicant's counsel (Tr. 3845). Thereafter, by way of clarification at the request of NRC staff counsel, the Board Chairman observed that the ruling would have little application beyond Mr. Rentfro because "I don't know of any other intervenor whose discernible interest is so miniscule and so limited and so restricted as is [his], which is very limited and restricted to adverse health impacts of high voltage transmission lines" (Tr. 3846). The Chairman did note that Dr. Marrack might "perhaps" be affected but added that "I'm not getting into that" (*ibid.*). Rather, he stressed, the ruling made by the Board applied only to Mr. Rentfro at that point; if the question arose again, "we will just have to rule on a party by party basis" (Tr. 3847).

What is now before us is a motion seeking review of the ruling under our directed certification authority. 10 CFR 2.718(i); *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 482-83 (1975).[3] The movant is not, however, Mr. Rentfro or anyone purporting to represent him in this proceeding. Instead, the motion was submitted by Dr. Marrack acting on his own behalf.

For at least two independent reasons, directed certification must be denied.

1. As just seen, the Licensing Board confined its ruling to Mr. Rentfro and whether that ruling will ever have application to Dr. Marrack is at best conjectural. Dr. Marrack has no standing to press before this Board the grievances of other parties to the proceeding who are not represented by him. *Puget Sound Power and Light Co.* (Skagit Nuclear Power Project, Units 1 and 2), ALAB-556, 10 NRC 30, 32-33 (1979); *Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-345, 4 NRC 212, 213 (1976). Nor is he entitled to complain himself of a licensing board ruling unless and until that ruling has worked a concrete injury to his personal interests. *Prairie Island*, ALAB-252, *supra*, 8 AEC at 1177; *Toledo Edison Co.* (Davis-Besse Nuclear Power Station), ALAB-157, 6 AEC 858, 859 (1973).

2. The question whether the Licensing Board correctly applied the *Prairie Island* cross-examination rule to Mr. Rentfro is scarcely worthy of our interlocutory examination. As we had occasion to reiterate in this proceeding just yesterday, we will not normally invoke our discretionary directed certification authority for the purpose of monitoring the day-to-day conduct of licensing board evidentiary hearings. ALAB-630, 13 NRC 84, 86.

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[3]Although its caption refers to the Licensing Board, the body of the motion makes clear that it is addressed to this Board alone.

In this connection, we do not understand Dr. Marrack to take issue with the Licensing Board's conclusion that Mr. Rentfro had not manifested a discernible interest in the matters to which the witness' testimony was addressed.[4] To the contrary, his dissatisfaction appears to be with the "discernible interest" requirement itself and the fact that its first application in this proceeding was to Mr. Rentfro. We have been given no cause, however, to reconsider our imposition of that requirement in *Prairie Island*.[5] And there is not substance to the claim that the "ground rules" for the hearing were changed in "mid session". The evidentiary hearing had commenced on January 16 and, as the Licensing Board pointed out when the same claim was presented to it, the "discernible interest" issue simply had not earlier surfaced in connection with proposed intervenor cross-examination of witnesses for other parties (Tr. 3845).

Directed certification *denied*.
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

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[4]We imply no opinion here on the correctness of that conclusion.
[5]It might be noted that the entire *Prairie Island* rule, including the "discernible interest" requirement, received explicit Commission endorsement. CLI-75-1, *supra*, 1 NRC at 2.

Cite as 13 NRC 91 (1981) **ALAB-632**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. John H. Buck

In the Matter of **Docket No. 50-471**

BOSTON EDISON COMPANY,
et al.
(Pilgrim Nuclear Power Station, Unit 2) **February 5, 1981**

The Appeal Board announces its tentative decision not to invoke the procedures of Appendix B to 10 CFR Part 2 (relating to consideration by an Appeal Board of the warrant for a stay) to the Licensing Board's partial initial decision (LBP-81-3) in this construction permit proceeding, subject to reconsideration upon the response of any party indicating disagreement with the basis for the Appeal Board's decision.

LIMITED WORK AUTHORIZATION: REQUIRED DETERMINATIONS

Before the Director of Nuclear Reactor Regulation may issue an LWA, a Licensing Board must make, *inter alia,* all the findings required by 10 CFR 51.52(b) and (c) to be made prior to the issuance of a construction permit for a facility. 10 CFR 50.10(e) (2).

RULES OF PRACTICE: SPECIAL STAY PROCEDURE (APPENDIX B TO PART 2)

The provisions of Appendix B to 10 CFR Part 2 which relate to the consideration by an appeal board of the warrant for a stay (either on motion or *sua sponte*) apply only to licensing board decisions which of themselves provide the underpinnings of "licensing action." *Duke Power*

Co. (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-626, 13 NRC 17, 19 (1981).

MEMORANDUM

On February 2, 1981, the Licensing Board issued a partial initial decision in this construction permit proceeding involving Unit 2 of the Pilgrim nuclear facility. LBP-81-3, 13 NRC 103. Although, as its caption states, that decision addresses "all matters except emergency planning and TMI-2 related issues", we do not take it as paving the way for the issuance by the Director of Nuclear Reactor Regulation of a limited work authorization (LWA) under 10 CFR §50.10(e) (1).

Before the Director may issue an LWA, Commission regulations require the Licensing Board to have made, *inter alia,* "all the findings required by §51.52(b) and (c) of this chapter to be made prior to issuance of the construction permit for the facility * * *".[1] Section 51.52(c) (3) directs the Board to strike an ultimate cost/benefit balance in compliance with the requirements of the National Environmental Policy Act; *i.e.,* to determine

> after weighing the environmental, economic, technical, and other benefits against environmental and other costs, and considering available alternatives whether the construction permit or license to manufacture should be issued, denied, or appropriately conditioned to protect environmetal values.

This has not as yet been done. To the contrary, in finding 384, 13 NRC at 203, the Board below expressly states:

> The costs and benefits of emergency planning and TMI-related issues have not been factored into this [NEPA] cost-benefit analysis. After evidentiary hearings on those issues are completed the Board will reassess its cost-benefit balance.

In the circumstances, it would appear that the partial initial decision does not bring into play that portion of Appendix B to 10 CFR Part 2 which relates to the consideration by an appeal board of the warrant for a stay (either on motion or *sua sponte*). Such consideration is called for only with respect to decisions which of themselves provide the underpinnings of "licensing action". *Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-626, 13 NRC at 19 (January 6, 1981).

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[1] 10 CFR §50.10(e) (2).

For this reason, this Board does not now propose to invoke the Appendix B procedures with regard to the February 2 decision.[2] Any party that disagrees with our interpretation of that decision may, however, so advise us in writing within 10 days of the service of this memorandum. Responses will be due within 10 days thereafter, following which the matter will be open for reconsideration.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

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[2]Although the Licensing Board's decision does not allow issuance of an LWA, it is nevertheless itself appealable now under the Rules of Practice because it disposes of a major segment of the case; our memorandum is not meant to imply otherwise. *Houston Lighting and Power Co.* (Allens Creek Station, Units 1 and 2), ALAB-301, 2 NRC 853, 854 (1975). *See also Toledo Edison Co.* (Davis-Besse Station), ALAB-300, 2 NRC 752, 758 (1975).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. Lawrence R. Quarles

In the Matter of **Docket No. 50-358 OL**

CINCINNATI GAS & ELECTRIC COMPANY, *et al.*
(William H. Zimmer Station) **February 9, 1981**

The Appeal Board dismisses as interlocutory an intervenor's appeal from the Licensing Board's summary disposition (LBP-81-2) of one of his contentions.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Under the Commission's Rules of Practice, appeals from interlocutory orders issued by licensing boards must await the "initial decision" rendered by the board at the end of the case. 10 CFR §§2.760 and 2.762.

APPEARANCES

Mr. John Woliver, Batavia, Ohio, for David Fankhauser, *intervenor.*

MEMORANDUM AND ORDER

Dr. David E. Fankhauser is an intervenor in this operating license proceeding. He seeks to appeal now from a Licensing Board ruling disposing summarily of one of his contentions. LBP-81-2, 13 NRC 36 (January 23, 1981). Because that ruling did not eliminate Dr. Fankhauser as a party in the proceeding it is an interlocutory order. Under the Rules of Practice, appeals from such orders must await the "initial decision" rendered by the Board at the end of the case. 10 CFR §§2.760 and 2.762;

Boston Edison Co. (Pilgrim Station, Unit 2), ALAB-269, 1 NRC 411, 413 (1975).

We therefore dismiss Dr. Fankhauser's present appeal for want of jurisdiction. We do so on our own motion to spare all counsel the necessity of briefing a matter we may not now entertain. This dismissal is of course without prejudice and carries no implication about the merits of the Licensing Board's ruling.

Appeal *dismissed.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

This order is issued by the Board Chairman under 10 CFR §2.787(b)(1). Dr. Quarles took no part in the disposition of this matter.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket Nos. 50-329 OM & OL**
50-330 OM & OL

CONSUMERS POWER COMPANY
(Midland Plant, Units 1 and 2)

February 19, 1981

The Appeal Board declines to accept the Licensing Board's referral under 10 CFR 2.730(f) of a discovery ruling compelling the deposition of a named NRC staff member.

RULES OF PRACTICE: DISCOVERY (AGAINST NRC STAFF)

The staff may take the deposition of other parties in Commission proceedings subject only to the general strictures on discovery contained in the Rules of Practice. Discovery against the staff, however, is more limited.

RULES OF PRACTICE: DISCOVERY (AGAINST NRC STAFF)

The Executive Director for Operations designates those staff members available to respond to discovery requests; others may be deposed only if a Licensing Board allows it upon a showing of "special circumstances." Those circumstances are present, for example, if the staff member who is sought to be deposed "has direct personal knowledge of a material fact not known to the witnesses made available by the Executive Director for Operations." 10 CFR 2.720(h)(1) and (2).

RULES OF PRACTICE: DISCRETIONARY INTERLOCUTORY REVIEW

Interlocutory appeals are not favored in Commission proceedings any more than in judicial practice. Whether review should be undertaken on certification pursuant to 10 CFR 2.718(i) or by referral under 10 CFR 2.720(f) before the end of the case turns on whether a failure to address the issue would seriously harm the public interest, result in unusual delay or expense, or affect the basic structure of the proceeding in some pervasive or unusual manner. Discovery rulings rarely meet those tests.

RULES OF PRACTICE: REFERRAL OF RULINGS

Appeal Boards may not second guess a Licensing Board in its discovery rulings. Rather, they may take up referrals of those rulings only if there is an indication that the Licensing Board abused the discretion the Commission gave it in this area.

RULES OF PRACTICE: DISCOVERY

Whether or not the person to be deposed will later be called to testify as a witness is irrelevant for discovery purposes. The test is, rather, whether the information sought appears reasonably calculated to lead to the discovery of admissible evidence. 10 CFR 2.740.

APPEARANCES

Mr. Michael I. Miller, Chicago, Illinois, for the Consumers Power Company, *applicant.*

Mr. Bradley W. Jones for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

1. The staff may take the depositions of other parties in Commission proceedings subject only to the general strictures on discovery contained in the Rules of Practice. Discovery against the staff, however, is more limited.[1] One limitation pertinent to the matter before us is that the Executive Director for Operations designates those staff members available to

[1] *See, Pennsylvania Power and Light Co.* (Susquehanna Station, Units 1 and 2), ALAB-613, 12 NRC 317, 321 (1980).

respond to discovery requests; others may be deposed only if a Licensing Board allows it upon a showing of "special circumstances." An example of special circumstances recognized in the rules themselves is a situation in which the staff member who is sought to be deposed "has direct personal knowledge of a material fact not known to the witnesses made available by the Executive Director for Operations * * *." 10 CFR §2.720(h)(1) and (2).

In this case the applicant sought to depose Mr. Harold Thornburg, a member of the staff whom the Director had not made available. For reasons explained in a written opinion, the Licensing Board determined that Mr. Thornburg appeared to have information "essential" to the proper evaluation of an issue before it in the proceeding. On the ground that this information was neither available from other sources nor "possessed by others made available by the staff," the Board found that exceptional circumstances justified deposing Mr. Thornburg. LBP-81-4, 13 NRC 216, 221-23 (February 12, 1981). It subjected the taking of his deposition, however, to two conditions: (1) that certain other individuals made available by the staff be deposed first; and (2) that if they do not possess the information sought, then Mr. Thornburg could be questioned only on factual matters, with an express provision that he need not reveal the staff's deliberative process or his recommendations to his superiors.[2] *Id.* at 223-24.

2. Discovery rulings involving parties to the case are interlocutory and not appealable at this stage.[3] At the staff's request, the Board below "referred" its ruling to us under 10 CFR §2.730(f) on the possibility, not elaborated in its opinion, that it "might have public interest implications."[4] In doing so, however, the Board voiced strong belief in the correctness of its ruling and declined to postpone the taking of Mr. Thornburg's deposition. *Id.* at 224.

Section 2.730(f) does not oblige us to accept all referred rulings.[5] Because Mr. Thornburg's deposition is imminent, we invited the staff, the applicant, and any other party wishing to be heard to file memoranda addressing whether the ruling warrants our consideration now and, if so, whether we should stay the taking of his deposition until our decision on

[2]Assuming that it is necessary to do so, Mr. Thornburg is scheduled to be deposed on February 20.

[3]See *Toledo Edison Co.* (Davis-Besse Station), ALAB-300, 2 NRC 752, 768-69 (1975).

[4]10 CFR §2.730(f) provides in pertinent part that, "[n]o interlocutory appeal may be taken * * * from a ruling of the presiding officer. When in the judgment of the presiding officer prompt decision is necessary to prevent detriment to the public interest or unusual delay or expense, the presiding officer may refer the ruling * * *."

[5]*Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-438, 6 NRC 638 (1977); *Public Service Co. of Indiana, Inc.* (Marble Hill Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1191 (1977); *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-116, 6 AEC 258 (1973).

the merits of the referred question. The staff and the applicant responded to our invitation, the former favoring the referral and the latter opposing it.

3. Interlocutory appeals are not favored in Commission any more than in judicial practice.[6] Whether review should be undertaken on "certification"[7] or by referral before the end of the case turns on whether a failure to address the issue would seriously harm the public interest, result in unusual delay or expense, or affect the basic structure of the proceeding in some pervasive or unusual manner.[8] Discovery rulings rarely meet those tests.[9]

Our reluctance to take up interlocutory discovery matters is not arbitrary. It simply recognizes that deciding such things as whether an interrogatory "relate[s] to matters in controversy," or is "relevant to the subject matter of the proceeding," requires a nice familiarity with the case that the trial board possesses and we do not.[10] The same considerations govern the issue here — whether "exceptional circumstances" have been demonstrated so as to require deposing a named staff member.

Our disinclination to enter the discovery thicket mirrors the Commission's own renitence in this regard. Before 1975, the Rules of Practice required interlocutory review by the Commission itself before a board could compel discovery of Commission personnel or documents.[11] The Commission deleted that requirement and amended Section 2.720(h) to its present form as a result of discovery requests arising in proceedings involving this very reactor. *Consumers Power Company* (Midland Plant), CLI-74-27, 8 AEC 4 (1974).[12] The Commission did so explicitly because it doubted the usefulness of interlocutory review of discovery requests. The Commission stressed in *Midland* that (8 AEC at 6):

> A requirement of interlocutory review contributes to delay in adjudication, forcing parties and boards to await Commission rulings before proceeding. Second, the rules require that this Commission review board determinations as to what details of discovery are necessary to a proper decision and what materials are not reasonably obtainable elsewhere. As we explained in *Virginia Electric and Power Co.* [CLI-74-16, 7 AEC 313 (1974)], in procedural matters, only the Licensing

[6]*Davis-Besse, supra,* ALAB-300, 2 NRC at 758.

[7]See 10 CFR §2.718(i).

[8]*See, Marble Hill, supra,* ALAB-405, 5 NRC at 1192-93; *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 483-86 (1975).

[9]See, *e.g., Long Island Lighting Co.* (Jamesport Station, Units 1 and 2), ALAB-318, 3 NRC 186 (1976); *Davis-Besse, supra,* ALAB-300, 2 NRC at 769.

[10]We have previously acknowledged that discovery "matters are particularly within a trial board's competence and appellate review of such rulings is usually best conducted at the end of the case." *Susquehanna, supra,* ALAB-613, 12 NRC at 321.

[11]10 CFR §2.720(h)(2)(ii) and 2.744(e) (1974).

[12]See also, 40 *Fed. Reg.* 2973 (January 17, 1975).

> Boards have "first-hand contact with and appreciation for all the circumstances" surrounding a particular case. For this reason, the Commission is hardly in a position to second guess its own boards with respect to such factual details. Finally, the rule accords a preferential status to Commission personnel and documents, while discriminating against other parties (applicants and intervenors) who cannot invoke Commission review of their refusals to respond to discovery.

The rule change permitted "the presiding officer in his discretion" to allow discovery from the staff (40 Fed. Reg. at 2973), and reflected an express Commission purpose to place "maximum reliance" on his determinations. See 8 AEC at 5. We must, of course, implement that policy.

4. Returning to the facts of this case, we note that the Licensing Board set forth at length its reasons for concluding that the applicant must be allowed to depose Mr. Thornburg. Our ruling on the merits of that question might or might not be the same. But at our present distance from the proceeding, we doubt that our judgment would be better than the Licensing Board's. For the reasons just stated, we may not "second guess" that Board. Rather, we may take up this referral only if there is an indication that the Board abused the discretion the Commission gave it in this area.

The staff's papers proffer essentially three reasons for granting interlocutory review with an eye to overturning the discovery order in question: (1) that the Board below improperly allowed the deposition of a "non-witness;" (2) that a "senior staff division director" is immune from the discovery process even in exceptional circumstances; and (3) taking Mr. Thornburg's deposition now will perforce vitiate a staff claim of "executive privilege." None of them is persuasive.

(a) To begin with, whether or not the person to be deposed will later be called to testify as a witness is irrelevant for discovery purposes. The test is, rather, whether "the information sought appears reasonably calculated to lead to the discovery of admissible evidence." 10 CFR §2.740. It is frequently not until after a deposition is taken that a party can tell whether the deponent will be needed as a testimonial witness.[13] The staff's objection puts the cart before the horse.

(b) The rules in terms exempt all NRC personnel from responding to discovery requests unless the Executive Director for Operations makes them available or a licensing board finds that "exceptional circumstances" necessitate their doing so. Tacitly recognizing that the rules contain no express exception for "a senior staff division director," the staff would

[13]See, generally, 4 *Moore's Federal Practice* (1979 ed.) ¶26.56[4].

apparently have us infer one. We note, however, that the identical provision of the 1975 amendments to Section 2.720(h) that placed both "the Commission and named NRC personnel" beyond the range of a licensing board subpoena expressly allows the subpoena "of named NRC personnel" in "exceptional circumstances." In short, the Commissioners considered whether to allow any exceptions from the reach of a licensing board's discovery subpoena in "exceptional circumstances" — and chose to exempt only themselves.

The staff suggests no basis for reading into Section 2.720(h) (2) (ii) an additional exception for a "senior staff division director." We therefore cannot conclude that the Licensing Board abused its discretion in declining to do so, particularly when that Board believes that the individual in question might have important and material information not available elsewhere about a serious safety question in issue before it. See 13 NRC at 221-22.[14]

(c) Finally, the staff sees the subpoena to Mr. Thornburg as sanctioning an inquiry into the staff's decisional processes and "privileged communications" among senior officials. The short answer is that the Board's order in terms allows inquiry only into "facts" communicated by Mr. Thornburg to his superiors and proscribes inquiry into his recommendations to them or other deliberative information. 13 NRC at 223-24. Should the applicant's questions stray into protected areas, as the applicant itself concedes, "the staff may object and instruct Mr. Thornburg to refuse to answer pending a ruling by the Licensing Board." (Applicant's memorandum at 5-6.) In these circumstances, the staff will not waive any claim of "executive privilege" merely because Mr. Thornburg is deposed. There is time enough to consider that difficult question when (and if) it actually arises.[15]

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[14]*Cf., Virginia Electric & Power Co.* (North Anna Station), CLI-74-16, 7 AEC 313, 314-15 (1974).

[15]While we do not reach the issue, the existence of that "executive privilege" is touched upon by the Board below, 13 NRC at 223 and by Judge Smith in *Consumers Power Co.* (Palisades facility), ALJ-80-1, 12 NRC 117 (1980). See also *Virginia Electric & Power Co.* (North Anna Station), *supra*, CLI-74-16, 7 AEC 313.

5. In sum, the Board's order was not abusive of delegated discretion and we find no cause to review it now. Accordingly, the referral is *declined;* our disposition of the matter leaves no occasion to consider whether the deposition should be stayed.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Andrew C. Goodhope, Chairman
Dr. A. Dixon Callihan
Dr. Richard F. Cole

In the Matter of **Docket No. 50-471-CP**

BOSTON EDISON COMPANY,
et al.
(Pilgrim Nuclear Power Station, Unit 2) **February 2, 1981**

In a partial initial decision, the Licensing Board rules that a permit to construct Unit 2 of the Pilgrim facility should be issued, subject to certain conditions, upon favorable completion of hearings on emergency planning and issues related to the Three Mile Island accident as well as a reassessment of the cost-benefit balance for the plant following those hearings.

FINANCIAL QUALIFICATIONS: APPLICABLE STANDARD

An applicant for a construction permit must show that it either possesses or has reasonable assurance of being able to obtain the funds necessary to cover estimated construction and related fuel costs. 10 CFR 20.33(f) and 10 CFR Part 50, Appendix C. A reasonable assurance does not mean a demonstration of near certainty that an applicant will never be pressed for funds in the course of construction. It does mean that the applicant must have a reasonable plan in light of relevant circumstances. *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-78-1, 7 NRC 1, 18 (1978).

NEPA: "FEDERAL ACTION"

Licensing the construction of a nuclear power plant is a "major federal action" within the meaning of Section 102(2)(C) of the National Environmental Policy Act. That section requires the Commission to consider whether reasonable alternatives less harmful to the environment exist before allowing a utility to proceed with construction.

NEPA: CONSIDERATION OF ALTERNATIVES

To satisfy NEPA, the NRC must identify, study and compare alternative sites for the location of a proposed facility. In determining whether a proposed site is environmentally acceptable, the Board must find that after giving each alternative site a hard look, none is found obviously superior to the one proposed by the applicant. *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-77-8, 5 NRC 503 (1977).

TECHNICAL ISSUES DISCUSSED:

- Technical qualificatons
- Financial qualifications
- Theft and sabotage
- Generic safety problems
- Steam generator tube integrity
- Quality Assurance
- Compliance with Appendix I
- Site suitability
  - LPZ and population center distance requirements
  - Industrial and transportation hazards
  - Hydrology and meteorology
  - Geology and seismology
- Environmental matters
  - Need for power
  - Impacts of construction
  - Impacts of operation
  - Radiological impacts
  - Fuel availability
  - Waste disposal facilities
  - Consideration of alternatives
  - Alternate sites
  - Alternative sources of energy
  - Alternate condenser cooling

Environmental effects of postulated accidents
Aircraft crash risk
Release of radioactive materials in effluents
Transportation of nuclear material
As low as reasonably achievable (ALARA)

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Patrick J. Kenny, Esq., General Counsel

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PARTIAL INITIAL DECISION

FINDINGS OF FACT AND CONCLUSIONS OF LAW ON ALL MATTERS EXCEPT EMERGENCY PLANNING AND TMI-2 RELATED ISSUES

I. PRELIMINARY STATEMENT

1. On June 7, 1973, pursuant to § 103 of the Atomic Energy Act of 1954, as amended (the Act), Boston Edison Company (BECo) filed with the Atomic Energy Commission, now the Nuclear Regulatory Commission (NRC or Commission), an application on behalf of itself and ten public utility companies and eleven municipal light departments or plants (the Applicants)[1] for authorization to construct an 1180 megawatt electric (MWe) pressurized water reactor, designated as Pilgrim Unit 2, (Unit 2) to be located on the western shore of Cape Cod Bay in Plymouth County, Massachusetts. At the same time, BECo filed on its own behalf a similar application for Pilgrim Unit 3, to be built at an adjacent location. After revision, the applications were resubmitted on November 24, 1973, and the Commission docketed them as Nos. 50-471 and 50-472, respectively, on December 21, 1973. A Notice of Hearing on the applications was published at 39 FR 1786 on January 14, 1974, which ordered a hearing to consider issues pursuant to the Act and to the National Environmental Policy Act of 1969 (NEPA) (42 U.S.C. § 4321 *et seq.*).

2. Pursuant to the Commission's Notice of Hearing, timely petitions to intervene were filed by the Commonwealth of Massachusetts (Commonwealth), the Massachusetts Wildlife Federation (MWF), Daniel F. Ford (Ford), and Alan and Marion Cleeton (Cleetons). A special prehearing conference was held pursuant to 10 CFR §2.751a on April 19, 1974, to consider these petitions and other matters. By Memorandum and Order of May 30, 1974, the Board admitted as parties to the proceeding the Commonwealth, MWF, Mr. Ford, and Mr. and Mrs. Cleeton in light of their interests and the identification of at least one valid contention.

3. A non-timely petition to intervene was filed on July 15, 1974, by William S. Abbott on behalf of the Plymouth County Nuclear Information

[1]Since the filing of the original application the utility systems participating as Applicants have changed. The present fourteen Applicants are Boston Edison Company, The Electric Light Department of the City of Burlington, Central Maine Power Company, Central Vermont Public Service Corporation, Fitchburg Gas and Electric Light Company, Town of Hudson Light and Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New Bedford Gas and Edison Light Company, New England Power Company, Public Service Company of New Hampshire, The United Illuminating Company, Tauton Municipal Lighting Plant Commission, and Vermont Electric Cooperative, Inc.

Committee. That petition was opposed by both the Applicant and the Staff and was supported by the Commonwealth. On August 30, 1974, the Board denied the late petition to intervene and the Appeal Board thereafter affirmed.[2]

4. Prehearing conferences were held on July 15, October 3 and December 4, 1974 on the contentions proposed by intervenors. By Memorandum and Order dated February 18, 1975, the Board ruled on the Parties' contentions stated here in summary and in detail in Part V *infra*. The following contentions of the Commonwealth were admitted.

1(a)-(h).	the effects of operation on the Cape Cod ecosystem;
2.	alternative cooling systems;
3.	alternative energy sources;
4.	alternative sites;
5.	Financial qualifications;
6.	the need for power;
8.	overstatement of production of electrical energy;
9.	the risk of theft and sabotage;
10.	technical qualifications of the Applicants, Bechtel Corporation, and Combustion Engineering, Inc.;
11.	the inadequacy of the NRC inspection programs; and
12.	alternate siting from a population density and environmental standpoint.

[2]*Boston Edison Company* (Pilgrim Nuclear Generating Power Station, Unit 2), ALAB-238, 8 AEC 656 (October 22, 1974).

Contentions 13 and 14 were also admitted but were withdrawn by the Commonwealth in a letter dated November 17, 1975. The Board admitted the following MWF contentions:

1(a).	compliance with the Commission's "as low as practicable" standards; and
1(b).	failure to consider alternate sites.

MWF Contentions 2(a), (b), (d), (e) and (f) and 4 were admitted by the Board but were subsequently withdrawn. (Tr. 781, 3679-3680) MWF Contentions 2(c), 3, and 5 were also admitted but were withdrawn as a result of a settlement agreement between the Applicants and MWF. (Tr. 6360-61, 6460) The Board rejected MWF Contentions 6-10 as factual contentions holding that these were more appropriately to be addressed as legal issues.[3]

The Board accepted the following contentions of the Cleetons:

B.	transportation risks;
C.	aircraft risks;
E.	routine discharges of effluents;
H.	the need for power;
I.	alternate sources of power; and
K.	unavailability of adequate nuclear fuel.

5. Prior to the commencement of the evidentiary hearings, Intervenor Ford informed the Board by letter dated October 15, 1975, that he did not intend to participate in the evidentiary hearings but that he reserved the right to seek "administrative and judicial review." On October 30, 1975, on motion of the NRC Staff (Staff), the Board issued an Order directing Mr. Ford to show cause why he should not be held in default and why certain of

[3]On January 8, 1977, MWF served a Memorandum with respect to its Contentions 6 through 10 in which it announced it would only pursue one legal contention that of the legality of Sec. II-D of Appendix I to 10 CFR Part 50, which relates to a balancing of a dollar value per man-rem with augments in the radwaste system. In lieu of proposed findings of fact and conclusions of law, and in order to preserve its rights on appeal, MWF served on November 30, 1979 a statement describing its exceptions to a Board Order dated July 14, 1978 concerning the application of Sec. II-D of Appendix I to 10 CFR 50 to this proceeding.

his contentions should not be dismissed from the proceeding. After a response was filed by Mr. Ford on November 14, 1975, the Board issued an Order on February 20, 1976, holding Mr. Ford in default because of his failure to meet the responsibilities of his participation in this proceeding. The Board reviewed the Ford contentions, however, to ensure that his legitimate concerns would be considered at the evidentiary hearings. All except one of his contentions were dismissed because they were included in those of other parties or lacked specificity. The remaining contention on the integrity of steam generator tubes was made the subject of Board inquiry.

6. The decisional record of this proceeding consists of a) the Commission's Notice of Hearing; b) the petitions and pleadings filed by the parties; c) the memoranda and orders of the Board; d) the transcript of the hearing; and e) the exhibits received into evidence. The principal documents filed by the Applicants are the Preliminary Safety Analysis Report (PSAR) as amended (Applicants' Exhibits 1-B through 1-J, 1-N through 1-BB, 23, 24 and 25); the Environmental Report (ER) as amended (Applicants' Exhibits 1-K, 1-L, 1-M and 1-CC). The Staff's principal documents include the Safety Evaluation Report (SER) as amended (Staff's Exhibits 4, 5, 7, 21 and 50); the Draft Environmental Statement (DES); the Final Environmental Statement (FES) (bound following Tr. 897) as amended including the Final Supplement to the FES (received at Tr. 9852 and bound following Tr. 9952), and Staff Exhibits 10, 11-A, 11-B, 11-C, 13, 14, 15, 16, 17, 19, 53 and 66); and the NRC Site Suitability Report (SSR, Staff Exhibit 9).

7. On June 18, 1974, the Staff issued the Draft Environmental Statement (DES) for the proposed Pilgrim Units 2 and 3 which addressed the environmental impacts of construction and operation of the two units. Comments on the statement were received from the Applicants, from a number of federal and state agencies, and from an individual.

8. After the DES had been issued, and prior to the issuance of the Final Environmental Statement (FES), BECo, in June 1974, advised the Commission that construction of Unit 3 was to be deferred until the need for its output was established. BECo submitted a motion on July 1, 1974 requesting withdrawal of Unit 3 since the financial commitment for its construction would not be prudent. On August 9, 1974, the Board found that BECo had demonstrated good cause for withdrawal of the Unit 3 application and imposed no conditions upon the withdrawal.

9. Since most of the environmental impacts of Units 2 and 3 had been addressed in the DES, both separately and collectively, the Staff found an issuance of a draft statement for Unit 2 alone to be unnecessary.

10. Pursuant to the Board's August 9, 1974 Order, the Staff submitted to the Board and the parties on August 20, 1974, a document summarizing the changes in the proposed FES resulting from the Applicants' withdrawal of the application for Unit 3.[4] After reviewing this document, the Board determined that recirculation of the changes to the appropriate agencies was advisable and directed the Staff accordingly. The Staff's motion for reconsideration of the Board's Order to seek these additional comments was denied.[5]

11. Pursuant to the Board Order of October 10, 1974, the Staff distributed on November 15, 1974, the "Summary of New and Revised Sections" to various agencies and also requested comments from interested persons in a *Federal Register* Notice published on November 12, 1974 (39 FR 40881). The Staff published[6] a response to comments received from several agencies.

12. The FES assessing the benefits and costs of the proposed Unit 2, based on the DES but reflecting the absence of Unit 3, was prepared by the Staff under date of September 1974. The FES and a supplement reporting the Staff's response to the comments on the DES received as a result of the withdrawal of Unit 3 were received at Tr. 897.[7] Certain additional supplements and corrections to the FES introduced into the proceeding subsequently will be cited later as appropriate.

13. On June 27, 1975, the Staff issued its Safety Evaluation Report (SER)[8] for the proposed Unit 2 containing the Staff's evaluation of safety aspects of the proposed facility, including Section 2 relating to the characteristics of the proposed site. Supplements Nos. 1, 2, 3 and 4 to the

[4]"Summary of New or Revised Sections of the Final Environmental Statement for Pilgrim Nuclear Power Station Unit 2 which were Required as a Result of Withdrawal of the Application for Pilgrim Nuclear Power Station Unit 3."

[5]Circulation of the summary of "new and revised sections" to those agencies, organizations and individuals from whom comment on the DES was requested was ordered by the Board under date of September 6, 1974. After denial of a motion for reconsideration, filed by the Staff on September 13, the order was reaffirmed by the Board orally on October 3 at Tr. 243 and by an Order filed October 10, 1974.

[6]"Response to Comments on the Summary of New or Revised Sections of the Final Environmental Statement for Pilgrim Nuclear Power Station, Unit 2 which were Required as a Result of Withdrawal of the Application for Pilgrim Nuclear Power Station, Unit 3, Final Version" (May, 1975), following Tr. 897. As noted by the Staff at fn 1, p. 2 of this "Response," some of the comments were directed to the FES and not to the Summary of the New and Revised Sections.

[7]The concurrence of the Environmental Protection Agency with the Staff's decision against reissuing the DES following withdrawal of Unit 3 appears at A-47, FES.

[8]Staff Exhibit 4, following Tr. 3717.

SER, were issued by the Staff[9] on November 3, 1975, January 27, 1976, August 31, 1977, and January 19, 1979. Comments and recommendations on the Application and SER by the Advisory Committee on Reactor Safeguards (ACRS) were transmitted to the Commission[10] [11] on November 14, 1975, and on October 12, 1977.

14. Evidentiary hearings commenced in Plymouth, Massachusetts on October 20, 1975 and continued intermittently until July 1, 1977 when the sessions were adjourned for the filing of proposed findings of fact and conclusions of law on a request by the Applicants for a Limited Work Authorization (LWA) Request for Unit 2 dated October 13, 1976.

15. On November 30, 1977, the Board issued a partial inital decision[12] denying the request on the basis of an incomplete record on possible alternate sites for Unit 2.

16. Evidentiary sessions resumed on March 6, 1978, and continued from time to time until August 28, 1979 concluding hearings on all of the then-established issues[13] except emergency planning,[14] which was deferred indefinitely at the request of the Staff.[15]

17. During the course of the proceedings the Board heard a number of limited appearance statements from members of the public. These statements have been considered by the Board in this partial initial decision.

18. In preparing the following findings of fact and conclusions of law, the Board reviewed and considered the entire record in this case and the findings of fact and conclusions of law proposed by the parties. Those proposed findings not incorporated directly or inferentially in this Partial Initial Decision are rejected as being unsupported by the record of this case or as being unnecessary to the rendering of this decision.

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[9]Staff Exhibits 5, 7 and 21 respectively, following Tr. 3717, Tr. 5394, Tr. 8921; Staff Exhibit 50 received at Tr. 9509 and bound following Tr. 10046.

[10]Staff Exhibit 7, SER Supplement 2, Appendix B, pp. 1-3, following Tr. 5394.

[11]Staff Exhibit 50, SER Supplement 4, Appendix B at B-1 to B-3, following Tr. 10046.

[12]Partial Initial Decision Regarding Request For Limited Work Authorization, 6 NRC 839 (November 1977). Affirmed, Boston Edison Company (Pilgrim Nuclear Generating Station, Unit 2), ALAB-479, 7 NRC 774 (1978).

[13]During this period hearings were held on certain remaining health and safety issues, financial qualifications, alternate site re-review and the reopened need for power issue. Since that time certain additional issues have arisen. All are directly related to NRC task orders resulting from the Three Mile Island (TMI) incident of March 28, 1979.

[14]On April 27, 1978 and April 4, 1979 the Commonwealth by separate motions sought to reopen the issue of "need for power" and to introduce contentions relating to emergency planning. On May 9, 1979, the Board granted Commonwealth's motion to reopen the need for power issue and on May 24, 1979 granted Commonwealth's motion to admit a late-filed contention on emergency planning.

[15]Board Order dated September 13, 1979 granted Staff motion to defer hearings until the Staff has completed its review of emergency planning at the Pilgrim site.

II. FINDINGS OF FACT - RADIOLOGICAL HEALTH AND SAFETY MATTERS

A. General

19. The Notice of Hearing issued with respect to this proceeding dated January 9, 1974 and published in the *Federal Register* on January 14, 1974 (39 FR 1786), requires the Board, pursuant to the Atomic Energy Act of 1954, as amended, to consider and decide:

> "1. Whether in accordance with the provisions of 10 CFR § 50.35(a):
>
> (a) The applicants have described the proposed design of the facilities including, but not limited to, the principal architectural and engineering criteria for the design, and have identified the major features or components incorporated therein for the protection of the health and safety of the public;
>
> (b) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;
>
> (c) Safety features or components, if any, which require research or development have been described by the applicants and the applicants have identified, and there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features or components; and
>
> (d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the applications for completion of construction of the proposed facilities, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public.
>
> "2. Whether the applicants are technically qualified to design and construct the proposed facilities;
>
> "3. Whether the applicants are financially qualified to design and construct the proposed facilities; and
>
> "4. Whether the issuance of permits for construction of the facilities will be inimical to the common defense and security or to the health and safety of the public."

The notice of hearing further states that... "in the event this proceeding becomes a contested proceeding" [which it is]... "the Board will consider and initially decide, as issues in the proceeding, Items 1-5"...[the above listed four items pertaining to the Act and Item 5 pertaining to implementation of the National Environmental Policy Act of 1969 stated in Section IV of this decision]..."as a basis for determining whether construction permits should be issued to the Applicants."

II. B. Facility Description and Compliance with 10 CFR § 50.35(a)

20. The proposed facility is described in the Preliminary Safety Analysis Report (PSAR) as amended (Applicants' Exhibits 1-B through 1-J, 1-N through 1-BB), in the Environmental Report (ER) as amended (Applicants' Exhibits 1-K through 1-M, and 1-CC) and in the Staff's SER as amended (Staff Exhibits 4, 5, 7, 21, and 50 and FES (following Tr. 897) as amended by Staff Exhibits 10 through 11-C, 13 through 15, 19 and 53.

21. Pilgrim Unit 2 is proposed to be located on the western shore of Cape Cod Bay in Plymouth, Massachusetts on a 528 acre site adjacent to Pilgrim Unit 1, an operating 655 MWe boiling water reactor. (PSAR §§1.1, 1.2.1; SER § 2.0)

22. The nuclear steam supply system (NSSS) will consist of a Combustion Engineering, Inc. (CE) pressurized water reactor and a two loop reactor coolant system rated at a thermal power output of 3473 MW. Each loop of the reactor coolant system will consist of an outlet pipe (hot leg), one steam generator, two inlet pipes (cold legs) and two reactor coolant pumps, one in each cold leg. An electrically heated pressurizer will be connected to one loop to establish and maintain the reactor coolant pressure. The reactor core will be composed of uranium dioxide pellets enclosed in Zircaloy-4 tubes with welded end plugs. Water will serve both as the neutron moderator and the coolant and will be circulated through the reactor vessel and core by four reactor coolant pumps. The heated water will flow through two steam generators where heat will be transferred to the secondary system and ultimately converted to electric energy in the turbine generator. The reactor will be controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. The control elements, whose drive mechanisms will penetrate the top of the reactor vessel, will be moved vertically within the core by individual control rod drive mechanisms. A plant protection system which automatically initiates action when preestablished limits are approached will shut down the reactor, close isolation valves and initiate operation of the engineered safety features should they be required. (SER § 1.2, PSAR §§ 1.2.5, 1.2.6)

23. The NSSS will be housed in a steel-lined reinforced concrete structure designed and constructed by the Bechtel Corporation (Bechtel) and prestressed by post-tensioned tendons. (SER § 1.2)

24. The reactor core for Unit 2 is similar to the design approved for San Onofre Units 2 and 3. (Docket Nos. 50-361 and 50-362) The Unit 2 core will contain 217 fuel assemblies, each with a 16 x 16 rod array. CE is committed to perform tests to verify the adequacy of the fuel assembly mechanical design, to finalize values for thermal, hydraulic and structural design parameters and to develop analytical models for confirming that the design meets specified criteria. (SER § 4.0) Each fuel assembly will consist of 236 fuel or fuel-poison rods with pellets of about 1.9 to 3.0 percent U-235 enriched uranium oxide at 95 percent theoretical density, sealed in Zircaloy tubes pressurized with helium. (PSAR Table 1.3-1) The differences from the San Onofre fuel design previously reviewed and approved by the Staff are geometric (San Onofre employs a 14 x 14 array) and will result in a lower linear power density in the Unit 2 fuel rods, thus increasing thermal performance margins. (SER § 4.2.1)

25. The principal components of the reactor coolant system for the facility consist of a reactor vessel, two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps, and a pressurizer connected to one of the reactor vessel outlet pipes. All components of the system will be located inside the containment building. (SER § 5.1)

26. The containment systems will include the reactor containment structure, heat removal system, air purification and clean up system, isolation system, combustible gas control system and provisions for containment leakage testing. (SER §§ 6.1, 6.2) The containment structure will completely enclose the reactor coolant system, the safety injection systems tanks, the containment cooling system's fan coolers and the circulation fans. The containment spray system is designed to reduce rapidly the containment pressure and temperature and to supply chemically treated water to control fission product inventory following a loss-of-coolant accident. The containment combustible gas control system which consists of redundant hydrogen recombiners located outside of the containment and a backup purge system, is designed to maintain the hydrogen concentration below the flammability limit of 4.0 volume percent following a loss-of-coolant accident. (SER § 6.2.4) The isolation system, consisting of the circuitry and isolation valves, provides appropriate containment isolation following a loss-of-coolant accident. These are the principal means by which plant personnel and the public will be protected from excessive exposure to radioactive materials should a major accident occur in the facility. (PSAR § 1.2.6)

27. The Unit 2 emergency core cooling system (ECCS) is designed to provide cooling for those postulated accident conditions where a failure in the reactor coolant system piping results in a loss of coolant greater than

the makeup capacity of normal operation equipment. It will also be designed to protect against the consequences of a main steam line break. (SER § 6.3.1) The ECCS will consist of four safety injection tanks, a high pressure safety injection system and a low pressure safety injection system, with provisions for recirculation of the borated water after injection. (SER § 6.3.2) The system will be designed so that various combinations of the system will assure core cooling for the complete spectrum of postulated break sizes. (*Id.*)

28. The Atomic Energy Commission (now NRC) on January 4, 1974 issued acceptance criteria for emergency core cooling systems for light water reactors. (39 FR 1004.) The criteria as set forth in 10 CFR §§ 50.34(a)(4), 50.46(a)(1) and Appendix K to 10 CFR Part 50 require evaluation of core cooling in accordance with certain criteria using an acceptable evaluation model. (SER § 6.3.3) The Staff reviewed the information submitted by the Applicants and Combustion Engineering and concluded that the design of the Unit 2 emergency core cooling system is acceptable. (SER § 6.3.3; SER Supplement 1, § 6.3.4; SER Supplement 2, § 6.3.3; SER Supplement 3, § 6.3.3)[16]

29. The proposed design of the protection and control systems for the facility is in several respects similar to that of Calvert Cliffs Units 1 and 2 which was previously reviewed and approved. The design will, however, include core protection calculators (digital mini-computers) which will be utilized to generate a reactor trip signal at a low value of the departure from nucleate boiling ratio (low DNBR) or high local power density. (SER §7.1) The reactor protection system will be comprised of four redundant and independent protection channels per trip input. Each channel trip input will deenergize three relays when a trip setpoint is exceeded. The contacts from these relays will be arranged into six independent logic matrices representing all possible two-out-of-four trip combinations for the four protection channels. A trip output from any one of the six logic matrices will interrupt power to the control rod supply breakers and will cause insertion by gravity of all full length control rods and thereby shut down the reactor. The reactor protection and control system will be designed in conformance to the Commission's General Design Criteria and IEEE Standard 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations."[17] (SER § 7.1, 7.2; PSAR § 1.2.7.1)

[16]By letter dated April 2, 1979, the Staff advised the Board that it is evaluating new information related to Combustion Engineering's flow blockage model for the Unit 2 Emergency Core Cooling System. The Staff evaluation is ongoing and the Board opines that this matter can abide the operating license review.

[17]This Standard was reaffirmed in 1978.

30. The facility's safety-related instrumentation and controls of the engineered safety features will include (1) the engineered safety feature protective systems which will consist of the electrical and mechanical devices and logic circuitry involved in generating signals that actuate the required engineered safety feature systems, and (2) the arrangement of components that will perform protective actions after receiving a signal from either the engineered safety feature protective system or the operator. All of the engineered safety feature protective systems will be identical except for the input parameters and include four redundant and independent channels per trip input. (SER § 7.3)

31. Unit 2 will be connected to the New England power grid through two 345 kV and one 115 kV transmission lines. These lines and associated circuits will constitute the two physically independent circuits required by Criterion 17 of the General Design Criteria. (10 CFR Part 50, Appendix A) To maintain independence between the 345 kV and 115 kV circuits, the 115 kV line will be run underground to the facility switchyard from a substation located at Manomet, Mass. (SER § 8.2; PSAR, §1.2.8) The 345 kV ring bus which currently serves Pilgrim Unit 1 will be modified to accommodate Unit 2. During normal power generation, the auxiliary and safety related a-c power distribution systems will be supplied by the unit a-c power supply via the generator load switch and three unit auxiliary transformers. In the event of turbine or reactor trip the generator load switch will be automatically opened. The 345 kV preferred a-c power supply will remain connected and will provide uninterrupted power to the auxiliary and safety-related a-c power distribution systems via the main and unit auxiliary transformers. In the event that the preferred 345 kV power is lost, 115 kV power will be supplied to the auxiliary and safety related bus bars by automatic transfer to the reserve transformers. (SER § 8.2)

32. Onsite standby a-c power will be supplied for the facility by two diesel generators. Each diesel generator will supply one of two redundant 4160 V emergency bus bars arranged in a two-division split-bus configuration. Among the design features to be included in the standby diesel generators and their associated a-c power distribution systems are: (a) electrical independence from each other; (b) starting and operation of either diesel will not be conditioned by operation of the other; (c) each diesel will be started by an undervoltage signal from its respective bus bar or by an engineered safety feature actuation signal; (d) separate onsite fuel storage for each diesel sufficient for seven days operation at accident load and (e) each diesel generator and its auxiliary systems will be housed in a separate seismic Category I installation. (SER § 8.3)

33. The d-c power system for the facility will consist of four redundant and independent d-c load groups, each composed of a 125 V battery, a

battery charger, distribution bus, distribution panel, interconnecting cables and connected loads. The system is in conformance with General Design Criteria 17 and 18, Regulatory Guide 1.6 and appropriate IEEE standards. (SER § 8.3; PSAR §1.2.8)

34. The ultimate heat sink for Unit 2 is Cape Cod Bay. Sufficient heat removal capacity will be provided for an indefinite time in conformance with Regulatory Guide 1.27. (SER § 9.2.4)

35. Plant cooling requirements during power operation and shutdown of the facility will be met by the reactor coolant system, the shutdown cooling system and by four segregated water systems consisting of (a) the turbine building cooling water, (b) the component cooling water, (c) the auxiliary building cooling water, and (d) the service water. The last three systems are required for safe shutdown of the plant following a design basis accident. These systems will be designed for 100 percent redundancy with functional and physical separation of each train of redundant components. The systems are interconnected so that functional and physical separation of each train of redundant components will be maintained. (PSAR, § 1.2.9.2) The component cooling system is designed to circulate water through two physically separated seismic Category I closed loops. Each loop will remove heat from the containment, shutdown heat exchangers, spent fuel pool heat exchangers, engineered safety features equipment, boric acid concentrator package and the waste concentrator package. Only one train of these components is required for safe plant shutdown following any postulated accident. (SER 9.2.2) The auxiliary building cooling water will circulate through two physically separated loops each independently capable of providing the required cooling for the components of the engineered safety features support system. (SER § 9.2.3)

36. The facility's station service system, which will meet Criterion 44 of General Design Criteria, 10 CFR Part 50, Appendix A, will supply water to two identical trains of safety related equipment. Each train will be capable of providing sufficient water for the component, auxiliary building, and diesel generator cooling water systems. The station service water system will be designed so that a single failure of its components or of the onsite power supply will not prevent a safe shutdown. (SER § 9.2.1)

37. The facility's fire protection system and its components will be designed so that a failure or inadvertent operation of the fire protection systems will not result in loss of function of safety related equipment. Sprinklers will be provided in the engineered safety feature pump rooms, the standby diesel generator rooms and the turbine building. Fixed automatic chemical extinguishing systems will be provided for the cable spreading rooms, computer room and unoccupied areas housing electrical equipment. The facility's proposed fire protection system, as currently

designed, meets Criterion 3 of the General Design Criteria. (10 CFR Part 50, Appendix A) (SER Supplement 3, § 9.5.1)

38. The unit 2 steam and power conversion system will be of conventional design and similar to those of previously approved plants. The heat of the reactor coolant will be removed through two steam generators and converted to electrical energy through the turbine driven generator. The condenser will transfer unusable heat in the cycle to the condenser cooling water. (SER §10.1)

39. The radioactive waste (radwaste) system will consist of solid, liquid and gaseous waste systems. The design objective of each system is to restrict the amount of radioactive material released to the environment to as low as reasonably achievable in conformance with the requirements of 10 CFR § 50.34a and 10 CFR Part 50, Appendix I.[18]

40. The facility's liquid waste system will process input from decontamination, chemical regenerants, steam generator blowdown, equipment and floor drains. The gaseous waste systems for the facility will provide holdup capacity to allow decay of short-lived noble gases stripped from the primary coolant. Charcoal adsorbers will be used to remove radioiodine from the main condenser offgas and from the air purged from the containment building. The solid waste system will provide for the packaging and solidification of low level radioactive wastes generated during station operation. These will be shipped to a licensed disposal facility. (SER § 11.0)

41. The offsite radiological consequences of design basis accidents have been evaluated by the Staff and found to be within the guidelines of 10 CFR Part 100. (SER § 15.0; SER Supplement 3, §§ 15.5, 15.6)

42. The research and development necessary for the safe operation of Unit 2 have been identified by the Applicants and will be performed on a timely schedule. (SER § 1.7)

43. Health and safety issues raised in these proceedings by intervenors are addressed in Part V of this decision. The Staff testified that there are no additional health and safety matters that cannot be favorably resolved prior to completion of construction. (SER § 21.1; SER Supplement 4, § 21.1)

II. C. Technical Qualifications of Applicants

44. Testimony on the technical qualifications of the Applicants and of their principal contractors was prepared and presented by a series of panels. The Staff's testimony was similarily presented. No direct testimony was

[18]Findings of Fact by the Board pertaining to Appendix I are contained in Part II.G.c. *infra*.

offered by the intervenors who argued their cases through cross examination of Applicants' and Staff's witnesses.[19]

45. As described in Paragraph 69 of this Decision the Applicants in this action are a consortium of a number of public utilities and municipalities. The members of this consortium constitute the ownership of the proposed generating station. The lead entity of this group is the Boston Edison Company. In that position, BECo represents and is contractually empowered to act for the other owners on matters of design, procurement, licensing, construction, operation and maintenance of Unit 2.

46. In a similar line of authority, the principal contractors, CE and Bechtel, are contractually responsible to BECo not only for the supply for the steam generating system and construction services, but also for assurance that their product will meet designated specifications and quality in accord with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Facilities." Additionally these contractors are to exercise prudent use of capital funds. To these ends BECo has the ultimate authority to reject completed work and to terminate further work through the use of stop-work orders. (Applicants' Witness Howard at 13 following Tr. 3735)

47. Under 10 CFR 50.40(b) the Staff is obligated to determine that an applicant is technically qualified to engage in the proposed activity in accordance with regulations, yet "The Staff has no specific quantitative guidelines for determining whether an applicant has the management capability to undertake the construction or operation of a nuclear reactor... A determination on this subject must be subjective and judgemental and

[19]The testimony on the technical qualifications of BECo was given by Panelists J.E. Howard, Vice-President Nuclear, R. M. Butler, Manager Nuclear Projects, W. M. Sides, Manager Quality Assurance and V. P. McMahon, Corporate Manager Quality Assurance, Kaiser Engineers, Inc. (Following Tr. 3735)

The Panel on the qualifications of the Bechtel Power Corporation, the architect-engineer, was comprised of F. A. Hollenbach, Manager Project Operations, T. D. Dow, Supervisor QA Program, M. J. Jacobson, QA Engineer, J. D. Blatchford, Project Engineer, G. K. Stavro, Inspection Manager and D. R. Johnson, Field Engineer, Quality Control. (Following Tr. 3987)

The Panelists for Combustion Engineering, Inc., the nuclear steam supplier, were C. R. Waterman, Unit 2 Project Manager, C. W. Hoffman, Director Quality Assurance, W. E. Midinger, Manager, QA Systems and W. K. Couch, Manager Quality Control. (Following Tr. 4185)

The Staff Witnesses were D. L. Caphton, D. M. Sternberg, and R. F. Heishman all of the U.S. Nuclear Regulatory Commission's Office of Inspection and Enforcement. (Following Tr. 4234) Further evidence by the Staff on this contention was prepared by D. B. Vassallo and M. B. Aycock of the Office of Nuclear Reactor Regulation. (Following Tr. 5534) Still further is the testimony of A. M. Garland, of the NRC Quality Assurance Branch, Division of Reactor Licensing (following Tr. 4425), and by R. H. Vollmer, Office of Nuclear Regulation. (Following Tr. 4464) each utility must be evaluated individually. The best test is a functional one."[20] The Pilgrim Unit 1 station has been operated by BECo since 1972. The managerial and operational experience with that was, accordingly, taken as a measure of expectations of Unit 2.

48. This issue was almost exclusively addressed in the testimony through discussions of assurances that the products and services of the suppliers would be of the requisite quality.

49. The corporate structure of BECo includes a Vice-President, Nuclear, reporting to the Office of the President and receiving reports from managers of various functions within the project. One of these is the Quality Assurance (QA) Department Manager to whom reports QA Engineering. The QA Department is responsible for establishing a QA program applicable to all safety related activities performed by BECo and its principal contractors in accord with the established QA program. [At 14 and 47 (BECo Exhibit BE-TQ-1) following Tr. 3735; see also Fig. 1, unnumbered p. 19 following Tr. 5534] All reporting along this chain is independent of other project activities including operations, construction, cost control and engineering.

50. The managers of the QA and the Nuclear Projects Departments (the latter administers engineering and construction) and the Vice-President to whom they report each have more than 20 years experience in nuclear energy. These experiences include responsibilities at commercial power generating stations at production reactors and with naval-propulsion units. (at 2 through 4 following Tr. 3735)

51. Applicants' Witness Sides testified to the stability of the QA Engineering Staff at BECo asserting to the absence of turnover and of any need for disciplinary action. As of February 1976, BECo Staff had accumulated considerable experience in QA matters at Pilgrim Unit 1. (Tr. 3904)

52. The roster of the BECo QA Department is comprised of eight managerial and professional personnel with an enlargement authorized as necessary to the activities required during the construction of Unit 2. Each position requires an academic degree in engineering augmented by up to five years experience in QA, or related activities, in the nuclear industry, the exact amount depending upon the specific position. Within the Department there must be knowledge of applicable Federal Regulations (10 CFR 50), ASME Codes, American National Standards and NRC Regulatory Guides as well as familiarity with internal programs and activities related to QA. (Applicants' Witness Sides at 38 and 39 following Tr. 3735)

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[20] In the Matter of Carolina Power and Light Company (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4) LBP-79-19; 10 NRC 37 at p. 41 (1979).

53. In addition to BECo's line organization for QA is the Quality Assurance Review Committee, a staff group, with membership composed of the Vice-President - Nuclear as chairman and four department managers. This Committee provides a continuing review of BECo's QA Program to assess its scope, implementation and effectiveness. There is also the Nuclear Safety Review and Audit Committee, chaired by the QA Manager, which has the responsibility of reviewing the nuclear safety of Unit 1 operating in conformance with NRC-issued technical specifications. (Applicants' Witness Howard at 9 and 10 following Tr. 3735)

54. The organizational relation between BECo and its principal contractors is shown in Applicants' Exhibit BE-TQ-2. (At unnumbered p.48 following Tr. 3735) By this arrangement the contractors' QA organizations report to the BECo QA manager. Similarly the contractors' project managers report to the BECo Nuclear Project Manager thence, in both instances, to the BECo Vice-President - Nuclear thereby effecting authority and control through an interface established by procedures. Audits and surveillance of fabrication and construction activities for safety-related structures, systems, and components are performed by the QA Department. Necessary corrective actions are taken by the contractors through, in the extreme, stop-work orders. (Applicants' Witness Butler at 25 following Tr. 3735; see also Staff Witnesses Vassallo and Aycock at 6 through 8 following Tr. 5534)

55. The Staff concludes from its investigation of the Applicants and their principal contractors that BECo is technically qualified to carry out the responsibilities attendant to the design and construction of Unit 2. In support of its finding, the Staff cites its observation of a favorable attitude of the management of BECo toward safety and environmental characteristics of Unit 2. Further it cites the practice of BECo in seeking advice and guidance from outside experts[21] on those specialties beyond the ken of its staff. (Staff Witnesses Vassallo and Aycock at 14 through 17 following Tr. 5534)

56. The principal contractors of BECo, Bechtel (architect-engineer, construction service) and CE (nuclear steam system supplier) are large well established organizations with long industrial experience. Each has been engaged, in its respective field, in the nuclear industry for upwards of two decades. Additionally, each has great experience in more conventional energy conversion systems, that of CE dating back almost a century. (Applicants' Witness Hollenbach at 9 through 30 following Tr. 3987;

[21]For example, BECo has obtained an independent assessment of its QA program from Kaiser Engineers, Inc., a qualified QA consultant. (Applicants' Witness Howard at 41, following Tr. 3735)

Applicant Witness Waterman at 10 through 31 following Tr. 4185, see also testimony of Staff Witness Vollmer following Tr. 4464)

57. Representatives of each of these contractors presented detailed descriptions of its organization including quality assurance activities and responsibilities. Each instance can be typified by the BECo pattern, *supra*, though differing in details. Worthy of note is the existence of a line of reporting and responsibility of the QA staff to upper-level management entirely independent of segments controlling operations, construction, procurement, etc. [See BECo Exhibit BPC-TQ-1, unnumbered page concluding testimony following Tr. 3987; also Applicants' Witness Hollenbach at 25 following Tr. 3987 and BECo Exhibit CE-TQ-5 (at unnumbered page 35 following Tr. 4185)]

58. Some measure of the qualifications of the Applicants in these proceedings is to be expected from BECo's experience at Unit 1, a boiling water reactor which began operation in 1972. During the hearing a number of reports derived from inspections of Unit 1 by AEC Division of Compliance (now NRC Inspection and Enforcement) were reviewed.[22]

59. Intervenor Commonwealth introduced several of these reports for the purpose of illustrating BECo's poor performance at Unit 1 and, consequently, an absence of technical competence to construct and operate Unit 2. Several of these reports concerned procedural matters, interpretation of the results of weld testing, and some design changes.[23]

60. One consequence of the findings of AEC inspectors was the assessment of three $4000 civil penalties against BECo. One item concerned the qualification rating of an inspector (employed by a secondary contractor of BECo of ultrasonic examination of certain welds in the primary-coolant coolant piping. Another concerned the calibration of the ultrasonic testing equipment; still another had to do with the presence, as allegedly required, of BECo QA personnel as a witness to the ultrasonic testing. The $12,000 fine was paid. (Applicants' Witness Howard at Tr. 3850-3889)

61. There is uncertainty in the bases for the allegations which involve interpretation of Section XI of the American Society of Mechanical Engineers Code on inservice inspection (which references American Society for Nondestructive Testing (ASNT) Recommended Practice SNT-TC-1A, and some apparent conflict in personnel records. At any rate, although there were plans to repeat the inspections such was not required. (Applicants' Witness Howard at Tr. 3874, 4014 and 4016) [Some were,

[22]Inspections are listed in Appendix A of testimony of Staff Witness Caphton following Tr. 4234.

[23]Commonwealth Exhibits 3 through 8 received at Tr. 3847; Exhibits 9 and 10, at Tr. 3860, *et seq.*

however, repeated. (BECo letter to NRC dated June 18, 1975 appended to Commonwealth Exhibit 11, received at Tr. 3949)]

62. As a consequence of these events BECo increased the frequency of audits, initiated better procedures, clarified its appropriate inspection manual, and advised its contractor to institute more thorough personnel training. (Applicants' Witness Howard at Tr. 3864, 3875; BECo letter to NRC dated June 18, 1975 appended to Commonwealth Exhibit 11)

63. Applicants' Witness Howard testified that the matter of weld inspection had been resolved with the AEC/NRC. (Tr. 3855) Current practice on the specific item of inspection qualification has been accepted by NRC. (Tr. 4003)

64. Summary Staff testimony on the technical qualifications of BECo and its principal contractor was prepared by Messrs. Vassallo and Aycock. (Following Tr. 5534) These and other witnesses recognize the inevitable appearance of deviations from specifications and procedures in past similar operations of the Applicants. They occurred in varying degree. (See testimony of Staff Witnesses Caphton and Sternberg following Tr. 4234.) Of great importance, however, in the evaluation of the qualifications of the Applicants in future actions is the severity of those infractions in the response and remedial actions of the licensee, and in the reception of them by the regulatory agency. Additional factors for investigation are the Applicants' organizational structure and manpower and the technical qualifications of their principal contractors.

65. On the basis of these several considerations, the Staff concluded, in its overall evaluation, that BECo is technically qualified to enter into the construction of Unit 2 with the support of the principal contractors it has named. (Staff Witnesses Vassallo and Aycock at Tr. 5630-5647)

II. D. Financial Qualifications of Applicants

66. Initial evidential presentations on the financial qualifications of the Applicants were made in February 1976 by the Applicants and the Staff supporting and supplementing their positions contained in the Pilgrim Station License Application, Section VI and in Supplement 1 of the SER, respectively. (Applicants' Witnesses Houston following Tr. 5078, and Kelmon, Mefferman and Mitiguy following Tr. 5103)

67. Subsequent changes in the proposed ownership of Unit 2 and in revised plant costs necessitated additional information from the Applicants and review by the Staff[24] and further hearing before this Board. The Applicants' supplemental testimony was through BECo Treasurer Kelmon

[24]Staff Exhibit 21, SER Supplement 3, p. 1-2 following Tr. 8921; Staff Exhibit 50, SER Supplement 4, p. 20-1 following Tr. 10046.

and Assistant Treasurer May. (following Tr. 9234) The Staff presented its further evaluation of the financial qualifications of the Applicants through Witness Karlowicz. (following Tr. 9513) Intervenor Commonwealth's evidence was presented by Witness Levy. (following Tr. 9434)

68. The Commission requires, in 10 CFR § 20.33(f) and 10 CFR 50 Appendix C, that an applicant show it either possesses or has reasonable assurance it can obtain the funds necessary to cover estimate construction and related fuel costs. Guidance was provided by the Commission:

> "...given the history of the present rule and the relatively modest implementing requirements in Appendix C (footnote omitted), a 'reasonable assurance' does not mean a demonstration of near certainty that an applicant will never be pressed for funds in the course of construction. It does mean the applicant must have a reasonable plan in light of relevant circumstances."[25]

69. The ownership of Unit 2 is presently distributed among investor-owned and non-investor-owned utilities in this proportion:[26]

a)	Boston Edison Company	59.026%
b)	The Electric Light Department of City of Burlington	0.330
c)	Central Maine Power Company	2.850
d)	Central Vermont Public Service Corporation	1.780
e)	Fitchburg Gas and Electric Light Company	0.190
f)	Town of Hudson Light and Power Department	0.174
g)	Massachusetts Municipal Wholesale Electric Company	13.240
h)	Montaup Electric Company	2.150
i)	New Bedford Gas and Edison Light Company	1.530
j)	New England Power Company	11.160
k)	Public Service Company of New Hampshire	3.470
l)	The United Illuminating Company	3.300

[25]Public Service Company of New Hampshire (Seabrook Station Units 1 and 2) 7 NRC 1 at p. 18 (1978).
[26]Staff Exhibit 50, SER Supplement 4, Appendix C at C-1, following Tr. 10046.

m)	Tauton Municipal Lighting Plant Commission	0.600
n)	Vermont Electric Cooperative, Inc.	0.200
		100.000%

70. "The cost of Pilgrim Unit 2, including site and 'common facilities' (i.e., common to Pilgrim Unit 1), the initial nuclear fuel core, and transmission and switching facilities is estimated to be $1,319 million. With the inclusion of allowance for funds used during construction (AFUDC)," the projected total cost of the facility is $2,037.5 million.[27]

71. Consistent with Commission requirements the investor-owned applicants filed statements of sources and uses of funds and non-investor-owned applicants filed alternative financial data.[28] Applicants expect to rely upon a combination of internally generated funds (39 percent of the requirement) and the sale of debt and equity (61 percent) to finance the construction and initial fueling of the facility.[29]

72. Whereas the commitment of BECo to the overall financial schedule is significant, it is not extraordinarily great relative to that utility's recent experience in construction expenditures.[30]

73. The Staff found to be reasonable BECo's projections of the rate of return on equity, internal cash generation, interest coverage and capital structure.[31]

74. Applicants other than BECo submitted plans for financing their portion of Unit 2 consisting, primarily, of issuance of general obligation and revenue bonds, with interest and principal to be paid from revenues. The Staff concludes that the members of this group of co-applicants have developed reasonable financing plans, recognizing them not to be necessarily what will actually occur. This demonstration of one possible way of financing the construction suffices Commission requirements.[32]

75. The portion of Supplement 4 of the Staff Safety Evaluation Report (Exhibit 50), cited above, addressing the financial qualifications of BECo was supported by Staff Witness Karlowicz (Tr. 9514) who had used state-of-the-art techniques of financial analysis, accepted by the financial

[27]License Application, Amendment 9, Applicants' Exhibit 1-00 at V-1, Tr. 9601. Staff Exhibit 50, SER Supplement 4, Appendix C at C-1 following Tr. 10046.
[28]License Application, Amendment 8, Applicants' Exhibits 1-NN (1), (2) and (3), Tr. 9601; Staff Exhibit 50, SER Supplement 4, Appendix C at C-2 following Tr. 10046.
[29]Applicants' Witnesses Kelmon and May at 6 and 7 following Tr. 9234.
[30]Applicants' Witnesses Kelmon and May at 7 and 8 following Tr. 9234.
[31]Staff Exhibit 50, SER Supplement 4, at C-8 through C-14 following Tr. 10046.
[32]*Id.* at C-15 through C-48.

community, in review of information supplied by BECo with supplements from various investment rating agencies. (Tr. 9519) The testimony of this witness is the Staff's evaluation of BECo's financial qualifications.[33] The conclusion of the Staff review affirms the ability of BECo to assume the financial obligation of a 59 percent ownership of Unit 2.

76. Testimony on behalf of the Commonwealth on the matter of financial qualifications of the Applicants was presented to the Board and the Record by Paul F. Levy.[34] This witness served in the Massachusetts Energy Policy Office from mid-1974 through 1977. During calendar year 1978 he was a Commissioner of the Commonwealth's Department of Public Utilities. On the basis of these limited experiences and of other statements in the record (Tr. 9414 through 9434) the Board finds the qualification of Mr. Levy to present evidence on the subject matter to be marginal and, hence, accords appropriate weight to his testimony.

77. The testimony of Witness Levy was largely based on two internal BECo memoranda, prepared in mid-1978, and on the testimony presented by Mr. Kelmon at a pending rate case before the (Massachusetts) Department of Public Utilities (DPU-19991).[35]

78. For reasons appearing in his testimony,[36] Witness Levy stated that BECo would encounter increasing difficulty in issuing debt and equity securities within the construction schedule of Unit 2. One reason was that further stock issuances, if necessary because of the possible high percent of allowance for funds used during construction (AFUDC), will dilute the book value of current stock, thereby reducing the interest of potential investors. Upon cross examination the witness could cite no instance of recent sale of electric utility stock above book value and, in fact, he considered sale of stock below book value not to be unusual. Nonetheless, electric utilities have been successful in marketing stocks. (Tr. 9470-71)

79. On November 1, 1979 the Applicants filed with the Board and all parties in this proceeding a Base and Standby Revolving Credit and Term Loan Agreement, dated July 31, 1979 with an Amendment of October 12, 1979. This Agreement, as amended, was approved by the Department of Public Utilities and the sale of the subject securities was authorized by Order 20145 dated September 17, 1979, as amended by Order 20145-A of October 17, 1979. The Agreement makes available to BECo a principal amount of $500 million in aggregate to be used for the Company's general corporate purposes including capital expenditures.

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[33] *Id.* at 20-1 and at C-1 through C-15.
[34] Following Tr. 9434. Mr. Levy had testified before this Board in February 1976 on the issue of cost comparison of various future baseload generating stations at Tr. 4990.
[35] Commonwealth Exhibits 100, 101 and 102 at Tr. 9270, 9275 and 9276 respectively.
[36] At 6 and 7 following Tr. 9434.

II. E. Common Defense and Security

80. The activities proposed to be conducted under the construction permit will be within the jurisdiction of the United States and all directors and principal officers of the Applicants are citizens of the United States. The Applicants are not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. Although the activities to be conducted do not depend upon any restricted data, the Applicants have agreed to safeguard any such data in accordance with the requirements of 10 CFR Part 50. The Applicants will obtain fuel as needed from sources available for civilian purposes, so that no diversion of special nuclear material from military sources will occur. (SER §19.0)

81. Pursuant to Commission regulations and earlier rulings,[37] consideration of potential sabotage of Unit 2, by armed acts of force both by enemies of the U.S. Government and by other armed personnel regardless of origin and basic intent, were ruled inadmissible into these hearings on the application for a construction permit. Accordingly, deliberations on this contention were limited by the Board to theft and sabotage of radioactive materials during transport to and from the Pilgrim site and to actions within the proposed plant by a few employees or by a small group of outsiders following surreptitious entry. The potential of an armed band, "terrorists," has not been included.

82. Intervenor Commonwealth, the Applicants, and the Staff presented witnesses who testified and were cross-examined on this contention.

83. The testimony of Commonwealth Witness Rathjens (following Tr. 4380) was based on the findings of a Commission charged by the Commonwealth to study "...its role in assuring the safety of nuclear power plants...." chaired by this witness. Although the testimony as filed encompasses security in a nuclear power station as well as in transport, consideration was limited, without prejudice, as stated above.

84. Whereas he testified that the self-damage to a reactor plant resulting from actions by an employee could conceivably be severe, the likelihood of such occurrences is low because of the protection afforded by monitoring devices and the dispersal of the information they derive among the operations staff. To be otherwise would require collusion among a number of employees with similar motivation, a condition controllable by effective

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[37] 10 CFR 50.13. See also, for example, *Siegel v. AEC*, 400 F.2d 778 (D.C. Cir. 1968); *Florida Power and Light Co.* (Turkey Point Units 3 and 4) 4 AEC 218 (1969); *Long Island Lighting Co.* (Shoreham Station) 6 AEC 831 (1973); *Consolidated Edison Co. of New York, Inc.* (Indian Point Station, Unit 2), ALAB-202, 7 AEC 826 (1974); *Potomac Electric Power Co.* (Douglas Point Station, Units 1 and 2), ALAB-218, 8 AEC 79 (1974). Although the Board senses these cited rulings to be at variance with 10 CFR 73.55(a)(1) it has followed the mandates of these decisions pending clarification of an apparent conflict within the Commission's regulations.

screening of personnel. To bring about such an event of sufficient magnitude to grossly affect the public through a large release of radioactivity is even less likely. (at 123 to 126 following Tr. 4380)

85. Staff Witness Sears also testified that the likelihood of theft and industrial sabotage within a nuclear power plant by unarmed persons is very low. His judgment was based on requirements for employee selection, for surveillance and search of employees and visitors upon entry, for fulfillment of physical protection objectives[38] which require intrusion monitoring and alarm systems and location of vital equipment[39] within an area encompassed by three barriers representing, progressively, increasing levels of security control, for redundancy in and separation of vital equipment thereby protecting against malicious outages, and for special equipment for diversion of special nuclear materials.[40]

86. This witness also addressed the cost of anti-sabotage and anti-theft measures at an electric generating station, including downtime and necessary repairs and replacements as a consequence of sabotage, and concluded that such costs were independent of the type of fuel. (Tr. 2226) Further, the cost estimates, $250,000 capital and $200,000/yr operating, are similar for a non-nuclear installation and are insufficient to shift the cost-benefit analysis away from nuclear. He concluded that the risk of theft of special nuclear material from a nuclear power plant is small because of the hazard to the potential thief from the associated highly radioactive substances. Supporting this conclusion is the absence of successful and identified theft and sabotage attempts at any domestic operating nuclear power reactor. (following Tr. 2210)

87. In the course of his testimony on the risks associated with the transport of radioactive materials, Commonwealth witness Rathjens pointed out that, although used fuel in transit may be a more attractive target than a reactor for those having extortion or coercion as their goal, the potential for damage to the public is much less. This potential could arise from materialization of a threat to disperse radioactivity were demands not met. Potential concommitant panic due to the public aversion to radiation exposure is also a consequence. To effect a probable lethal exposure from sabotage, however, would require breaching the container with explosives, the availability of heavy equipment and remote manipulators, and subjecting the contents to temperatures of the order of a thousand degrees

[38]These objectives are described in part in the Commission's Regulatory Guide 1.70.15 and in 10 CFR 73.55.

[39]Vital equipment is defined as any whose failure could directly or indirectly endanger public health and safety by exposure to radiation.

[40]Irradiated fuel is the sole material within a nuclear power plant designated as special nuclear material.

to vaporize them preparatory to atmospheric dispersal. (at 128 following Tr. 4380; Tr. 4393, 4419)

88. Diversion of used fuel for fabrication of illicit nuclear weapons is not a viable threat. Witness Rathjens reiterated the necessity of first coping with the activity of the accompanying fission products and of chemically purifying the special nuclear material to arrive finally at an inferior weapon. He concluded that more attractive and practical channels existed for illegally acquiring a nuclear weapon. (Tr. 4410, *et seq.*)

89. He, however, strongly recommended accompanying these shipments to and from the Pilgrim site with armed escorts retaining constant communication with law-enforcement authorities. (at 129 following Tr. 4380)

90. Applicants' Witnesses Rodger and Low testified that the risk of theft and sabotage of low-level wastes is non-existent because of its negative financial value, the absence of a health hazard even if dispersed, and the deterring legal penalties. These witnesses concluded that the low monetary value of used nuclear fuel, even after costly processes for separation of fission products, was not an incentive for its theft. The radiological consequences of such theft would not be significantly different from those resulting from a transportation accident. (at 8, 13 following Tr. 2024)

91. The testimony of Staff Witness Barker of theft and sabotage in transport (following Tr. 2275)[41] is summarized in the record as:

> "Based on consideration of the low enrichment of the fuel, the negative value of the waste, criminal laws against theft and sabotage, the type and form of the material, and the size, weight, and designs of the rugged packaging required by the regulatory standards, the probability of such acts causing a release is so small as to not require additional analysis or protection. Although the probability cannot be easily quantified, our lack of such experience indicates it is much less than the probability for very severe accidents.
>
> "Furthermore, in the unlikely event such an accident could cause or did cause a release, principally because of the nature and form of the fuel and waste, the consequences would not be significantly different from those assessed in WASH-1238." (Tr. 2280-81)

92. These conclusions are supported by the witness' observation of the rugged construction of the massive shipping containers, the properties of the materials which make handling and purification difficult and expensive,

[41]Cross-examination of Mr. Barker is continued beginning at Tr. 2458.

and the proven ability to detect relatively low-intensity radiation by, for example, aerial surveillance, allowing discovery of illegally diverted radioactive materials including used fuel. In his opinion, a saboteur intent upon procurement of explosives or otherwise harming the public can find more readily available, more efficient and economic, and more positively acting materials in commerce. Examples are ordinary explosives, compressed or liquified gases such as chlorine and propane, and biological and etiological agents. (at 9 following Tr. 2275)

93. Staff Witnesses Kasun and Hodge (following Tr. 8459) and Hodge with Sawyer (Staff Exhibit 68 served January 17, 1980 admitted by Board Order December 30, 1980) supplemented the testimony of Barker by sponsoring a Commission study of the radioactive materials.[42] This recent analysis by the Staff of the consequences of a transport accident to or sabotage of a 10-fuel-element shipping cask assumes the maximum credible breach resulting in the release of 100 percent of the gases and 1 percent of the volatile and non-volatile solids[43] as respirable aerosols. The number of health effects in an area with a population density of 100 persons per square mile and average meteorological conditions are calculated to be less than one early cancer death and approximately 38 latent cancer fatalities. (at 4 following Tr. 8459)

II. F. Generic Issues

94. In the ongoing evolution of nuclear fueled power generating stations the Staff maintains surveillance of advances in technology, of their potential for increasing the safety of operating those stations, and of concerns and safety issues as they may develop from operating experiences. Accordingly, the Staff maintains a list of items which would be potentially benefitted by additional information and investigation. Generally these items concern a type or class of stations rather than a single installation and, hence, are designated as "generic safety issues."

95. The importance of each issue to safety establishes a priority to the effort for its resolution. The issues listed, therefore, change from time-to-time as research, experience, identification, etc. occur.

96. It is apparent that solutions to these generic items become important at times more near to the operation of Unit 2 than at this review of a construction permit application. At this time the Board has a responsibility

[42]"Calculation of Radiological Consequences from Sabotage of Shipping Casks for Spent Fuel and High-Level Wastes," NUREG-0194, Feb. 1977. This information was not available at the time of Mr. Barker's testimony.

[43]The more intense emission postulated by Commonwealth Witness Rathjens required subjecting the fuel elements to very high temperatures. (Tr. 4398 *et seq.*) No analysis of ensuing health effects was offered nor was the credibility of such an event established.

to judge the likelihood of a predictive satisfactory timely solution. An Appeal Board has given some guidance.[44]

97. In this record the Staff has cited some 133 generic issues of which 28 are judged to be related to plant safety and applicable to Unit 2. Further the Staff has described[45] each of these issues, summarized the present status of its solution, projected future investigative programs, and evaluated the impact of the issue on the operation of Unit 2.

98. The Staff concluded that none of the 28 issues applicable to Unit 2 is cause for denying the construction permit.

II. G. Additional Health and Safety Issues
a. Steam Generator Tube Integrity

99. On February 20, 1976, the Board issued an Order dismissing intervenor Daniel F. Ford from the proceeding for failure to carry out the responsibilities of being an intervenor in this proceeding. The Board reviewed Mr. Ford's contentions and dismissed all of them. Ford Contention K, however, alleged that the proposed steam generator tubes will not maintain their integrity during a loss of coolant accident. The Board adopted this contention in modified form as its own and directed that evidence regarding the overall integrity of the proposed steam generator tubes be presented. In response to that direction, testimony was presented by Staff and Applicants on May 24 and 25, 1976. In subsequent experiences at certain CE PWR generating stations, damage to tubes in steam generators was observed as deformations in the vicinity of supporting tube plates. This circumstance, called "denting," was addressed by Staff and Applicants in a reopened hearing on March 6 and 7, 1978.

100. The integrity of the steam generator tubes is highly significant from a radiological safety standpoint since they represent an integral part of a major barrier between the radioactive reactor coolant fluid, which circulates inside the tubes at high temperature and pressure, and the secondary two-phased coolant. Rupture of steam generator tubing would result in a release of the radioactive primary fluid into the secondary coolant. Any subsequent releases from the secondary system would result in discharge of radioactivity to the outside environment. The weakening of these tubes due to service induced tube degradation processes could in the event of a loss of coolant accident (LOCA), result in rupture of tubes and release of the fluid energy from the secondary system into the containment or into the reactor vessel. This in turn could interfere with the emergency core cooling water reflooding rate with major radiological safety implications. (Staff Witness

[44]*Gulf State Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 775 (1977).
[45]Staff Exhibit 50, SER Supplement 4 at D-16 through D-31, following Tr. 10046.

Rajan at 2 following Tr. 5847) The steam generator tubing being an integral part of the reactor coolant pressure boundary is designed to meet Criteria 14, 15, 31 and 32 of Appendix A to 10 CFR Part 50.

101. In order to meet these criteria the Staff requires that the steam generator tubes be designed with sufficiently thick walls so that:

> "1. tubes with detected acceptable defects will not be stressed during the full range of the normal reactor operation beyond the elastic range of the tube material;
> "2. a factor of safety of three is maintained against burst of the tubes during *normal* operation (as required by ASME [Boiler and Pressure Vessel (BPV)] Code, Section III);
> "3. available margins to failure under postulated accident conditions are comparable to those margins provided by Appendix F 'Rules for Evaluation of Faulted Conditions' of Section III of the ASME [BPV] Code for such loadings for all other components of the reactor coolant pressure boundary;
> "4. crack-type defects that could lead to tube rupture, either during normal operation or under postulated accident conditions, will not be accepted;
> "5. the natural frequencies of the tubes will be sufficiently different from the exciting forcing frequencies during normal operation, as well as during postulated accident conditions, so that the steam generator tubing supports will not experience any damaging vibrations;
> "6. the fatigue effects of cyclic loading forces will not cause failure of thinned tubes or tubes with service induced defects during normal operation or postulated accidents. The design transients which produce the cyclic loads, include normal power operation, expected instrument failure, equipment malfunction, and operator errors which result in reactor trips."

In satisfying the above stated six design criteria the Staff believes that the requirements of criteria 14, 15, 31 and 32 of Appendix A to 10 CFR Part 50 are also met. (Staff Witness Rajan at 3 following Tr. 5847)

102. The NRC Staff concluded that the steam generator tubes, tube sheet and other components for the proposed Unit 2 have been designed to meet all requirements of the Staff's design criteria to accommodate the system pressures and temperatures obtained under all expected modes of operation including all anticipated transients and to maintain the stresses within applicable limits. It is the Staff's view that this design assures that the steam generator tube integrity will not be reduced below the level

acceptable for adequate margins of safety and is also considered adequate for issuance of a construction permit. (*Id.* at 4 through 18 and at Tr. 5850)

103. Focusing on the safety issues associated with the consequences of postulated loss of tube integrity and the necessary controls and surveillance requirements to provide reasonable assurance that steam generator tube integrity is not reduced below a level for acceptable service, Staff Witness Almeter listed five specific criteria, the meeting of which would demonstrate an adequate margin of safety. The specific criteria are:

"1. The steam generators shall be of advanced design features with improved secondary water flow characteristics.

"2. The design of the steam generators shall permit in-service inspection of the tubes by methods that will detect incipient tube degradation. Tubes that could further degrade to marginal conditions shall be taken out of service by plugging.

"3. The secondary system water chemistry shall be compatible with steam generator tube material to minimize the probability of tube degradation.

"4. Provisions for monitoring the secondary water chemistry shall be included. These shall be used to detect the presence of deleterious impurities before significant tube degradation can occur.

"5. Provisions for monitoring reactor coolant leakage to the secondary side shall be included in the design, and the limits on such leakage established to assure that tube degradation, should it occur, will be detected before it develops into serious deterioration of tube integrity." (Staff Witness Almeter at 3 and 4 following Tr. 5847)

104. The steam generator tubes in proposed Unit 2, as a part of the reactor coolant pressure boundary, are designed to Class I requirements of the ASME BPV Code Section III in conformance with 10 CFR Part 50.55(a). Design of the steam generator tubes considers "faulted conditions" (PSAR, § 5.5.2) of which a LOCA is included. (Applicants' Panel at 11 following Tr. 6021.) PSAR §§ 5.5.2 and 5.2.1.2 present the transients for which the reactor coolant system is designed. These events are far in excess, in both number and severity, of those which are anticipated to occur during the life of the facility. (*Id.*)

105. The steam generator is of an advanced design utilizing an integral economizer to improve secondary circulation. In the economizer design the feed water enters near the bottom of the secondary side of the steam generator and flows across the tube sheet, thus minimizing the susceptibility for solids accumulation on the tube sheet. (Applicants' Panel at 13 following Tr. 6021) This design has the potential of minimizing the local

concentration of impurities and the deposition of solids carried in by the feed water, thereby providing further protection against tube degradation by stress corrosion cracking and localized tube wall thinning (wastage). Applicants have committed in Amendment 18 to the PSAR, dated April 28, 1975, to conduct a steam generator development program to confirm the adequacy of the integral economizer design. (Staff Witness Almeter at 5 following Tr. 5847)

106. The chemical treatment of the secondary coolant of Unit 2 will not utilize the phosphate method which has been related to corrosion at some plants. The secondary water will be treated by an all-volatile method (AVT) in conjunction with full flow demineralization to minimize the buildup of caustic-forming impurities and scale-forming solids in the steam generator. (Staff Witness Almeter at 6 following Tr. 5847)

107. The steam generator tubes will be made of Inconel-600, which is resistant to corrosion by chloride impurities, thereby reducing potential degradation from seawater intrusion. (Staff Witness Almeter at 5, following Tr. 5847) Additionally, the design thickness of the tube wall incorporates a general corrosion allowance that will provide for reliable operation over the plant lifetime. (Applicants' panel at 13-14 following Tr. 6021) Inconel-600 is also resistant to degradation by radiation. (Staff Witness Almeter at Tr. 5863)

108. The design of the Unit 2 steam generators permits in-service inspection of the tubes by methods that will detect potential tube degradation. Periodic surveillance in accordance with Regulatory Guide 1.83 will detect tube degradation in a timely manner to permit plugging. (Staff Witness Almeter at 5 following Tr. 5847)

109. Localized corrosion has led to steam generator tube leakage in some operating reactor plants through stress assisted caustic cracking and wastage. The caustic stress corrosion type of failure is minimized by controlling bulkwater chemistry to a specification which reduces free caustic in the generator. Wastage has not been a problem when the AVT control was used. (Staff Witness Almeter at Tr. 5889, *et seq.*) Monitoring of the secondary water chemical properties will detect out-of-specification conditions on a timely basis to minimize buildup of impurities in the steam generators. Maintaining low levels of impurities will decrease the probability of steam generator tube degradation and enhance tube integrity. (Staff Witness Almeter at 6 following Tr. 5847) Witness Almeter stated that he would expect no more corrosion or erosion to occur in the secondary system than occurs in the primary system which he estimated at 0.02 mils per year or less. (Tr. 5891) He also stated that with the program outlined by the Applicant in the PSAR he would not anticipate any problem with stress corrosion. (Tr. 5890)

110. Prior to the May 1976 testimony a phenomenon known as tube denting was observed only in those steam generators with coolant treated with phosphate secondary water chemistry for some time before conversion to AVT. (Staff Witness Rajan at 2 following Tr. 9044) The subject of denting was not discussed in Applicants' or Staff's earlier testimony. Recently, however, denting has been observed at two CE PWR facilities—Main Yankee and Millstone Unit 2—both of which have operated exclusively on AVT. In this context the Applicants and Staff presented evidence regarding the denting phenomenon in relation to Unit 2. (Applicants' Witness McCracken following Tr. 8903 and Staff's Witness Rajan following Tr. 9044)

111. According to Applicants' Witness McCracken, operating experience and laboratory testing indicate that at least the following conditions must exist simultaneously to produce denting: (a) a region capable of concentrating impurities must exist adjacent to a tube (historically an annulus between a tube and a tube-support plate); (b) a carbon steel tube support plate; and (c) the ingress of impurities that can produce a local acidic environment. Denting is caused by accelerated corrosion of carbon steal in the tube/tube support plate annular regions. The corrosion product magnetite has about half the density of carbon steel. As corrosion proceeds the tube is crushed by an advancing front of hard adherent magnetite. Applicants' Witness McCracken and Staff Witness Rajan both discussed the conditions necessary to produce denting and the improvements in the mechanical design, the materials of construction and the operating procedures which would eliminate or at least minimize the potential for denting of the steam generator tubes proposed for Unit 2. The improvements are:

(1) Tube support design. The tube bundle will be supported by egg crate structures rather than drilled support plates. These have improved flow characteristics and improved corrosion resistance in the region of contact with the tubes. The resistance to denting in tubes supported by egg crate structures has already been verified in CE plants, whereas denting has been observed at the drilled carbon steel support plates.[46]
(2) Flow distribution baffles. These baffles will provide balanced flow through the economizer and boiler region of the tube bundle thereby minimizing areas of low flow velocity where sludge

[46]Denting in the tube sheet region has not been a problem with CE commercial steam generators apparently because of the process used in joining the tubes and tube sheet which virtually eliminates areas where denting might occur. (Applicants' Witness McCracken at 6 and 7 following Tr. 8903)

buildup might occur. Additionally, the flow velocities in this region will be sufficiently high that the area between the tube and its support will be swept clean of any suspended corrosion products or low solubility materials.

(3) Tube support materials. The egg crate support structures will be fabricated of type 409 stainless steel. The flow distribution baffles are type 405 stainless steel. Neither of these materials is susceptible to the accelerated corrosion responsible for denting at carbon steel supports. (Rajan testimony following Tr. 9044; McCracken testimony following Tr. 8903)

In addition, improvements in the steam condenser design, the incorporation of full flow demineralization procedures, and the adaption of condenser leakage detection systems will allow better control of the purity of the secondary coolant and will minimize the possibility of producing a local acidic environment—one of the necessary conditions for denting. (Applicants' Witness McCracken at 7 following Tr. 8903; Staff Witness Rajan at Tr. 9046)

112. CE steam generator tube integrity has been identified as one of a group of "generic issues" which the Staff has reviewed. (SER Supplement 4, Appendix D at p. D-16, 17, Task A-4 Combustion Engineering Steam Generator Tube Integrity.) The Staff concluded that based on its review of the measures that will be taken by Applicants to assure that the tubes will not be subjected to conditions that will cause deleterious wastage or cracking, a construction permit for Unit 2 can be issued with reasonable assurance that there will be no undue risk to the health and safety of the public. The Staff further stated: "The efforts under A-3 (Westinghouse Steam Generator Tube Integrity) regarding steam generator tube integrity may result in improved criteria that could provide further assistance in this regard. However, such improvements are likely to be procedural rather than system modifications and their application to the Pilgrim Unit 2 facility is a matter that can reasonably be left to the operating license stage of review. Accordingly, our previous conclusions in the Pilgrim Unit 2 SER regarding the issuance of a construction permit are unaffected by this on-going generic task." (*Id.* at D-17)

113. Seawater will cool the condenser at Unit 2 through tubes of a titanium alloy. Titanium also resists chloride attack and will reduce the probability of seawater in-leakage through the condenser system. (Staff Witness Almeter at 5, following Tr. 5847) Titanium has been used very successfully for this purpose in a number of nuclear power plant

condensers, for example at San Onofre 1 in California. (Applicants' Witness McCracken at Tr. 8906, 8907)

II. G. b. Adequacy of Regulatory Staff Inspection Practices

114. Staff Witness Reinmuth described the NRC program for the inspection of nuclear power plant manufacturers. He testified that in considering the overall adequacy of the NRC's inspection effort, it is important to note that there are four levels of inspection which follow the defense in depth concept. The first level is the requirement that each individual vendor have a directly employed inspection staff independent of the personnel actually performing the manufacturing work (vendor QA program requirements). The second line of inspection (or defense) is that provided by the buyer's (Applicants or their agent) inspection activity and as specified in Appendix B to 10 CFR Part 50. The third level in the case of ASME coded products is a third party review of vendor's QA program. Coded-product vendors must contract for the services of one or more authorized inspectors. The inspectors are employed by a state or an authorized inspection agency, usually an insurance company. Before a coded product is used in a reactor facility, that product must be stamped with an ASME code symbol and a report prepared certifying that the product meets code requirements. The code inspector as well as the manufacturer (vendor) must sign the certification. The fourth level of inspection is that performed by NRC, which is an audit of each of the other levels and thus provides assurance that the much larger program of the other three levels is effectively carried out. The total nuclear inspection activity is thus pyramided, with each layer of activity verified, inspected and/or audited by those above. (Staff Witness Reinmuth at 8 and 9 following Tr. 4520)

115. Reinmuth described in detail the NRC inspection program directed to vendors. This program utilizes a special technical staff, highly qualified both by education and experience, who inspect vendors on a nationwide basis. Typical vendors inspected included nuclear steam supply systems suppliers, architect-engineering firms and manufacturers of components. The selection of vendors and the frequency of inspection depends upon the importance of the product or the service to safety, the inspection efforts of others, the past performance of the particular vendor and necessity of investigating problem cases that may arise. During a typical eleven-month period this process included 149 inspections of 104 vendors including eight team inspections of Bechtel and three of CE. These inspections were conducted by a staff of thirteen. (Staff Witness Reinmuth, at 3 through 6 following Tr. 4520)

116. In response to questioning by Intervenor Cleetons concerning a 1973 task force report alleging deficiencies in the NRC vendor inspection practices, Staff Witness Reinmuth testified that there had been substantial improvement in the vendor inspection effort since 1973 evidenced, in part at least, by a significant increase in inspector manpower. (Tr. 4536 through 4540) The witness, in response to questioning by members of the Board, further ascribed the recent improvement in inspection practices to the growing role and acceptance of quality assurance and the general upgrading and improvement of relevant codes and standards. (Tr. 4559 through 4562)

II. G. c. Compliance with Appendix I

117. Appendix I to 10 CFR Part 50 requires that, in addition to demonstrating compliance with certain numerical guidelines on design objectives for doses to individuals from radioactive effluents released to unrestricted areas,

> "...the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis." (10 CFR 50 Appendix I § II.D)

118. Applicants' Witness Larson testified that the procedure used to demonstrate compliance with the numerical guidance of Appendix I, was as follows:

> "We began with a base case system (identified as Alternate A) capable of meeting the liquid effluent numerical design objectives of Appendix I for doses to individuals. Then we added equipment sequentially to this system defining such incrementally augmented systems as alternatives B, C, D, E.... From this radiological and cost information, using

values of $1000 per total body-rem and thyroid-rem, it was determined whether or not the system augment was required." (at 10 following Tr. 7248)

119. Descriptions of the Applicants' "base case" radwaste treatment systems are contained in Applicants' PSAR §11.0 and particularly Figures 11G-1, 11G-7-1, 11G-8, 11G-9, 11G-10, and 11G-11-1. (See also Applicants' testimony following Tr. 7248 and Staff testimony following Tr. 6482 and 7659)

120. The current design of the liquid radwaste system of Unit 2 incorporates Alternates B through E as augments to the base case, Alternate A. An analysis by the Applicants of the effects of these four additions on the basis of the $1000 per man-rem guide (10 CFR 50 Appendix I, § II.D) shows that none of these additions (not even B alone) will provide whole-body protection of the population from exposure at a cost of $1000/man-rem or less. [The Applicants and the Board have interpreted the language of Appendix I as a total annual cost of $1000 to reduce the exposure by 1 man-rem per year. (Applicants' Witness Larson at Tr. 7301)] The addition of the initial augment, leading to Alternate B, is expected to reduce the exposure by 14.1 man-rem/year at a total cost of $49,400 per year or $3500/man-rem. Similarly, augments leading to Alternate E will reduce the exposure at a cost of $69,000/man-rem. (PSAR Table 11G-15 also included in Applicants' Panel testimony following Tr. 7248 as BECo Exhibit RA-1) At this stage of design and review, however, it will not be economic to make alterations removing these augments. (Applicants' Witness Larson at 11 following Tr. 7248)

121. Similar analyses were made of the six separate systems from which gaseous radioactive effluents are expected to derive. These are:

1. gaseous waste management;
2. vent collection;
3. condenser air ejector;
4. containment purge;
5. auxiliary building vent; and
6. turbine building ventilation.

122. The last of these was predicted to release so little activity when designed as Alternate A that no perceived augment would be cost effective and, consequently, no additional analysis was made. Analyses of the

remaining five, however, showed that, as in the liquid effluent study, the annual cost of viable augments would be greater than the prescribed $1000/man-rem. It is noted, however, that Item 1, the gaseous waste management system, has been designed to Alternate E, i.e., with four augments, Items 2 and 4 have been designed to Alternate B, while Items 3 and 5 remain in the base case. (PSAR Tables 11G-16, -28, -34, -43, and -49 also included as BECo Exhibits RA-2 through RA-6 attached to Applicants' Panel testimony following Tr. 7248; see also Applicants' Witness Larson at 13 and at 24 through 29 following Tr. 7248)

123. The Staff independently evaluated the Applicants' radwaste systems for conformance with Appendix I requirements. The Staff evaluation consisted of (1) a review of the Applicants' radwaste systems and supplemental information describing the plant and the environment within a 50 mile radius as described in the PSAR, the ER and information provided by letter in response to Staff request; (2) independent Staff calculations of expected radioactive release based on PWR operating experience; (3) calculation of individual and population doses out to a 50 mile radius in accordance with standardized methods; (4) evaluation of the cost-benefit ratio for potential radwaste-system additions in accordance with standardized methods. The Staff's standardized methods for evaluating compliance with Appendix I are found in NUREG-0017 and Regulatory Guides 1.109 and 1.110. (Staff Witnesses Weller and Gotchy, at 1 through 4 and Table 3 following Tr. 6482 and following Tr. 7659)

124. The Staff's independent evaluation of the Applicants' design concluded that the release of radioactive materials from Unit 2 will not result in exposure of any individual in an unrestricted area in excess of limits established by 10 CFR § 50.34a and Appendix I. These are specifically:

(1) an annual dose or dose commitment from all radioactive materials released in liquid effluents in excess of 3 mrem to the total body and 10 mrem to any organ;

(2) an annual dose from all radioactive materials in gaseous effluents in excess of 10 mrad for gamma radiation and 20 mrad for beta radiation; and

(3) an annual dose or dose commitment from radioactive iodines and particulates in excess of 15 mrem to any organ.

(Staff Witnesses Weller and Gotchy at 4, 5 and Table 4 following Tr. 6482 as amended by supplemental testimony at 2, 3 and Table 4 following Tr. 7659)

125. Population dose calculations from liquid and gaseous releases from Applicants' "base case" systems for radwaste handling and treatment demonstrate compliance with Appendix I effluent design objectives. The calculations show less than 1 man-rem to the total body and less than 1 man-thyroid-rem from liquid releases and 1.8 man-rem to the total body and 3.4 man-thyroid-rem from gaseous releases. (*Id.* at 5 and Table 5 as amended by p. 3 and Table 5 following Tr. 7659) In accordance with the $1000 per man-rem criterion the maximum expenditure that could be required for a radwaste system augment is less than $1000 for liquids and $3400 for gases assuming that augments would reduce the discharge to zero.

126. No evidence was presented to refute the Applicants' demonstration that the design of Unit 2 complies with the ALARA Standards of Appendix I to 10 CFR 50.

III. FINDINGS OF FACT - SITE SUITABILITY

A. Geography and Exclusion Area

127. The proposed site for Unit 2 is a 528-acre tract on the western shore of Cape Cod Bay located in the Town of Plymouth in Plymouth County, Massachusetts, approximately 4.4 miles east-southeast from the center of Town and approximately 38 miles southeast of Boston, Massachusetts. (PSAR § 2.1.1; SER § 2.1, following Tr. 3717)

128. The site is generally rectangular in shape, about 0.45 miles wide, with its long dimension of 1.8 miles roughly parallel to the Bay shore. The elevation varies from sea level to approximately 280 feet. Open water occupies about 60 percent of the area within a 50-mile radius. (PSAR § 2.1.2)

129. BECo owns all of the land and mineral rights within the site boundary except for a triangular tract adjacent to the exclusion area and a portion of the land beneath the proposed Unit 2 structure which is jointly owned as tenants-in-common with the other Applicants. BECo possesses authority to determine activities. (PSAR § 2.1.2; SER § 2.1 and SER Figure 2.2)

130. Applicants can control all activity within the exclusion area except for the use of a public way, Rocky Hill Road, which provides access to Priscilla Beach. The Plymouth Police Department has agreed with BECo to barricade this road at site boundaries in the event of an emergency. Visitors, at Applicants' discretion, are permitted inside the exclusion area to

the station overlook and to the shorefront-breakwater recreational area. (PSAR § 2.1.2.1; SER § 2.1)

131. The Staff has calculated the radiological doses for postulated design basis accidents and has concluded that 10 CFR Part 100 doses will not be exceeded at the boundary of the exclusion area which, at its closest point, is 441 meters from the proposed Unit 2. (PSAR § 2.1.2.1; SER § 2.1 and Table 15.2; Staff Exhibit 9, Report on Site Suitability at 3 following Tr. 7466)

III. B. Demography, Low Population Zone and Population Center Distance

132. Guidance on considerations of the demography of a proposed nuclear generating station site is furnished to the Staff through § 2.1.3 of the Standard Review Plan.[47] The population density within a 30-mile radius of the proposed installation need not be a factor in comparison of alternative sites if the density at startup and at the end of the projected life[48] do not exceed 500 and 1000 persons per square mile, respectively. (Staff Witnesses Grimes and Soffer at 2 following Tr. 1842)

133. On the basis of data from the 1970 census updated by the Applicants to August 1975, the Staff projects the maximum population densities in 1980 and at the conclusion of plant operation to be 370 and 903 persons per square mile, respectively. Accordingly, the Staff concluded that no special consideration of demography was necessary in the review of alternate sites. (at 3 following Tr. 1842)

134. Since the proposed site of Unit 2 is coastal, an area of a 30-mile radius[49] centered at the site encompasses both land and water areas. The Staff determined the population density to be the ratio of the population within the circle of 30-mile radius to its area regardless of the topography. (Tr. 1903, 1920)

135. In a rationale of this method Staff Witness Grimes pointed out that the dispersal of air-borne contaminants is distributed in a manner characteristic of a wind-rose regardless of the population distribution.

[47]NUREG-75/087, "Standard Review Plan For The Review Of Safety Analysis Reports For Nuclear Power Plants LWR Edition" (September 1975).

[48]The most recent schedule for Pilgrim 2 establishes commercial operation during December 1985. (Applicants' Exhibit 22 served September 25, 1980, at 20) The population density lies commensurate with this revised schedule will be considered in the forthcoming hearings on emergency planning.

[49]The 30-mile distance was established as the limit within which the effects of a major accident may be significant. (Grimes Tr. 1904)

During onshore winds, exposures may occur; during offshore winds, there would be few if any exposures. The Staff's method, therefore, provides a suitable averaging process. (Staff Witness Grimes at Tr. 1921)

136. The area adjacent to the Pilgrim site is attractive to vacationers and tourists. The present permanent population of Plymouth is 29,000; additionally, there are 14,000 seasonal residents for periods of three to four months. An estimated 300,000 transients are in the area in the course of a year, most for only a few hours. In determining the population density the Staff considered a weighted average of these three components. (Staff Witnesses Grimes and Soffer at 3 following Tr. 1842; Tr. 8446, Tr. 8453)

137. The testimony of Cleeton Witness Frieden on the population density in the vicinity of the proposed Unit 2 does not disagree with that of the Staff.[50] He presented the findings of two local regional planning groups including some projected population statistics. (Tr. 8417) A detailed comparison of these data with the Staff's values was not satisfactory because the boundaries of the several areas considered did not coincide.

138. Witness Frieden estimated the population density of Plymouth would be 310 persons per square mile in 1980; 400 persons per square mile in 1990; and 450 persons per square mile in 1995. In these determinations he "did not consider water areas." (Tr. 8434, 8436) Presumably the densities will be less by the Staff's method. He judged that an area of a 30-mile radius around Unit 2 would have a population density in 1986 less than 500 persons per square mile were the salt water regions included. (Tr. 8448, 8452) He also concluded that the population projections appearing in the Staff's Site Suitability Report (Staff Exhibit 9 following Tr. 7466) are not grossly in error. (Tr. 8438)

139. In accordance with Commission Regulations [10 CFR §100.11(a)] the Applicants and the Staff established a low population zone (LPZ) and a population center distance.[51] These entities are related by the requirement that the distance from a reactor to a population center shall be at least one and one-third times the radius of the LPZ.

140. Investigations by the Staff of the distribution of population and of community institutions in and near central Plymouth led to the identification of an area encompassing those institutions and other parts of Plymouth as the population center. The corresponding distance from the Pilgrim Site is about 3.1 miles. Accordingly, the radius of the LPZ was set by the Staff at

[50]Mr. Frieden's oral testimony begins at Tr. 8406. No written testimony was submitted.

[51]Absent a specific description, Applicants and Staff chose as a population center an area enclosing 25,000 individuals with a density of 2,000 persons per square mile at any time during the lifetime of the reactor. (SER Supplement 1 at 2-1 and 2-2 following Tr. 3717)

2.3 miles, maximum. A Staff evaluation of the potential radiological consequences at a distance of 2.3 miles from a design basis accident concludes that they will be within the guidelines of 10 CFR Part 100. (SER Supplement 3 at 2-5 following Tr. 8921)

141. Although the record does not support the acceptance by the Applicants of an LPZ of this size, Revision 38 of the PSAR describes an LPZ of 2.3 mile radius. (PSAR at 2.1-5, Staff Witness Licitra at Tr. 8927; see also discussion at Tr. 9087)

III.C. Nearby Industrial, Transportation and Military Facilities

142. Except as noted in the following paragraph, there are no nearby industrial, transportation or military facilities which would affect the suitability of the proposed Unit 2 site. (PSAR § 2.2.1; SER § 2.2) There are no airports within five miles of the proposed site and no aircraft flight patterns within two miles. State Highway No. 3A (a 2-lane undivided paved road) is approximately 0.7 miles west of the site and State Highway No. 3 (a 6-lane divided road) is approximately three miles west of the proposed site. Boats which use Plymouth Harbor pass two to three miles north of the station, and ships which use Cape Cod Canal pass about four or more miles east of the station. (PSAR §§ 2.2.1, 2.2.2; SER § 2.2)

143. A witness for Intervenors Cleeton testified that contrary to PSAR § 2.2.1 there is a petroleum products storage facility (fuel tank farm) within five miles of the proposed site. (Frieden Tr. 8422) Following that testimony the Applicants determined that such a facility is located about one-half mile south of Route 3 in Plymouth at a point about 4.1 miles from the proposed Unit 2. The Facility is licensed to store 30,000 gallons of diesel fuel, 250,000 gallons of No. 2 fuel oil, 120,000 gallons of gasoline, and 75,000 gallons of propane. Applicants have assumed that at a time when the wind is blowing from the propane tank toward Unit 2 the tank ruptures when its content is the licensed maximum, and thereafter all the propane boils within a very short time, forming an initial puff which moves through adverse meterological diffusion condition and ignites at the worst possible location. Under these assumptions, the over-pressure at Unit 2 due to the propane explosion would be less than 0.1 psi. The Applicants further calculated that the over-pressure to be expected from a gasoline vapor-air mixture under similar conditions would be significantly less than those caused by a propane release and that releases of diesel fuel or No. 2 fuel oil would not result in explosions at locations away from the storage tanks. Safety-related structures at the proposed Unit 2 are designed to withstand about 2.3 psi over-pressure. The Staff has reviewed the Applicants' analysis of the risk potential of releases from the fuel tank farm and finds them conservative.

(SER, Supplement 3 at 2-5, following Tr. 9821) A comparison with an earlier independent analysis by the Staff of a similar risk potential (Wolf Creek Generating Station, Docket STN 50-482) confirms the Staff's current findings. (*Id.* at 2-6)

144. The impact of aircraft using Boston's Logan Airport on the Pilgrim site was considered by Applicants' Witnesses W. Wade Larson and Robert J. Merlino. (following Tr. 4577) That testimony was to the effect that there are no airways or airports close enough to the Pilgrim site to require an analysis of aircraft hazards to the proposed plant (*Id.* at 5 and 6) The nearest airway (V-141) is 3.6 miles distant and Logan Airport is 36 miles away. Airways more than two miles away and airports further than five miles away generally need not be analyzed under Regulatory Guide 1.70. The exception is that airports having a large number of operations must be considered, under the Guide, if a formula relating distance from the site and the number of annual operations yields certain results. Considering the number of operations at Logan, it does not need analysis under the Regulatory Guide. (*Id.*)[52]

145. Applicants analyzed the effect on the Unit 2 site of Logan Airport traffic and Airway V-141. In 1974, Logan had 295,000 aircraft movements and the Airway had a daily average of about 70 flights, ranging from 100 in the summer to 40 to 50 flights in the winter. Applicants' witnesses testified that the probability of an aircraft using Airway V-141 crashing at the Pilgrim site, calculated according to Section 3.5.1.6, Aircraft Hazards, of the NRC Standard Review Plan (NUREG-75/087), would be less than one in 10 million per year. Aircraft landing and taking off from Logan, except when on Airway V-141, do not generally approach the Pilgrim site. (*Id.* at 3 through 10)

146. On cross-examination by Mrs. Cleeton, Applicants' witnesses testified that the calculations of crash probability assumed that the aircraft always flew on course. (Tr. 4583) Absent knowledge by the witnesses of an established width of this, or any airway, a value equal to twice the distance from the center of the airway to the site of Unit 2 was assumed in the probability calculation. (Tr. 4627) This selection effectively abuts, in a vertical plane, the airway and the plant. It was further elicited by Mrs. Cleeton that aircraft movements at Logan were 316,744 in 1971; 306,202 in 1972; and 307,257 in 1973. (Tr. 4584) The applicants have not projected Logan movements in 1985 or beyond but it was the judgment of the witness

[52]Exempted are airports where projected operations are less than 1000 d^2 where d, the distance in miles between the airport and the site, is greater than 10. In this instance the number of operations limiting the analysis is greater than 1,000,000. The Applicants and the Staff, however, did address the subject because it was a contention of one of the intervenors. (Staff Witness Fontecilla following Tr. 4654)

that considering the possibilities for expansion of Logan, the movements in the future will not significantly vary from the present number. (Tr. 4584) Private aircraft deviate from the airways. (Tr. 4594, 4595)

147. Staff Witness Fontecilla testified that the Pilgrim site is outside the Logan Airport controlled airspace. It is also outside an area within five miles of the runways at Logan and accident rates are highest within that distance of runways. (At 1 following Tr. 4654) Commercial traffic approaches Boston on Airways V-16, V-139 and V-141. V-16 is at least 40 miles from the Pilgrim location and V-139 passes about 20 miles away. The center of V-141 passes about 3.5 miles east of Pilgrim. (*Id.* at 8) There are about 100 to 150 flights at altitudes from 2000 to 5000 feet on V-141 in the summer of which 40 to 50 are scheduled flights of an airline running between Boston and Hyannis and the balance are unscheduled light craft. These flights, as they pass Pilgrim, are not on descending flight paths. (*Id.* at 2) The Staff calculated a probability of less than one in 30 million per year of an aircraft crashing into Unit 2 from V-141 and resulting in radiological consequences in excess of those defined in 10 CFR Part 100. (*Id.* at 1)

148. On cross-examination by Mrs. Cleeton, Witness Fontecilla said that airplanes frequently fly off-course and sometimes descend when not landing. (Tr. 4663) He said further that the difference in his calculation of a damaging crash probability and that of the Applicants, is that he considered only commercial and military flights and assumed that the other aircraft on V-141 were private or general aviation aircraft too small to cause significant damage to Unit 2 in a crash. (Tr. 4664) The witness thought there would be little reason for military traffic from Boston to Hyannis. (Tr. 4665)

149. The Plymouth Airport is nearby but cannot handle aircraft of more than 12,500 pounds; a plane of that size could not cause significant damage to the plant but it could start a small fire. (Tr. 4666) This was expanded upon in response to a Board inquiry and Witness Fontecilla said that a nuclear plant in the Pilgrim area would be designed to withstand tornadoes of 360 miles per hour and associated missiles as well as seismic events and for that reason it could withstand a crash by a light aircraft. (Tr. 4672)

150. Witness Frieden testified that the airport at Plymouth is not within five miles of the Pilgrim site and that he was not knowledgeable of the landing pattern. He said the Planning Board of Plymouth reported 25,000 annual operations with light unscheduled aircraft at Plymouth Airport in 1972 and that an operation is either a landing or a takeoff or a touch-and-go procedure. The Planning Board predicted 75,000 operations by 1982 and 141,000 in 1992. (Tr. 8424-8439)

151. The aircraft activity at Weymouth Naval Air Station, located nearly 25 miles northeast of the Pilgrim Site, and its potential effect on Unit

2 was analyzed by neither the Applicants nor the Staff. The Weymouth air traffic is sufficiently low to place an analysis under the distance exemption of Regulatory Guide 1.70. (Staff Witness Merlino at Tr. 4602)

III. D. Hydrology

152. Cape Cod Bay is a broad, open-mouthed body of water facing northward, having a surface area of approximately 365,000 acres. The Unit 2 Site, adjacent to the shore, is in a rectangular drainage basin the long axis of which runs approximately parallel to the shoreline. The immediate plant area of 50 acres is flat and gently slopes toward Cape Cod Bay. This area will be drained by a system of catch basins and culverts will flow directly into the Bay. The western section of the basin drains in a northerly direction to a marshy area which flows into a peat bog south of the existing switchyard and parking area. (PSAR §§ 2.4.1.1, 2.4.1.2; SER § 2.4.1; Staff Exhibit 9 at 5 and 6)

153. The grade elevation at the site of the proposed structures is approximately 22.5 feet above mean sea level (MSL). Except for the intake structure, all of the exterior accesses to safety-related structures are at or above the elevation 23 feet MSL. (PSAR § 2.4.1.1; SER § 2.4.1; Staff Exhibit 9 at 5 and 6)

154. The Applicants and the Staff have evaluated the potential for flooding the safety-related structures. The probable maximum precipitation at the site would result in water levels of 23.5 feet MSL. This is about 0.5 feet above the floor grade on the turbine building side. If such flooding were to occur, there might be water seepage around a small door in the auxiliary building and around four doors leading into the turbine building. This leakage could be discharged by the in-plant drainage system, including the sump pumps located in the turbine building basement. (PSAR § 2.4.10.1; SER § 2.4.2) The proposed plant grade elevation of 22.5 feet above MSL for Unit 2 will provide adequate protection against the potential for flooding from the maximum probable precipitation.

155. The Applicants and the Staff have analyzed the potential for flooding due to a maximum probable hurricane. In such a case, waves could reach a height of about 3 feet on the northerly face of the auxiliary building. It will be waterproofed to protect against possible damage. The maximum leakage around each of its closed doors is estimated to be 20 gpm; this will be handled by the floor drain system and will be well within the capacity of the sump pumps. (PSAR § 2.4.10.2; SER § 2.4.1.1; Staff Exhibit 9 at 5 and 6) The proposed plant grade elevation of 22.5 feet above MSL for Pilgrim Unit 2 will provide adequate protection against the potential for flooding from the maximum probable hurricane and its wind and wave effects.

156. The intake structure, protected by breakwaters, is designed for a Bay-water surge level of 14.7 feet above MSL. A reinforced concrete substructure and the superstucture which houses the service water pumps will be designed for the static and dynamic effects of these waves. (PSAR § 2.4.10.2) The safety-related equipment in the intake structure will be protected against the maximum probable flood. Drawdown of water at the intake structure may occur due to the stress of offshore winds. The Applicants and the Staff agree that the probable maximum drawdown of water at the intake structure due to a hurricane will be 10.1 feet below MSL. (PSAR § 2.4.11.2; SER § 2.4.3) This predicted minimum low water level will be about 13 feet above the suction bell of each pump, located 23 feet below MSL, thereby assuring a dependable water supply to safely shut down Unit 2.

157. The hydraulic gradient of the groundwater under the Unit 2 site slopes toward Cape Cod Bay. Because of this flow pattern, there is little likelihood of contamination of public or private wells caused by accidental releases of radioactive materials into the groundwater. The Applicants will not use any groundwater for the operation of proposed Unit 2. (PSAR § 2.4.13.1; SER § 2.4.4)

158. The Applicants intend to use water from the Bay during shutdown of the proposed facility under normal and emergency conditions. The Bay will provide an adequate supply of water for safety-related purposes. (SER § 2.4.3)[53]

III. E. Meteorology

159. Eastern Massachusetts experiences various types of storms including intense thundershowers, snow and ice storms, hurricanes, and northeasters. Northeasters are coastal cyclones which occur during the winter and are characterized by high winds and intense rainfall. Hurricanes with high winds and intense rainfall occur occasionally. Maximum sustained five-minute wind speeds at Logan Airport in Boston since 1933 have been between 52 and 87 mph. The latter speed was during the September 1938 great hurricane which registered sustained winds of 121 mph and gusts of 183 mph at Blue Hill Observatory, Milton, Massachusetts. Between 1886

[53]By letter of July 5, 1979 Staff Counsel informed the Board of a report on break-water damage at Pilgrim Unit No. 1. Since this breakwater will be used by the proposed Unit 2, the report was relevant to this case. The report indicated that the Applicants have committed to submit the final design of the Unit 2 intake structure for review and approval by the Staff prior to commencement of its construction. (SER § 2.4.2) Also as stated on p. 5 of this report, "...the Unit 2 intake structure can be designed to withstand the design basis flood without credit for the effects of the breakwater." The Board is of the opinion that this potential problem regarding the stability of the breakwater can be resolved and concurs with the procedure recommended by the Staff and committed to by the Applicants.

and 1970, coastal Massachusetts experienced 18 tropical cyclones (sustained winds of 40 mph or more), 6 hurricanes (winds of over 74 mph), and the 1938 hurricane (wind speeds of over 125 mph). Tornados are not common; those that occur are not severe. (PSAR §§ 2.3.1, 2.3.2, Table 2.3-4; SER §§ 2.3.1, 2.3.2) Safety-related structures proposed for Unit 2 are designed to withstand tornados having a maximum wind speed of 360 mph. (PSAR § 3.3.2.1)

160. Local meteorological data have been collected at the Pilgrim site since May 1968, when a 200-foot-high meteorological tower was placed in operation. In April 1974, new equipment was placed on this tower and, in addition, a new 160-foot high tower was erected in order to collect the necessary data for the proposed facility. (PSAR § 2.3.3; SER § 2.3.3)

161. Estimation of radiation doses at the boundary of the exclusion area which might arise as a consequence of a release of radioactive materials during a design basis accident requires knowledge of atmospheric dispersion at the site. Necessary is an evaluation of the relative concentration of emissions (X/Q). Staff has recently[54] supplied a value of X/Q equal to 5.6 X 10^{-4} sec/(meter)^3 expected to occur at the 441-meter exclusion radius in a direction towards Cape Cod Bay. This is the highest value of X/Q during the time interval up to two hours, following the emission, that will prevail for more than five percent of the time. Consideration only of the periods of onshore winds reduces X/Q to 2.3 X 10^{-4} sec/(meter)^3, all other conditions remaining. The corresponding value reported earlier[55] (PSAR § 2.3.4; SER § 2.3.4, SSR at 5) is 4.0 X 10^{-4} sec/(meter)^3. This reduction in X/Q stems from relatively recent meterological data obtained during a year following May 1974 from improved monitors together with the use of a direction-dependent model which considers a) plume characteristics that deviate from theory under stable conditions with light winds, b) the existence of variable exclusion boundaries, and c) directional dependence of dispersion conditions. This modified methodology is embodied in Regulatory Guide 1.145. The recent value of X/Q [2.3 X 10^{-4} sec/(meter)^3] represents a greater dilution factor than does the value in the PSAR. Consequently, this lower value makes the Pilgrim Site even more attractive than formerly believed.

[54]Staff Exhibit 67 dated November 19, 1980 and served on the Board and all parties. This exhibit was received into the record by Board Order dated December 16, 1980.

[55]Traditionally a distinction has been made between inland and coastal sites when evaluating atmospheric conditions. The calculation of X/Q at inland sites has been based on consideration of dispersion conditions for winds in all directions whereas for coastal sites only onshore winds were included. This distinction at the Pilgrim Site, on the shore of the Bay, resulted in elimination of about 38 percent of the meteorological data. The impact of the different assumptions for the two types of sites on the value of X/Q has only recently been recognized. ("Differences in Procedures for Estimating Atmospheric Dispersion Conditions at Inland and Coastal Sites," Board Notification dated April 4, 1979)

162. To counter the potential for exposure of water borne individuals where X/Q has a higher value [5.6×10^{-4} sec/(meter)^3, maximum], the Applicants have established an arrangement with the U.S. Coast Guard whereby offshore areas will be evacuated as necessary to limit the exposure of boaters to no more than that of onshore personnel. (Staff Exhibit 67 at 2)

III. F. Geology and Seismology

163. As required by the provisions of the Commission's regulations set forth in 10 CFR, §§ 50.34(a)(1), 100.10(c)(1), and Appendix A to Part 100, the Applicants have submitted information to the Staff on the geology and seismology of the proposed site and on foundation engineering for the proposed facility.

164. There are no identifiable faults or other geological structures in the immediate vicinity of the site which might be expected to localize earthquakes there. The nearest fault which has been mapped is located about 17 miles from the site, although the Staff indicates that a possiblity of faulting exists about 10 miles away. (PSAR § 2.5.1.2.3; SER Supplement 3 at 2-9) So far as is known, the earthquake that has occurred closest to the site was 6 miles distant; this happened in 1881 and had a Modified Mercalli (MM) intensity of II. Twelve earthquakes of MM intensity V to VI have occurred within 50 miles of the site, the nearest of which was about 15 miles to the southwest. There is no indication of faulting in the vicinity which would affect the suitability of the proposed site.

165. The Applicants' submittal has combined earlier data with information from recent investigations including original field explorations and theoretical studies to comprehensively explore the correlation of the larger New England earthquakes to identifiable tectonic structures. The program was designed to review historical New England seismicity together with the results of both earlier field investigations and recent onshore and offshore geophysical research including aerial, land and marine magnetic surveys, land gravity observations, and seismic reflection and refraction data. These results were correlated with integrative theoretical models. (Applicants Witness Famiglietti at 4 following Tr. 8830) According to the Applicants, the proposed Unit 2 site is located in the southeastern New England platform. (PSAR Figure 2.5-4A) No earthquake of intensity greater than MM VI has been recorded in the tectonic province within which the Site is located. (Applicants' Witness Famiglietti at 2 following Tr. 8830)

166. The analysis centered around an intense earthquake (Modified Mercalli VIII) which occurred off Cape Ann, Massachusetts, in 1755.

Applicants' position is that the Cape Ann quake, as well as others in New England, are associated with particular mafic plutons[56] and with their setting in anomalous-faulted[57] rock of the middle Cretaceous Age. Recent studies by the Applicants and their contractors have defined the southern boundary of the Cape Ann tectonic structure as 35 miles from proposed Unit 2. (Applicants' Witness Famiglietti at Tr. 8858) While neither Staff nor its consultant, U.S. Geological Survey (USGS), are in complete agreement with Applicants' position that the larger New England earthquakes (in particular the 1755 Cape Ann earthquake) are directly associated with a unique tectonic model of cylindrical mafic plutons and tangential faults, both Staff and USGS agree that the zone of high earthquake activity extends no closer than 35 miles from the Pilgrim Site. (SER Supplement No. 3 at 2-15 and Appendix B at B-2) The differences then become academic since the end result as regards seismic design would be identical, i.e., an Intensity VIII quake at a distance of 35 miles from the Site as the basis for seismic design. Accordingly, the Pilgrim Site, being within a different seismic region than Cape Ann, is considered not to be susceptible to such extremes of tectonic activity.

167. The analysis consisted of, first, an assumption that an Intensity VIII quake would occur within the Cape Ann structure at a location most proximate to the Pilgrim Site; second, an empirical extrapolation of that intensity a distance of 35 miles to the Site; and, lastly, a conversion, also empiric, to a horizontal ground acceleration. The results of this analysis lead to the conservative specification, at the Site, of a disturbance of effective intensity MM VII producing a ground acceleration having a horizontal component equal to 0.20 g.[58] (Applicants' Witness Famiglietti at 5 following Tr. 8830)

168. Accordingly, the Applicants propose MM VII as the Safe Shutdown Earthquake (SSE)[59] and assign 0.20 g as the horizontal compo-

[56]A pluton is a large body of once-liquid rock that rose to the earth's crust and solidified. The term mafic refers to the composition of the rock. (Tr. 8863) Cape Ann is some 50 miles north of the Pilgrim site. No mafic pluton has been discovered within 50 miles of the site. (Applicants' Witness Famiglietti at 39 following Tr. 8830)

[57]An anomalous structure is one mapped on land and inferred offshore by aeromagnetic studies. (Applicants' Witness Famiglietti at Tr. 8864)

[58]Conservatism is incorporated in these values in the following ways. MM VIII is an upper limit on the Cape Ann intensity [MM VII is indicated by some observations, (at 9 following Tr. 8830)]; the proposed MM VIII is placed in the Cape Ann structure at a point nearest the Pilgrim Site; the extrapolation resulted in an equivalent MM VI at the Site corresponding to an acceleration 0.13 g. (At 5 and 16 following Tr. 8830) MM VII and 0.20 g were, however, specified; the latter incorporates a soil amplification factor.

[59]The magnitude of the SSE is conservatively estimated to be 6.0 on the Richter scale. (Applicants' Witness Holt at 10 following Tr. 8830)

nent of the corresponding ground acceleration. The Staff concurs in these values.[60] (SER Supplement 3 at 2-15 following Tr. 8921; Staff Witnesses Bennett, Jackson and Kane at 6 following Tr. 8945; Tr. 8995)

169. Applicants propose a horizontal ground acceleration of 0.1 g (one-half of the value for SSE) for the operating basis earthquake (OBE). The probability of experiencing an OBE during the 40-year operating life of the plant is estimated to be 0.1 (PSAR 3.71 and Figure 3.7-3) The Staff considers the horizontal acceleration of 0.1 g for the OBE to be acceptable and in accordance with the requirements of Appendix A to 10 CFR Part 100. (SER Supplement 4 at 2-2 following Tr. 10046)

170. The Applicants and their contractors testified to their testing and analyses of the soil at the Pilgrim Site and the assessment of the margin of safety against potential soil liquefaction[61] during a safe shutdown earthquake. (Witnesses Famiglietti, Ferris, Seed and Poulos following Tr. 8881)

171. The Applicants have investigated the subsurface materials at the proposed Unit 2 Site with 51 borings and by excavating two test pits below foundation level. The purpose of Applicants' investigation was to measure the *in situ* density of the soils and to determine from undisturbed samples the properties of the soils for the purpose of further evaluation. These investigations have shown that the soil consists of about 90 feet of dense, poorly-graded to well-graded sands and gravelly sands. The upper 20 to 30 feet of this soil contains layers of silt, silty clay, and sandy clay. Stratified sandy glacial outwash, which is the main load-bearing stratum at the site, overlies bedrock. Over the outwash is a complex mass of glacial till which is about 20 feet thick. Bedrock consists of contemporaenous igneous rocks, known as the Dedham granodiorite. Bedrock surface under the foundation area lies between 58 and 80 feet below MSL. (PSAR §§ 2.5.1.3.1, 2.5.1.3.3.1 and 2.5.1.3.3.2; SER at 8)

172. Liquefaction potential of sand is affected by its density, grain structure, history of subjection to sustained pressures, lateral earth forces, and prior seismic or other shear experience. A measure of resistance to liquefaction is the resistance to penetration. (Applicants' Witness Seed at 26 following Tr. 8881)

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[60]In a recent finding the Appeal Board associated a ground acceleration of 0.15 g with a MM VII event. [Consolidated Edison Company of New York, Inc. (Indian Point, Units 1, 2 and 3) ALAB-436, 6 NRC 547 at 624 (1977)] The larger acceleration in the instant proceeding arises, in part, from the local soil conditions.

[61]Soil liquefaction is the potential for development of strain deformations when sand, under load and saturated with water, is subjected to strong earthquake motion.

173. Although the density of the foundation soils, measured in the field, ranged between 80 and 130 lb/ft^3, it was not possible to determine the relative density[62] because of stratification. The gross density is consistent with the 15 ton/ft^2 glacial loading estimated from topographic evidence of the 600-foot-thick glacial layer. Preparatory to determining liquefication characteristics, minimally disturbed soil samples were vibratory compacted in the laboratory in a manner comparable to that to which the soil underlaying the structures will be subjected. Laboratory samples so treated are considered representative of field properties. (*Id.* at 28 through 32)

174. From laboratory and field data, soil was characterized by a relation between penetrability, measured in blowcounts,[63] and the corresponding cyclic stress ratio.[64] An empirical relation between these properties derived from historical earthquakes, of magnitude comparable to the SSE, defines a boundary between instances of liquefication and no liquefication. Data from soils at the Pilgrim Site lie in the "no liquefaction" region. [Applicants' Witness Ferris at 13 and unnumbered p. 42 (BECo Exhibit SL-2) following Tr. 8881]

175. The characteristics of the Pilgrim Site soil establish a vertical gradient in the horizontal ground acceleration arising from seismic activity. These characteristics led the Applicants to set 0.15 g at foundation level corresponding to 0.20 g at the surface. (Applicants' Witness Ferris at 5 following Tr. 8881) The Staff does not disagree. (Staff Witness Bennett at Tr. 9024)

176. For each of many locations under and around proposed Unit 2 structures a factor of safety was determined.[65] These safety factors are 2.0 or more for a ground surface acceleration of 0.25 g. [Applicants' Witness Ferris at 17 and unnumbered p. 47 (BECo Exhibit SL-5) following Tr. 8881]

177. On the basis of MM VII as the SSE and of the properties of the Pilgrim Site soil, the Applicants conclude that an adequate margin of safety

[62]Relative density, an important property in soil mechanics, rates a sample on a scale bounded by the actual density in the loosest condition (zero) and in the most tightly packed configuration (100%). (Applicants' Witness Seed at Tr. 8891)

[63]Blowcount is a measure of penetration being the number of impacts of a 140-lb mass, after a 30-in. free fall, necessary to drive a cylindrical annulus 12 in. into a sample. The high blowcount (<100) and the behavior of the adjacent Unit 1 foundation ("small settlement") point to the high preconsolidation pressure of the soil.

[64]The stress ratio is defined as the ratio of the average horizontal shear stress (induced by an earthquake) to the effective overburden pressure. (PSAR at 2A-79)

[65]A factor of safety is the ratio of the available soil strength to the (expected earthquake) induced stress. (Applicants' Witness Ferris at 11 following Tr. 8881)

against disruptive seismic forces exists at the Site without special preparation.[66] The Staff concurs. (Staff Panel at 6 following Tr. 8945; Staff Witness Kane following Tr. 8948)

178. The Staff concludes, based on past experience with the design of nuclear power plants, that reactors of the general type and size proposed can be, and have been, safely designed to withstand an event of intensity as great as MM VIII. (Staff Exhibit 9 at 8 following Tr. 7466)

IV. FINDINGS OF FACT - ENVIRONMENTAL MATTERS

A. General

179. Pursuant to the National Environmental Policy Act of 1969 (NEPA), the Notice of Hearing in this proceeding[67] requires this Board to consider and decide:

> "5. Whether in accordance with the requirements of Appendix D of 10 CFR Part 50 [now 10 CFR 51], the construction permits should be issued as proposed."[68]

The Notice of Hearing further stated the following: "With respect to the Commission's responsibilities under NEPA, and regardless of whether the proceeding is contested or uncontested, the Board will, in accordance with Section A.11 of Appendix D of 10 CFR Part 50: (1) determine whether the requirements of Section 102(2)(C) and (D) of NEPA[68A] and Appendix D of 10 CFR Part 50 [now 10 CFR 51] have been complied with in this proceeding; (2) independently consider the final balance among conflicting factors contained in the record of the proceeding with a view to determining the appropriate action to be taken; and (3) determine whether the construction permits should be issued, denied, or appropriately conditioned to protect environmental values."

IV.B. Compliance with Sections 102(2), (C) and (E) of the National Environmental Policy Act of 1969 (as amended) and 10 CFR Part 51

a.Need for Power

180. In its deliberations on this "need for power" the Board is cognizant of the substantial margin of uncertainty attendant to any quantitative

[66]Had the results of the investigations and analyses been different the Applicants were prepared to incorporate a procedure for permanently dewatering the Unit 2 foundation. (See Testimony of Applicants' Witness Poulos at 13 following Tr. 8207 and Staff Witness Kane following Tr. 7470)

[67]39 Fed. Reg. 1786, (January 14, 1974).

[68]Items 1 through 4 are issues pursuant to the Atomic Energy Act of 1954 (as amended). Those are listed within paragraph 19 *supra*.

[68A]In 1975, Subsection D was lettered as Subsection E. The wording of the Subsection was not changed by that amendment.

prediction of energy consumption in even the near future, to say nothing of the requirement nearly a decade hence when Unit 2 is proposed to operate. In the course of these hearings several witnesses approached the problem through estimates of the anticipated growth rates of both energy requirements and peak-power demands during the last part of this century with the realization that most of the predictive analyses were fraught with uncertainties. Some estimates were based primarily on judgmental extrapolations of past energy sales modified by economic trends, home building activity, plans of major commercial and industrial developers, etc., and tempered by national reports from *Electrical World* and the Edison Electric Institute. (Applicants' Witness Sweeney at 21 following Tr. 7927) More sophisticated methods, utilizing price elasticity[69] and econometric equations, were employed by others.

181. Applicants and Staff witnesses addressed the issue of price and demand elasticities as a forecast mechanism. Applicants' Witness Guth stated that "[o]ne detects an implicit argument made by supporters of economic models of energy demand for forecasting purposes, namely, that such forecasts represent an objective non-judgmental alternative to traditional methods. But this position is simply erroneous." The witness further observed that there is very little that is objective or non-judgmental about forecasts of future economic activity, populations, price levels of fuels, prices of electricity, and capital costs. Conclusions from econometric models will reflect these uncertainties. (at 37 through 39 following Tr. 2647)

182. Staff Witness Nash pointed out that the derivation of elasticities is very complex and by its nature rather inexact. A partial analysis relating only price and electricity use can be expected to give erroneous results since the demand for electricity is determined by complex interactions of economic, social and political factors. The advantage of econometric methods over others can be questioned since models predicting electricity demand, no matter how mathematically complex, are not free from subjective factors. (at 3 through 5 following Tr. 3110)

183. Recognition must also be made of the legal responsibilities imposed upon a public utility to provide to the public it serves a continuing and reliable product with attendant severe consequences in failure to so do. It is, therefore, incumbent upon the utility to be reasonable in its forecast. [*Kansas Gas and Electric Company, et al.* (Wolf Creek Generating Station, Unit No. 1), ALAB-462, 7 NRC 320 (1978); *Carolina Power and Light Company* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4), CLI-79-5, 9 NRC 607 at 609 (1979)]

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[69]Price elasticity is defined as the ratio of the fractional change in the quantity demanded to the fractional change in price.

184. For these reasons this Board turns to the concept that, absent a firmly established "need" for the energy output of Unit 2, its operation will be favorable in the national interest because of the traditional dependence of New England on oil in the production of electrical energy. This concept has been termed "substitution."

185. It is public policy that oil, particularly imported oil, either in short supply or invaluable for other uses as in the petrochemical industry, be replaced as a fuel in steam generating plants by coal, uranium, etc. Additional uranium-fueled units may allow retirement or restricted use of fossil fueled plants unable to meet pollution control requirements without extraordinary expense and difficulty.[70]

186. Applicants' Witness Weiner testified to the fundamental bases for the need of Unit 2. One is the substitution concept, *supra;* another is achievement of predicted energy requirements discussed, *infra.* The third is the cost savings expected to accrue to New England electrical energy consumers directly from the output of Unit 2 and indirectly from the retirement of older, less efficient and more costly fossil plants. (At 3 following Tr. 10430; Tr. 10940)

187. During the course of the hearing, however, much attention was accorded the prognostications on the energy requirements of New England towards the end of this century. Thirty witnesses were presented on this aspect of the need for Unit 2: 15 by the Applicants; four by the Staff; eight by the Commonwealth Attorney General; two by the Commonwealth Governor and one by the Board from the then Federal Power Commission (FPC). The growth rates expected in the requirements for energy and for peak power were projected independently by witnesses of several parties.

188. Applicants presented their assessment of the need for Unit 2 through its ER[71] and through two panels of witnesses[72]

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[70]Power Plant and Industrial Fuel Use Act of 1978, P.L. 96-620, 92 Stat. 3289. [Public Service Company of New Hampshire, *et al.* (Seabrook Station, Units 1 and 2), ALAB-422, 6 NRC 33 at 90 *et seq.* (1977); Public Service Company of Indiana, Inc. (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-459, 7 NRC 179 at 186 (1978); Dairyland Power Cooperative (La Crosse Boiling Water Reactor), ALAB-617, 12 NRC 430 (1980), affirming the Licensing Board decision on substitution at 11 NRC 44 at 77 (1980).]

[71]Environmental Report, Applicants' Exhibit 1(k), Volume 1 at 1-1 through 1-36.

[72]The duration of these proceedings was such as to require updating some early testimony. The Applicants' *first panel* testified in December 1975 (following Tr. 2647) and consisted of: John H. Ferguson, Director of Rate Research and Forecasting Department, BECo; Donald V. Bourcier, Senior Load Analyst, NEPLAN; Abraham Gerber, Vice President, National Economic Research Associates, Inc. (NERA); Louis A. Guth, Vice President, NERA; Moshe Weiss, Senior Consultant, NERA; Kenneth O. Sten, Manager of Research and Planning Department, BECo; and Benjamin H. Weiner, Vice President, Power Supply Administration, BECo. The Applicants' *second panel* of witnesses testified on June 20, 1977 (following Tr. 7927) and consisted of: Robert O. Bigelow, Vice President and Director of Planning and Power Supply, New England Power Service Company and member of the New England Power Pool

189. Applicants are members of NEPOOL, formed in late 1974 for the purpose of enhancing the adequacy, reliability and economy of the power systems in New England. NEPOOL is responsible for planning bulk power facilities in New England. NEPOOL members serve over 95 percent of the electric power load in New England. (ER, Volume I, at 1-1; FES at 8-1)

190. NEPOOL, through its New England Power Planning Division (NEPLAN), forecasts regional load requirements as well as the type of generating capacity needed: base-load, intermediate, or peaking. Individual utility members or consortia of utilities, as in the case of proposed Unit 2, however, have the responsibility of selecting and installing new generating facilities. (FES at 8-1)

191. The interdependence and interconnection among the suppliers of electrical energy throughout a region together with the differences in seasonal demands and in other characteristics of the individual constituent utilities within the region, make imperative a close cooperative relation among those constituents. Accordingly, the unified areal projections and resources of NEPOOL are important to good business administration.

192. The need for Unit 2 in this context, as advanced by NEPOOL, is based on periodic load and capacity projections developed by the NEPLAN staff. The most recent of the series of reports of these projections introduced in this record forecasts need for the period 1980-1995.[73]

193. In its most recent report NEPOOL has forecast for 1979 through 1995 a 2.6 percent annual compound growth rate in energy, a 2.0 percent growth rate in summer peak power demand, and a 2.7 percent rate for winter demand. These estimates are about one percentage point less than the values of April 1, 1979.

(NEPOOL) Planning Committee; Stephen J. Sweeney, Vice President, Steam Operations, Environmental Affairs, Planning and Research Organization, BECo; Cameron H. Daley, Manager of the Research and Planning Department, BECo; Louis A. Guth, NERA; Moshe Weiss, NERA; and Abraham Gerber, NERA. The Applicants' *third panel* was comprised of: B. H. Weiner, BECo; Philip A. Legrow, Generation Planning Engineer, BECo; Donald V. Bourcier, BECo; Arthur W. Barstow, Manager of Generation Planning, New England Power Planning. Its *fourth panel* consisted of F. Cort Turner, Vice President, Arthur D. Little, Inc.; and Niegel Godley and David Hanna both of the Energy Economics Section, Arthur D. Little, Inc. The testimony of the fourth panel was presented on July 16, 1979 (following Tr. 10430); that of the third panel on July 18, 1979 (Tr. 10730) except that of Mr. Barstow which was presented on August 27, 1979. (Tr. 11356) The written testimony of both the third and fourth panels is Applicants' Exhibit 19 bound following Tr. 10430.

[73]"NEPOOL Forecast for New England 1980-1995," April 1, 1980; New England Power Planning, West Springfield, MA, Applicants' Exhibit 22, served on the Board and all parties, September 25, 1980, admitted by Board Order dated October 2, 1980. The earliest report in the record is for the 1976-1987 interval taken from "New England Load and Capacity Report 1976-1987" introduced by Daley at 31, following Tr. 7927. The second forecast introduced is "New England Load and Capacity Report 1978-1989," Applicants' Exhibit 20-C, introduced by Barstow, a NEPOOL employee, at Tr. 10740 (see Tr. 11360 for correction to exhibit numbers).

194. Forecasts of the electrical energy and power demands within the BECo service area have also been presented. The most recent value for the 1979 and 1989 interval is an annual compounded growth rate of 2.0 percent in energy and 2.4 percent in peak power.[74] These values are significantly reduced from earlier predictions.[75]

195. The Northeast Power Coordinating Council has established a system reliability criterion whereby the area's electric generating power supply shall be less than the area load no more than one day in ten years. To fulfill this condition, a reserve of 23 percent of the peak-power load is required by NEPOOL. (FES at 8-9)

196. NEPOOL capacity and reserve-margin projections[76] based on predicted winter-peak power loads with Unit 2 coming on line in December 1985 are given in the following table.

With Unit 2:

Winter of	**Capacity (MW)**	**Reserve Margin (%)**
1984/85	23692	36.2
1985/86	25790	44.3
1988/89	26596	36.4
1990/91	27166	31.6
1991/92	27166	27.7
1992/93	27122	23.7
1993/94	27120	19.9
1995/96	27120	12.2

Without Unit 2:

1985/86	25790	37.9
1988/89	26596	30.5
1990/91	27166	26.0

[74]"Long-Range Forecast of Electric Power Needs and Requirements" Volume 1, p. 3, Applicants' Exhibit 21-A served September 25, 1980 admitted October 2, 1980.

[75]See, for example, the 1976 energy growth estimates of 3.7 to 5.8 percent for the period 1974-1980 and 3.3 to 5.1 percent for 1980-1986, Applicants' Witness Guth at 38 and BECo Exhibit NP-27 at unnumbered p. 67 following Tr. 7927; see also BECo Exhibit NP-21 at unnumbered p. 57 following Tr. 7927 for a series of 1976-1986 comparative forecasts.

[76]Applicants' Exhibit 22 at 20 served on the Board and all parties September 25, 1980 admitted on October 2, 1980.

1991/92	27166	22.3
1992/93	27122	18.4
1993/94	27120	14.8
1995/96	27120	11.6

Notes:

a. The capacities assume the addition of two Stony Brook Units (510 MW oil turbine) in 1981 and 1982; Seabrook Unit 1 (1150 MW nuclear) in April 1983, Seabrook Unit 2 (1150 MW nuclear) in February 1985, Pilgrim Unit 2 (1150 MW nuclear) in December 1985, Millstone Unit 3 (1150 MW nuclear) in May 1986, and Sears Island (568 MW, coal) in November 1987, together with a number of smaller stations.

b. Data are given for winter peak power because power demand within NEPOOL peaks in the winter period. Power in the BECo service area peaks during the summer months.

197. Experience indicates that slippage will occur in some of the startup schedules factored into the above predictions. Consequently, the required NEPOOL reserve margin, 23 percent, may not be available as early as the late 1980s in the additional absence of Unit 2. (Applicants' Witness Weiner at 10, following Tr. 10430)

198. Further, the required reserve margin increases with increased average size of the units within the system to which the margin applies. The trend in NEPOOL to more generating stations of large capacity, such as the present 1150 MW variety, together with decommissioning older and smaller plants will require NEPOOL to increase its reserve margin above the present 23 percent. (Staff Witness Feld, Tr. 10647)

199. Staff Witness Feld[77] reviewed the NEPOOL load forecast and was in essential agreement with the anticipated schedule for Unit 2 to meet the minimum reserve.[78]

[77]Throughout the course of these hearings the Staff offered four witnesses on the need for Unit 2: H. L. Thompson (Tr. 2939); Daryl Nash (Tr. 3102); Sidney Feld (Tr. 8150 and 10499); and W. S. Chern (Tr. 11231). The Commonwealth District Attorney offered eight: H. Houthakker (Tr. 2330); Carl Stein (Tr. 3297); J. H. Neely (Tr. 3518); Henry Lee and Paul Levy (Tr. 4959); Nancy Boxer (Tr. 8583); Paul Chernick and Susan Geller (Tr. 10952). The Massachusetts Governor's Office of Energy Resources presented J. G. Buckley (Tr. 10370) and J. S. Fitzpatrick (Tr. 10656). E. N. Fields of the Federal Power Commission appeared for the Board (Tr. 6080).

[78]The continually changing and complex pattern of energy requirements over the four-year interval of hearings on this topic has made much of the earlier testimony moot. No party has responded to the most recent filing of the Applicants, on September 25, 1980, on which much of the above is based. Apart from considerations given by the Board to that NEPOOL-BECo information, greater weight has been afforded to the next most recent presentation than to the older data.

200. The Staff in collaboration with the Oak Ridge National Laboratory (ORNL) independently developed forecasting capability and applied it to the future electrical energy requirements of New England. A conclusion of this analysis, based on a 3.4 percent energy-requirement annual growth rate established for a median fuel price, is the need for Unit 2 output during the 1988/89 winter to conform with NEPOOL reserve requirements. All planned and authorized additions are assumed to have come on line. This date is not grossly inconsistent with 1993/94 reported in paragraph 196 *supra,* recognizing the reduced growth rate (2.7 percent vs 3.4 percent) factored into the latter. (Staff Witness Feld at 6 *et seq.* admitted at 10501 bound following Tr. 10651)

201. The Staff/ORNL forecast utilized a regional econometric model, the ORNL model, built around a system of non-linear simultaneous equations and containing submodels for the various types of electrical service. (Chern *et al.,* Staff Exhibit 60 at 1-3 through 1-5) The energy demands in these types of service are assumed to be functions of such entities as the population, the cost of various fuels, the per capita personal income, the number of residential customers, weather data and the value added in manufacturing. (*Id.* at 1-3) The model adjusts to a range of fuel prices. (Feld at 6 following 10651) The codes are continually updated recognizing changes in cost indices, in technological developments at generating stations, in operating and maintenance costs, etc. (Feld at Tr. 11321) During its development, the model has been validated against historical data for the period 1955 to 1974 and was again tested against more recent information obtained in 1975 and 1976. In each comparison the demand forecast, with 1955 as the base year, varied from experience by no more than three percent. (Staff Exhibit 60 at 6-1 *et seq.*; Chern at Tr. 11238, 11301 *et seq.*)

202. In the most recent sessions of this hearing the Commonwealth District Attorney presented witnesses Chernick and Geller, each being employed by that office as a rate analyst. In their prepared testimony, these witnesses offer extensive criticisms of both the NEPOOL and the ORNL models for forecasting energy demands. (Commonwealth Witnesses Chernick and Geller at 5 through 45, following Tr. 11224) The criticisms[79] of NEPOOL were principally those of inadequate documentation and errors arising from incorrect concepts and input. (Tr. 10968) Further there was assertion of inadequate consideration of the effects of conservation. In the ORNL model the price elasticities were considered to be too low, natural gas was not considered as a substitute fuel, profit margins were implausible,

[79]Although the prepared testimony was filed jointly by the two witnesses, Mr. Chernick made the criticisms of NEPOOL and Ms. Geller criticized ORNL. (Tr. 10990, 11180)

and the model is based on a faulty understanding of how demand for electricity works. (Tr. 11165)

203. Witness Chernick predicted an approximate 1 percent annual growth rate of peak demand in New England over the next decade. More specifically, he set the value between 95 percent confidence bounds of -0.5 and 2.5 percent. This prediction was subjective without basis on any particular model. (Tr. 11162, 11192 through 11196)

204. Witness Geller, the principal critic of the ORNL method, was unable to revise the growth rate forecast so that its value would reflect the alleged shortcomings of the model. Her position was that. "The equations don't make any sense." (Tr. 11165)

205. The Massachusetts Governor's Office of Energy Resources presented testimony by Witness J. G. Buckley (received Tr. 10372, bound following Tr. 10947) and by J. S. Fitzpatrick (received Tr. 10659, bound following Tr. 10947). Mr. Buckley is the Vice President of Northeast Petroleum Industries, Inc.; he is currently the chairman of the Fuel Oil Marketing Committee of the U.S. Department of Energy. Mr. Fitzpatrick is Director, Massachusetts Office of Energy Resources.

206. These witnesses testified to the severity and potential consequences to the national economy of recent practices necessitating importing large quantities of oil. There are two principal effects of this importation. It imposes a tremendous drain on this country's international balance of payments which is reflected in the domestic economy. (Tr. 10386) Second, imported oil is a very insecure source of energy on which the domestic economy and livelihood so strongly depend. (Witness Buckley at 9 following Tr. 10947; Witness Fitzpatrick at 2, 3 following Tr. 10947)

207. Witness Buckley directed his testimony to an expected future increase in the cost of oil and to the prudence of encouraging alternate sources of energy. (Tr. 10410) Witness Fitzpatrick reported the policy of the Commonwealth to "discourage new uses of oil and encourage the reduction of present levels of use." (at 2 following 10947) He attributed a 27 percent decrease in oil imports between 1973 and 1977 primarily to the installation of 2800 MWe of nuclear-fueled generating capacity. (at 6 following Tr. 10947) Activation of Unit 2 would advance the oil conservation policy of the Executive Department of the Commonwealth. (at 7 following Tr. 10947)

208. The Massachusetts Governor's Office of Energy Resources has estimated the oil-equivalent cost saving of operation of Unit 2, that is the cost of oil to produce the same electrical energy, over the 35-year life, to be the order of 10 billion dollars. (Fitzpatrick at 8, 9 following Tr. 10947)

209. It was the opinion of both of these witnesses that the forecasts of future oil prices by Applicants' witnesses were conservatively low. (Tr. 10374, 10712)

210. The Applicants' position on this matter of oil-import reduction was discussed by Witness Weiner. (at 19 through 24 following Tr. 10430) An advance of the operation of Unit 2 to 1985 from 1988 would reduce oil consumption over that period by about 30 million barrels. (BECo Exhibit NP-42, at unnumbered p. 34 following Tr. 10430)

211. As noted above the presentation of testimony on "need for power" spanned nearly four years and some extremes in economic conditions. Consequently many of the earlier data became outmoded and were replaced by more current values of expected behavior of the economy. A few topics, however, even through discussed relatively early in the hearing, were not reheard in recent sessions. Some of them are addressed now.

212. The effect of "conservation of energy" on the projected future need for electricity was addressed by Applicants' Witness Weiss.[80] (at 81 through 104 following Tr. 2647, and at 42 through 49 following Tr. 7927)

213. This witness assessed the impact of conservation by applying an econometric model to the sale of electricity in the Boston Edison service area to account for the effects of price and income and then considering any unaccounted for residual reduction to be attributable to conservation. He concluded that conservation was responsible for a significant portion of the decline of electricity consumption experienced after the 1973/74 oil embargo but that this non-price related conservation was a short-lived phenomenon and would not affect electricity consumption in the future. (at 82 *et seq.* following Tr. 2647) Later he judged that the non-price or "patriotic" conservation observed in 1974 had no effect on the 1976 growth of electricity sales and that an examination of those economic factors affecting sales revealed patterns which could be accounted for solely by economics and that regulations will be required to effect true conservation. (at 42 through 49 following Tr. 7927; also Tr. 8131, 8132)

214. Witness Weiss commented on the potential of future government action for slowing the growth of electrical consumption, i.e., financial incentives for increasing home insulation and installation of equipment to utilize solar energy but stated that the extent to which such actions would affect electricity consumption will depend on some government activity. He did, however, indicate that the Administration has a target to reduce growth rates in electrical energy requirements in 1985 by about 10 percent. This impact, while not insignificant, is clearly not of the magnitude necessary to modify substantially the need for additional base-load capacity. (at 44 and 45 following Tr. 7927)

[80]Conservation was defined as a reduction in sales over and above that attributed to the effect of price and income. Elsewhere it was similarly defined "as a curtailment which results from a decision by a member of the public to use less energy . . . which is not influenced by economic pressures." (Tr. 2968)

215. The Staff presented evidence on the effects of "voluntary" conservation and estimated the potential impact of all "voluntary" conservation actions to be "much less than 1 percent of the total energy consumption." (Thompson at 4, following Tr. 2968)

216. Staff Witness Thompson also examined the potential for conservation of electricity as a result of new standards for improved building insulation, heating, lighting, and air conditioning and concluded that the maximum potential reduction in the residential and commercial consumption of electricity in 1970 would have been 17.5 percent and 15 percent, respectively. (Tr. 2969) Reviewing the results of the Federal Power Commission's study[81] on the reduction in the growth rate of effective energy demand due to conservation and other methods, the witness concluded that a reduction of 7 percent in the historic rate could be achieved. This reduction would account for about one-half the reduction anticipated by the Staff due to the combination of energy shortage, higher costs and conservation. (at 45 and 47 following Tr. 2968)

217. Commonwealth Witness Houthakker presented an econometric equation that illustrated price and income elasticities for residential electricity sales in New England and showed that increases in marginal prices (Tr. 2346) would lead to a decrease in sales and that increases in real income would lead to increases in sales. (Houthakker testimony at 2)[82]

218. The model does not include cost of competing fuels (Tr. 2361, 2386) and fuel substitutions. (Tr. 2362) The effect of the then-current recession had not been analyzed. (Tr. 2425) The witness further stated that the thrust of his testimony was "to show that consumption of electric power, by residences, is sensitive to the prices charged ... rather than to project the consumption in particular years. These projections are made on assumptions which I do not claim to be realistic." (Tr. 2374) In response to a question whether, as a utility company, he would rely on his econometric equation to make demand forecasts, he responded "certainly not." (Tr. 2386) The quantitative results of this testimony have been superseded by more recent offerings.

219. Commonwealth Witness Neely questioned Applicants' forecasts of the need for Unit 2, the Federal Power Commission reliability criteria and the resultant reserve margin requirements, and whether adequate consider-

[81]"FPC News" August 7, 1975, No. 21622 p. 45.

[82]The testimony was admitted at Tr. 2330, was prepared in mid-1974 and was not updated. (Tr. 2335) It was received into the record as Commonwealth Exhibit 18 by Board Order dated July 14, 1978.

ation was given to alternate energy systems such as solid waste. (Neely testimony at 1 through 4)[83]

220. This witness in his critique of Applicants' forecasting recommended *inter alia,* consideration of price elasticity and the state of the New England economy. (*Id.* at 10 and 11) He mentioned the failures of the utility demand forecasts in 1974 but proposed no substitute methodology, stating that poor forecasting methods are likely to lead to under-capacity or over-capacity and there are adverse consequences of each. For example, under-forecasting leads to blackouts and adverse economic effects, over-forecasting diverts capital and entails unnecessary land-use diversion. (*Id.* at 3 through 8) He did, however, opine that there is not sufficient information for adequate review, the demand forecast is probably too high, and the supply forecast is probably too low." (*Id.* at 9)

221. In his criticism of the FPC reliability criterion of a one-day loss of load due to insufficient capacity every 10 years, Witness Neely did not perform a complete cost-benefit analysis to justify lesser reserve margins but argued instead that, "since most blackouts result from equipment failure, a doubling or tripling of the number of blackouts from insufficient generating capacity would mean only a modest increase in the total ...." and that the economic benefits from not paying the carrying costs of "excess" reserve capacity would outweigh the economic costs of more frequent losses of load. (*Id.* at 4 and 5)

222. Commonwealth Witness Stein, an architect and specialist in energy conservation projects, testified to the magnitude of reductions in electrical energy demands which could be achieved by conservation in residential and commercial buildings. (following Tr. 3299) He stated that, at present, savings of about 11 percent and 36 percent in residential and non-residential buildings, respectively, are attainable. Future new commercial structures could be designed to require 50 percent less energy. Applying these projected savings to the New England area, he predicted that the demand for electrical energy in New England would remain virtually unchanged or would be reduced by the year 2000 even with a 40 percent growth in the residences in the service area. (Tr. 3300)

223. Commonwealth Witnesses Lee and Levy presented a report[84] addressed primarily to future relative costs of energy derived in New England from various fuels. It touched briefly on the matter of necessary growth of production facilities. A position of the report is the adoption of

[83]The Neely testimony was received at Tr. 3542. Contrary to a statement there, it was not bound into the record. It was designated as Commonwealth Exhibit 17 and was again received into the record by Board Order dated July 14, 1978.

[84]"The Economics of Nuclear Power: A New England Perspective," Energy Policy Office, Commonwealth of Massachusetts, December 1975. (following Tr. 4962)

"policies and price structures that discourage the growth in electricity demand[85] and that promote a more efficient use of both existing and new generation capacity." (at 3 following Tr. 4962)

224. Achievement of a reduced electric generation growth rate could be by reducing wastage of energy and by increasing the efficiency of its use. (Tr. 4967) A reduction of growth by these means could better the economy of New England by removing the burden to residents and industry of the high costs of unnecessary plant construction. (Tr. 5054)

225. The witnesses, nonetheless, believe the projected growth rate to be finite, pointing out that a Ford Foundation study[86] used a 3 percent annual rate, doubling consumption by the year 2000, in an extremely conservative projection. (at 7 following Tr. 4962)

226. Although Board Witness Fields, of the FPC (now FERC), did not address directly the question of electric power and energy requirements in the BECo service area, he did testify on the reliability of bulk power supplies and on the justification of the 23 percent reserve margin established by NEPOOL.[87] A FPC analysis for 1982/83 disclosed that the 23 percent margin[88] would not be maintained if the annual load growth exceeds 6 percent. (at 11 following Tr. 6080) He concluded that the NEPOOL system would be much more stable and reliable with the Unit 2 capacity available. (at 12 following Tr. 6080)

227. Applicants and Staff witnesses testified that peak-load pricing would not reduce the need for Unit 2, contending that peak-load pricing would likely increase base-load requirements thereby further justifying increased base-loaded capacity, such as by nuclear units, at the expense of installing peak-load units. They also indicated the possibility that electric energy consumption would increase as usage is shifted from peak to off-peak periods. (Applicants' Witness Guth at 6 to 8 following Tr. 2647; Staff Witness Nash at 23 following Tr. 3110)

228. Both Applicants and Staff further contend that implementation of peak-load pricing would be difficult and would require regulatory actions, and modifications of current metering systems and of other equipment to take advantage of off-peak power. (Nash at 23 following Tr. 3110; Tr. 3243; Guth at 6 to 8 following Tr. 2647; Ferguson at 32 following Tr. 2647)

[85]In this context "demand" connotes energy. (Tr. 4966)

[86]Energy Policy Project of the Ford Foundation, "*A Time to Choose: America's Energy Future,*" p. 508 (1974)

[87]The FPC analysis referred to NEPEX which performs dispatching and planning functions for NEPOOL. (Tr. 6202)

[88]Whereas the genesis of the 23 percent margin is a projected system reliability no less than a one-day outage each 10 years, the witness translated it into 0.5 hours per week (an order of magnitude greater) when applied to a span as short as one week. (Tr. 6210 *et seq.*)

229. A significant amount of evidence introduced in these proceedings on the propriety of construction of Unit 2 supports the thesis that a benefit of principal importance is the value of the Unit in replacing older, less economic, potentially polluting generating stations which in many instances consume oil to the disadvantage of this country's international economy and of the use of that natural resource in industries where it is a unique raw material.

230. The Applicants have most recently forecast, for the NEPOOL area, an annual compound growth rate in winter-season peak power of 2.7 percent. Other parties have not taken exception to this value. Assuming all planned additions to the generating system are effected on present schedule, retention of the 23 percent reserve margin, required by NEPOOL, will require operation of Unit 2 in advance of the 1993/94 winter season.

IV.B. b. Impacts of Construction

231. The Applicants have identified and described and the Staff has reviewed the environmental impacts associated with construction of the facility. (ER §§ 4 and 11; FES § 4)

i. Impact on Land Use, Terrestrial Ecology and Fresh Water Resources

232. The construction of the proposed Unit 2 will have some adverse impacts on land use and on the terrestrial biota. About 49 additional acres of the site, including areas for the water tank and meteorological tower, the construction personnel parking lot and access road, and the construction laydown and batch plant will be cleared for the construction of Unit 2. (ER § 4.11; FES § 4.1.1, following Tr. 897 as revised in Staff Exhibit 16 following Tr. 8542) In order to reduce construction impacts on certain wetlands, 47 of these 49 acres to be cleared are located south of Rocky Hill Road. Although clearing will remove the mixed-oak forest, no more than 2 percent of the mixed-oak habitat in this region will be affected. (Staff Exhibit 16 at 2, 3 following Tr. 8542) Such a loss is acceptable, particularly since it is compensated by the protection of the regional wetland resources. Further, much of the cleared area not subsequently occupied by permanent structures will be landscaped or allowed to return to its natural state. (FES at 4-7)

233. The major expected adverse terrestrial ecology effects are those associated with long-term loss of biological productivity through the removal of forest community acreage and replacement by buildings and pavement. The construction, however, is not expected to result in the elimination of any existing population of plants or animals. (FES at 4-4)

234. There are no historical, cultural, archaelogical or architectural resources which will be affected by construction of Unit No. 2. (FES § 2.3.2)

235. The existing transmission corridor from the Site, including the towers, which was established for Unit 1, will also serve Unit 2. (FES § 3.7) The installation of transmission lines will have a minimal impact on the land in the vicinity of the facility and along the transmission corridor. The Staff recommends, however, that replanting portions of the transmission corridor with a vegetative screen be a condition of the issuance of permits.

236. Water required during construction, at a maximum flow of 500 gpm, will be furnished by the Town of Plymouth and will not affect the local water supply. (FES at 4-2)

237. The construction of Unit 2 will have no effect on the quality of surface run-off water or groundwater used by others because the Unit 2 site is on the downstream edge of the basin, and all drainage from the site discharges directly to the Bay. (FES at 4-2)

ii. Impact on Cape Cod Bay

238. The construction of the proposed Unit 2 will have some adverse impacts on Cape Cod Bay water and on the aquatic biota. The major impacts on the water of the Bay were incurred when the intake and discharge channels for Unit 1 were constructed although some modification of those channels is now required together with dredging and construction of a barge unloading facility. There will result slight temporary impacts such as increased turbidity in the immediate area of construction and at nearby beaches. (ER § 4.1.4; FES at 4-3) The Applicants have estimated that, in addition to the destruction by construction of Unit 1 of 11 percent of the total area from which Irish moss is harvestable, another 3.5 percent will be destroyed during construction of Unit 2. Indications are, however, that recolonization will occur. Further, there will be some displacement of marine life from about two acres of the Bay bottom. (ER § 4.1.2.3; FES at 4-5)

239. Dredged spoils will be barged to an off-shore disposal area in accordance with U.S. Army Corps of Engineer regulations. Other surplus material removed by land-based equipment will be disposed of in an existing borrow area in accordance with specifications of the Massachusetts Departments of Natural Resources and of Public Works. (FES at 4-3)

240. Staff Witness Parsont testified on the effect of temperature on the uptake and elimination of radionuclides by aquatic organisms. Experience has shown a two- or three-fold increase in concentration factors and, hence, in radiation exposure, at elevated temperature. (These observations were

made, however, in radiation fields orders of magnitude greater than any expected in the Bay near the Pilgrim site.) Even a three-fold increase in the exposure of organisms in the Bay will not result in radiologic doses causing observable somatic effects. In general, induction of genetic effects is independent of temperature. The witness concluded that at the temperature and in the low radiation field characteristics of the proposed site no observable adverse somatic or genetic effects will be caused by radiation and by thermal-radiation interaction. ("Supplemental Testimony of M. A. Parsont Relative to Massachusetts Wildlife Foundation Contention 2(c)," Staff Exhibit 7A, identified at Tr. 6431 as Staff Exhibit 7 and received at Tr. 6432)

iii. Impact on the Community

241. Vehicular traffic and the general noise will increase due to construction. At peak construction about 1000 cars will transport workers to and from the site resulting in some traffic congestion. Noise during site preparation and construction will arise from trucks, earthmoving equipment, rock drills, pneumatic machinery, and pile-driving rigs. The noise at the nearest residence may reach 75 dB corresponding to that of a busy street. The Applicants will ameliorate these conditions by installing traffic controls, utilizing mufflers on vehicles, and minimizing truck usage of Rocky Hill Road. (ER § 4.1.1.5; FES at 4-5, 4-6)

242. Only 7 percent of the construction labor force is expected to reside in the Plymouth area and will be the cause of a small demand for housing and an increase of about 6 percent in the local school enrollment, both temporary and absorbable into existing facilities. The 160 permanent operating employees can be similarly accommodated. (ER §§ 8.2.2.1 and 8.2.2.3; FES at 5-4)

iv. Impact of Pilgrim Unit 1 on Construction Personnel

243. During construction of Unit 2 the labor force will experience an estimated radiation exposure of 100 man-rem arising from the operation of Unit 1. The main sources of this exposure are the gaseous effluents and air-scattered radiation from nitrogen-16 in the turbine. The Applicant is committed to limiting this exposure to as low as reasonably achievable.[89]

[89]Staff Exhibit 11A following Tr. 7828. This is § 4.6 of the FES filed as an addendum.

v. Proposed Measures to Mitigate Construction Impacts

244. The Applicants have proposed measures, broadly applicable to construction activities, intended to limit adverse environmental effects of the Unit 2 project. The Staff concludes from its evaluation of those anticipated measures that they, when supplemented by more specific items appearing elsewhere in this decision, will limit the impact to a practical minimum. The Applicants' proposals as stated by the Staff follow. (FES at v)

245. The Applicants will take the necessary mitigating actions (including those summarized in §§ 4.5.1 and 4.5.2 of the FES) during construction of the plant and of the associated transmission lines to avoid unnecessary adverse environmental impacts from construction.

246. Moreover, the Applicants will establish a program which will include written procedures and instructions to control all construction activities prescribed in the FES and by this decision and to provide periodic management audits to determine that all conditions are adequately implemented. The Applicants will maintain records showing compliance with all of the environment-related conditions imposed by this decision.

247. The Applicants will prepare and record an environmental evaluation before engaging in any construction activity not previously considered by the Staff. When that evaluation indicates that the activity may result in an adverse environmental impact not previously considered or in an impact considered in the FES to be less severe, the Applicants shall describe the activity in writing and shall obtain prior approval of the Director of the Office of Nuclear Reactor Regulation.

248. If unexpected harmful effects or evidence of serious damages are detected during construction, the Applicant will provide to the Staff an analysis of the problem and a plan to eliminate or significantly reduce those effects.

IV.B. c. Impacts of Operation

249. The Applicants have identified and described and the Staff has reviewed the environmental impacts associated with operation of Unit 2. (ER §§ 5, 8 and 11; FES § 5)

i. Terrestrial Impacts

250. The operation of proposed Unit 2 will have slight impact on land use. The Site is now occupied by Unit 1 and operation of both units with related facilities will permanently require about 45 acres out of a total site area of more than 500 acres. Although the major structures will be quite visible from the Bay, there will be no unusual visual impacts from the land side because the property surrounding the station is heavily wooded and is

at a lower elevation than the surrounding countryside. Availability of portions of the site to the public for recreational and educational purposes will continue. (FES § 5.1.1)

251. The impact of the operation of Unit 2 on terrestrial wildlife is expected to be negligible. (FES at 5-23)

252. Effects of operation and maintenance along the transmission corridors will be minimized by the utilization of helicopters for inspection and the retention of existing roads, built during construction, to provide access for maintenance. Brush control will be by selective application of herbicides in accord with Commonwealth regulations. (FES at 5-24)

ii. Aquatic Impact, Nonradiological

253. The operation of proposed Unit 2 will have some impact on the Bay, on terrestrial water, and on the aquatic biota. The aquatic biota of the Bay is typical of that found in north temperate climates and is more representative of a marine than an estuarine environment.

254. The flow rate of Bay water required for the once-through cooling system of both Units 1 and 2 when operating at full power is nearly 3000 cfs with return through a common discharge channel at about 22°F above the intake temperature. The only loss of water is through evaporation. Although wind and tidal patterns make a description of a discharge plume difficult, the estimated area in which the surface temperature will be 15 or more degrees Fahrenheit above ambient is one acre whereas the area of the Bay is about 365,000 acres. Correspondingly, the area of the plume encompassed by the 5°F above-ambient isotherm is 64 acres. (FES at 5-4, 5-30 *et seq.*)

255. Certain species of fish have congregated in the vicinity of the Unit 1 coolant discharge and, although no mortalities have been attributable directly to the increased temperature in that area, fish have died as a consequence of the supersaturation of nitrogen in the discharge. The ailment, called gas bubble disease, has, on two reported occasions, killed a quantity of menhaden equivalent to the order of 0.1 percent of the annual harvest from the Bay. (Staff Witness Froelich at Tr. 2194 *et seq.*; FES at 5-36; ER § 5.1.2.4)

256. Mortality of fish acclimated to warm water then suddenly exposed to a cold shock, as might result following a plant shutdown, has not been a problem at Unit 1. Even though the area of heated water will be larger with two units operating, the impact of shutting down one will be considerably reduced. The frequency of simultaneous shutdown of both Units 1 and 2 is estimated at seven per year, normally occurring during spring or fall months when ambient water temperatures are not extreme. The Applicants

must make every effort to insure that simultaneous shutdown of both units does not occur during the winter months. (ER § 5.1.2.5; FES at 5-36)

257. The National Pollutant Discharge Elimination System permit, issued to the Applicants, will require installation of a barrier near the terminal of the discharge canal to prevent entry of fish. Additional precautions are required if the barrier is not successful. (Staff Exhibit 18C at 15 following Tr. 8801)

258. Trash racks and traveling screens are provided in the cooling-water intake system to reduce the passage of fish and of debris into the condensers. Although the water speed in this screening system is about 1 fps, a value believed easily sustained by many species common to Cape Cod Bay, fish are expected to impinge on the screens. In spite of provision of a return path for fish thusly caught to return to the Bay, a loss is expected. Observations at Unit 1 over a period of one year reported collection of fish on the screens at a rate of 1.4 per hour, 60 percent being from the herring family and 27 percent smelt and silversides. Impingement with both units operating is expected to be 3.4 times greater.

259. Spores, eggs, larvae, small fish and other plankton will be entrained in the intake waters and will pass through the plant condensers. These organisms will be subject to mechanical and chemical stresses and to a thermal shock of as much as 22°F. Assuming 100 percent mortality of such organisms passing through the plant, the Staff has estimated that with both units operating at capacity up to 15 percent of the planktonic forms in a one-square-mile area of the Bay adjacent to the plant will be killed each day. (FES at 5-27)

260. The principal chemical contaminant of the water discharged into the Bay will be chlorine, from sodium hypochlorite, serving as a biocide in treatments, separately, of each half of the Unit 2 condenser. The chlorine concentration in the discharge from the canal is expected to be less than 0.2 ppm achieved by dilution with water from Unit 1 and from the half of Unit 2 not undergoing treatment. Chlorination will occur about one hour per day during two-thirds of the year. (FES at 3-28 and 5-4)

261. No evidence of expected violation of the current high quality of the local coastal waters (Class SA of the Massachusetts Water Quality Standards) was presented. Hence, operation of Unit 2 will not be cause for restriction of any water sports at nearby beaches. The high surface speed of the coolant discharge will provide a potential risk to small boats. (FES at 5-7)

262. Consideration was given to the impact of plant operation on commercial fishing, including bottom trawling for flounder, cod, haddock, etc., and harvesting lobster and Irish moss. No significant nonradiological impacts of the operation of Unit 2 on the fish and lobster are expected. In

addition to the loss of 3.5 percent of the productive area for Irish moss due to Unit 2 construction,[90] the thermal effects of operation will reduce the crop by 11 percent. (FES at 5-8) Recent fluctuations in Irish moss crops, attributable to natural causes, may obscure the predicted effects of thermal discharges. (FES at 5-40 and Tr. 10, 010)

263. The maximum annual requirement for fresh water from the Plymouth municipal water supply has been set at 30 million gallons (~ 60 gpm, average). (FES at 5-7, ER at 3-15) All drainage from the Pilgrim Site is directly into the Bay, thereby precluding any effect of the installation on potable water supplies. (FES at 5-7)

iii. Impact on the Community

264. The social and economic impacts of the operation of Unit 2 on the community are expected to be mainly beneficial. Benefits will derive from an increased tax base either as increased revenue or as a decreased tax rate. Eighty-five additional permanent personnel will be employed bringing about 100 additional children into the Plymouth schools, corresponding to a one percent increase in enrollment. No problems are anticipated in meeting the need for housing and for medical, transportation, and other municipal services. Noise from station operation is not expected to have any impact beyond the site boundary. (FES at 5-41; ER at 8-22 through 8-27)

iv. Radiological Impacts

265. Releases of radioactivity from nuclear power plants are subject to Commission regulations which, among other things, require the Applicants to "make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as reasonably achievable ... taking into account the state of technology, and the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest." [10 CFR § 20.1(c)]

266. The Cleetons introduced the direct testimony of Witnesses Tamplin (following Tr. 6959A), Bertell (following Tr. 7044) and Caldicott (following Tr. 7150). The testimony proffered by Martha Drake was ruled inadmissible on the grounds of relevance. It reported health effects near two boiling water reactors located in the mid-west and near one on the West

[90]Paragraph 238, *supra*.

Coast. The statement is included in the record as a limited appearance. (following Tr. 7138)

267. None of these witnesses addressed the impact on the Cleetons of the specific releases of radioactive materials from Unit 2 and no evidence was presented to show that the Cleetons would be at any greater risk from the doses of radiation resulting from the routine operation of Unit 2 than are other similarly situated members of the public. None disputed the validity of the Staff's radiation dose estimates. (Tr. 7007, 7118, 7119)

268. Witness Tamplin addressed generally the assessment of risk from exposure to low-level radiation[91] with particular reference to the assessment presented in the BEIR Report.[92] His major point was that "the estimates of the biological effects of radiation that are in current use most likely significantly underestimate both the somatic and genetic effects on both populations and individuals." (at 8 following Tr. 6959A) Reference was made to the work of Bross[93] who has shown a variation of several orders of magnitude in the sensitivity of individuals to radiation. The Witness opined that risk to individuals may be as much as 1000 times the upper limit of the range given in the BEIR Report. (Tr. 6965) To make this judgment, however, requires information on the individual's medical history and genetic background and those of members of his family. He further has no information upon which to base a judgment whether the Cleeton family could be among the group of individuals whose relative risk to radiation effects could be 1000 times worse than the average of the population. (Tr. 6969)

269. Cleeton Witness Bertell testified that the estimated discharges of radioactivity from Unit 2 have not been adequately tested against reality, that they are most probably not conservative, and that uncertainties involved in present monitoring are such that only a rough approximation of the possible radiation exposure to an individual is possible. (at 1 following Tr. 7044) She, however, had no specific information about Unit 2 other than its size and that it was a light-water reactor. When questioned about the estimated discharges from Unit 2 and the proposed monitoring program, the witness referred to a nation-wide "release" of 0.003 mrem from the entire nuclear industry presented before the EPA in 1976 by Roger Madsen, not otherwise identified, who also reported that radiation

[91]In this testimony "low-level" denotes radiation of intensity at or near that of natural background. (Tr. 6995)

[92]NAS BEIR Report, Report to the National Academy of Sciences, National Research Council Committee on the Biological Effects of Ionizing Radiation, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Washington, D.C., November, 1972. This report was revised in 1980.

[93]Bross, Irwin D. J., "Leukemia from Low-Level Radiation," *New England Jour. of Medicine*, Vol. 287, No. 3, 20 July 1972, pp. 107-110.

monitoring equipment was not available for nuclear plants. (Tr. 7052, 7055) She had no knowledge of the monitoring plan proposed for Unit 2 (Tr. 7056, 7059) and had not examined the estimates made by Staff and Applicants of the expected emissions from it nor had she read the PSAR, the SER or other documents supplied by the Applicants and the Staff describing radiological discharges and dose rates associated with the proposed Unit 2. (Tr. 7055, 7094, 7104) She believed the calculations by Staff Witness Gotchy were correct though based on outdated information in the BEIR Report. Further, in the Gotchy testimonies, only mortality was considered, not total health effects. (Tr. 7118-19) She stated that any discharge above zero would be unacceptable. (Tr. 7107)

270. Witness Bertell stated that the purpose of her testimony was to "... set forth some of the scientific evidence for my conclusion that the Cleeton family, and particularly Mrs. Cleeton and three grandchildren, ... would be exposed to an unreasonable risk to health and safety if the proposed Pilgrim 2 were to be constructed and operated." She referred to her research that has revealed classes of children with as much as 50 times the average susceptibility to leukemia and adults with demonstrated immuno-incompetency or already damaged genetic mechanisms making them several times more susceptible to further damage than are healthy persons. (at 3 following Tr. 7044) She concluded that Mrs. Cleeton's history of arrested tuberculosis and of cancer in her family places her and her family at a greater relative risk than if there were no such history. (Tr. 7063) Without quantification the additional risk from Unit 2 was stated to be unreasonable. (Tr. 7065) The witness had not determined whether the Cleeton grandchildren, now (1977) of age less than ten, were in the more-susceptible-than-average category. (Tr. 7084)

271. Witness Bertell criticized the proposed Unit 2 monitoring program, stating that it was inadequate to give early warning of a deterioration of human health (at 3 following Tr. 7044) and, in oral testimony, stated that the failure to measure health effects is the result of an outmoded public health system where morbidity is not included and the "public health measuring devices are not to the level of sophistication to handle the pollution problems from radiation, PCB's [polychlorinated biphenyls] and all the rest of the industrial pollution which is being put into our environment and our food." (Tr. 7074-75) She further testified that this is a generic problem and not peculiar to radiation or to Unit 2 and that the monitoring she proposes is not being done for any industrial pollutant. (Tr. 7076)

272. Witness Bertell implied but did not provide any probative evidence that Mrs. Cleeton or any member of her family, including the grandchildren, would be at greater-than-average risk of injury due to radiation from

proposed Unit 2. It is her position that no amount of radiation is acceptable and that exposure to any radiation would constitute an unreasonable risk to the health and safety of the Cleeton family. (Tr. 7107)

273. Cleeton Witness Caldicott testified that " ... Mrs. Cleeton with her medical history, is at great risk if exposed to any additional radiation" and "The Cleeton grandchildren, like all children, are more susceptible than adults to damage from radiation." (at 2 following Tr. 7150, also Tr. 7152) She did not, however, provide any probative evidence to demonstrate that the estimated incremental radiation from Unit 2 would have a synergistic effect over and above what might be predicted from the linear non-threshold theory espoused in the 1972 BEIR Report. She identified several inherited diseases which she alleged make persons more susceptible than others to the effects of radiation and identified groups with alleged radiation sensitivity including fetuses, infants and young children, children with allergies, and fair-skinned people, stressing that it is the young who are extremely sensitive. (Tr. 7181-82) The witness had very limited knowledge of the proposed radioactive releases and radiation dose rates associated with Unit 2 and could provide no estimate of the magnitude of the risk except to say that any radiation additional to the natural background is unacceptable. (Tr. 7153, 7180, and at 2 following Tr. 7150)

274. Applicants' Witnesses Larson and Cehn testified on the release of radioactive materials from liquid and gaseous effluents from the proposed Unit 2. Applicants' assessment of the liquid and gaseous effluents during normal operation of Unit 2, determined in accordance with 10 CFR 50 Appendix I, gives annual radiologic whole-body doses to the "maximum exposed" individual at the site boundary of 0.004 and 0.282 mrem, respectively, and are no more than 6 percent of the exposure limits established by Appendix I. The single-organ doses are correspondingly lower than the Appendix I limits. The gaseous effluent doses assume 0.1 percent of the fuel pins are defective. The predicted total annual dose to the population within 50 miles of the reactor resulting from both liquid and gaseous effluents is 1.9 person-rem and may be compared to 720,000 person-rem/yr arising in that area from natural radiation. (at 9 *et seq.* following Tr. 7352 as revised by updated Supplemental Testimony, also following Tr. 7352)

275. Applicants' Witness Larson testified that individuals residing 40 miles west of the Pilgrim site, as do the Cleetons, would experience radiation exposures estimated to be three orders of magnitude less than those predicted at the site boundary. Consumption of food originating near Unit 2 and utilization of Cape Cod recreational facilities could increase this factor to two orders of magnitude or the dose to 0.003 mrem/yr. (at 7 *et seq.* following Tr. 7352)

276. Applicants' Witness Cehn estimated 125 deaths per year within the 50-mile radius population attributable to background radiation exposure, 720,000 person-rem/yr. Superposition of the exposure expected from Unit 2 operaton, 1.9 person-rem/yr, would, by the linear dose model of the 1972 BEIR Report, cause an additional 0.003 cancer deaths per year, an increase of 1 in 4 x 10^5 events, due to radiation and of 1 in 3.8 x 10^7 cancer deaths from all causes. (at 16 following Tr. 7352 as revised)

277. Although the analyses by the Staff evaluating the health effects of routine radioactive emissions during operation of Unit 2 utilized models and assumptions different from those of the Applicants, the results show variances considered to be not unreasonable. (Tr. 7432, 7814) The Applicants' values of the annual dose to the "maximum exposed" individual and of the 50-mile population dose, 0.28 mrem and 1.9 person-rem, compare to those of the Staff, 3.6 mrem (at 3 following Tr. 7654) and 1.81 person-rem. (Tr. 7819; at 3, Staff Exhibit 8A following Tr. 7820) The Staff recognized that dose calculations may be uncertain by as much as an order of magnitude because of inadequacies in both model and input data. The practice, however, is to achieve conservatism in the result through introduction of data from the high-risk limits of the uncertainties and through "worst-case" assumptions. (Tr. 6625) Further, the Staff integrated the expected dose over the projected 30-year lifetime of Unit 2 and, additionally, extended the consideraton for 50 more years to assure taking into account the whole lifetime exposure experience of an infant at the time of plant startup. (Tr. 6535) There is presently before the National Committee on Radiaton Protection and Measurement a recommendation reducing the risk from low-dose rate exposure by a factor of five from that utilized by the Staff. (Tr. 6625)

278. In the evaluation of the annual dose to the "maximum-exposed" individual, Staff Witness Gotchy assumed a family residing near the site boundary, procuring all its food from nearby land and water and utilizing an adjacent beach for recreation. The source term in the evaluation was taken from the testimony of Staff Witness Weller. (Table 2 following Tr. 7659) The resulting whole-body dose for a child is 3.6 mrem/yr, shown to be greater than that for any individual organ and greater than that for any other age group. Accordingly, this value was used in subsequent dose evaluations.

279. From this exposure and the analysis proposed in the 1972 BEIR Report, the added lifetime risk, to this maximum exposed individual, of cancer mortality because of the operation of Unit 2 is 2.1 x 10^{-5}, or a mortality risk of 1 in 4.7 x 10^4. The risk 40 miles distant is orders of magnitude less. (at 6 following Tr. 7654) Extending the evaluations to the population exposure within 50 miles of the site resulted in the following.

The dose from the combined liquid and gaseous effluents is calculated to be 1.81 person-rem/yr and to cause 7.4×10^{-3} cancer deaths over the 30-year life of the plant. The incremental life-time risk of mortality from radiogenic cancer is 1.8×10^{-9} or 1 chance in 550 million. (Staff Exhibt 8A following Tr. 7820 correcting both Tr. 6721-2 and Gotchy affidavit dated April 5, 1977)

280. For comparison, statistics[94] describing common lifetime risks are tabulated:

Cause of Death	Individual Lifetime Risk		
Cardiovascular disease	0.526	or 1 in	1.9
Cancer	0.178	or 1 in	5.6
Influenza and Pneumonia	0.0317	or 1 in	32
Motor Vechicle Accidents[95]	0.281	or 1 in	35
Suicide	0.0127	or 1 in	79
Homicide	0.0104	or 1 in	96
Peptic Ulcer	0.0039	or 1 in	256
Drowning	0.0026	or 1 in	380
Poison	0.0016	or 1 in	630
Air Travel	0.00062	or 1 in	1,600
Electrocution	0.0004	or 1 in	2,500
Lightning	0.000056	or 1 in	18,000
Tornados	0.000041	or 1 in	24,000
Hurricanes	0.000032	or 1 in	32,000

281. The Staff assessed the radiological impact of the operation of Unit 2 by determining radiaton dose commitments to various segments of the population. The analysis begins with the anticipated liquid and gaseous

[94]These 1973 statistics were taken from "Statistical Abstract of the United States - 1975", U.S. Department of Commerce, Bureau of the Census.
[95]This value corresponds to about 4.3×10^{-8} deaths per driving mile.

effluents as source terms,[96] their migration along various pathways and culminate in expected doses. Doses are, in some instances, translated into health effects and, ultimately, into a segment of the cost-benefit analysis.

282. The gaseous effluents determined by the Staff (Table 2 following Tr. 7659) were compared with the effluents calculated independently by the Applicants using a different model and different input data. A comparison of the two sets of results by Witness Weller (Tr. 7766) showed those of the Staff to be generally higher.[97]

283. The annual radiation doses averaged over the projected 30-year operating life of Unit 2 to a "maximum individual"[98] arising from liquid and gaseous radioactive effluents were determined by the Staff to be less than the design objective doses stated in 10 CFR Part 50, Appendix I, by amounts ranging from 13 percent to more than 99 percent. Accordingly, the Staff concluded that the maximum individual would be subjected to an annual radiation dose of not more than 10 mrad from beta radiation, and 15 mrem from the iodines and particulate matter.

284. As a basis for judgment on a requirement for augments to the gaseous radwaste system, the Staff determined the annual population dose within a 50-mile radius of the reactor to be 1.8 man-rem total body and 3.4 man-rem thyroid. The annualized cost of the most effective augment considered is $11,500 which exceeds the cost-assessment value of $3,400 assigned to the thyroid dose in 10 CFR 50, Appendix I, Section II.D.

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[96]The source terms initially determined by the Staff appeared as Tables 3.4 and 3.5 of FES. (at 3-20, 3-26) Models and parameters provided in "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)", NUREG-0017 (April 1976) required modification of the source terms to conform to Appendix I of 10 CFR Part 50. Additionally, changes in the design of the gaseous radwaste system were proposed by the Applicants. Tables 1 and 2 of testimony by Staff Witnesses Weller and Gotchy (following Tr. 6482) resulted. This version reflects in the testimony of Staff Witness Gotchy. (following Tr. 6494) Still another revision was necessitated by a subsequent design change in March 1977. The latest results appear in testimony of Witnesses Weller and Gotchy (following Tr. 7659), where the updated source term appears as Table 2, and in the testimony of Witness Gotchy (following Tr. 7654). This evaluation is concluded and summarized at 11-1 *et seq.* (Staff Exhibit 21 following Tr. 8921. See also Tr. 9083.)

[97]In the Board's view this comparison may not be strictly proper. At the time of the testimony, the Staff's values included the March 1977 changes in the design of the gaseous radwaste system while the Applicants' values were taken from PSAR Table 11.3-9, Amendment 25, June 15, 1976 and could not reflect the March 1977 design change. Amendment 37 of the PSAR, issued August 1, 1977 (Applicants' Exhibit 24 admitted by Board Order dated December 16, 1980), included a revision of Table 11.3-9 listing source term components more properly comparable to the Staff's. Whereas between the latter there is general agreement, no trend is established.

[98]A "maximum individual" is defined as one occupying any location, near ground level in an unrestricted area, where the expected radiation exposure is maximum. (Applicants' Exhibit 24 admitted by Board Order December 16, 1980)

285. Translation by Witness Gotchy of anticipated exposures to health effects resulted in a conclusion that an individual living for 30 years at the Unit 2 site boundary, subsisting entirely on locally produced foods, and engaging in recreational activities nearby, would encounter an incremental risk of one in 47,000 of dying of cancer induced by radiation from Unit 2. (at 6 following Tr. 7654)

286. This probability of a fatality from Unit 2 radiation-induced cancer was extended to the population within an area of 50-mile radius centered at the reactor. The result is an incremental risk of one chance in 550×10^6 of such a fatality among nearly 6×10^6 residents over the 30-year plant life. (at 5 following Tr. 7820)

287. The transport of nuclear fuel to an electric power producing plant and of radioactive materials, including the fuel after use, from the plant is an operation common to all nuclear-power reactor installations and its anticipated impact on the environment is the subject of a regulation.[99]

288. The Applicants, through Witness Rosen, affirmed that the planned design and operation of Unit 2 are within the requirements and limitations established as requisites[100] for application of the regulation, and that all shipments of nuclear materials would conform to all applicable regulations.[101] (at 5 following Tr. 3651, Tr. 3657)

289. The Staff presented Witness Barker who sponsored several AEC/NRC documents[102] containing information and conclusions on which the referenced NRC regulation is based. The testimony and the response by the witness to cross-examination present the conclusions of WASH-1238 on the risk of radiological exposure due to accidents during transportation. These conclusions derive from specifications of the design of containers and the quality to be assured in their construction, of permissible radiation at the outer surface of the containers arising from their contents, of permissible quantities of contaminating substances on the outer surface, of

[99]10 CFR 51.20(g), particularly Summary Table S-4.

[100]10 CFR 51.20(g)(2).

[101]Packaging standards and criteria are found in the regulations of the Nuclear Regulatory Commission (10 CFR Part 71) and the regulations of the U.S. Department of Transportation (49 CFR Parts 171 through 179).

[102]The documents sponsored by Witness Barker are a) an excerpt from the *Federal Register*, Volume 40, pp. 1005-1009 (Staff Exhibit 1); b) "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," WASH-1238, December 1972 (Staff Exhibit 2); c) "Supplement One to WASH-1238," NUREG-75/038 (Staff Exhibit 3). These exhibits together with Witness Barker's testimony were received at Tr. 2537. The testimony was bound following Tr. 2536 in some but not all copies of the transcript. As a remedy of this omission the testimony was again received as Staff Exhibit 23 by Board Order dated July 14, 1978.

necessary and informative labels, and of the monitoring necessary to establish compliance with the regulations. (following Tr. 2536; Tr. 2552 *et seq.*)

290. At the request of the Board, the Staff presented as evidence the updated findings of an ongoing research program investigating the integrity of used-fuel shipping containers under severe potential-accident conditions. In the tests the truck, on which used-fuel containers loaded with unirradiated fuel were mounted, was subjected to collision with another truck or with a railroad locomotive in a simulated grade-crossing mishap. In an additional test a container impacted hard soil at speeds up to 250 mph equivalent to free fall through a distance of 2000 feet. In these tests the containment provided by the containers was either not breached or a trivial quantity of its liquid contents was lost through a leaking gasket. (Staff Witness Hodge at 2 following Tr. 8459) The results of these tests show a massive rupture of a spent-fuel cask to be essentially impossible.

v. Impact of the Uranium Fuel Cycle

291. Tables S-3 and S-4 in 10 CFR 51.20(e) and (g), respectively, summarize the environmental effects of the uranium fuel cycle including the impact of transportation of radioactive materials to and from the plant.[103]

292. The issue of radon-222 releases and effects was included in this proceeding by the Board's adoption (Tr. 9127) of the record of the Perkins' proceeding and the findings of the Perkins' licensing board.[104]

293. Viewing radon discharges of mining and milling operations as causing fractional increases in natural background radiation, "the increase ... is so small compared with background and so small in comparison with the fluctuations in background as to be completely undetectable. Under such a circumstance, the impact cannot be significant." (from finding of fact 51, LBP-78-25, 8 NRC 100; see fn 103 *supra.*)

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[103] "Table S-3 does not include the health effects from the effluents described in the Table, or estimates of the releases of radon-222 from the uranium fuel cycle or estimates of technetium-99 releases from waste management or reprocessing activities." Excerpt from footnote 1, Table S-3.

[104] Partial Initial Decision, *Environmental Consequences of the Uranium Fuel Cycle,* (Perkins Nuclear Station, Units 1, 2, and 3) LBP-78-25, 8 NRC 87 (1978). As pointed out by the Appeal Board, the use of Perkins as a "lead case" in the generic radon issue would result in large savings in time and effort yet not foreclose further pursuit of the issue by any litigants who might believe it warranted." Philadelphia Electric Company *et al.* ALAB-480, 7 NRC 796 (1978).

294. The health effects of the effluents described in Table S-3 are considered to have insignificant effect on the overall cost-benefit balancing of Unit 2.

vi. Availability of Fuel and Waste Disposal Facilities

295. Applicants' and Staff witnesses addressed the following topics: (1) the availability of uranium at an acceptable cost as a resource over an interval corresponding to the projected life-time of Unit 2; (2) the effect on that availability of recycling the uranium residue in used fuel elements into a supply for new elements; (3) the monetary gain to be derived from that recycle; (4) the monetary and environmental costs of prolonged used fuel storage at the Pilgrim Site including its indirect consequences on transportation effects.

296. The testimony of Applicants' Witness Stoller (at 46 *et seq.* following Tr. 955) based on early information concluded that U.S. uranium resources were adequate to fulfill the projected needs of 223 electric generating plants then projected. This witness later testified on the matter of utilizing uranium and plutonium obtained by recycle of used reactor fuel as an energy resource. The uranium-235 content of the uranium in used fuel, about 0.9 percent, enhances its value over that of natural (0.7 percent uranium-235) uranium as a raw material about 25 percent. That is, use of one part of recycled uranium would replace 1.25 parts of virgin material. Use of the salvable plutonium from recycle would further reduce the demand for uranium as fuel by about 20 percent. (at 3 *et seq.* following Tr. 4692)

297. Staff Witness Nash testified that 1.5 million tons of U_3O_8would be required to fuel 236 nuclear power reactors now operating, under construction, and planned, comprising a capacity of 242,000 MWe, throughout their projected 30-year lifetime. This estimate is based on a 0.3 percent uranium-235 content of uranium tails from gaseous diffusion separation plants and on the assumption of no light-water reactor fuel recycle. (at 3 following Tr. 4853) The resource available at acceptable costs to fill this need, as determined by the U.S. Energy Research and Development Administration, was reported by Staff Witnesses Nash and Fisher to be 3.0 million tons of U_3O_8 including 1.1 million tons having possible potential. (at 7 following Tr. 8304)

298. Commonwealth Witness Lee foresaw no shortage of fuel required for the operation of Unit 2. (Tr. 5030)

299. One of the limits of options conceived for the disposal of used fuel is permanent storage of the intact fuel elements, under the jurisdiction of U.S. Government, with no recovery of the value of the constituents of the

materials contained. This scheme is called the "throwaway" fuel cycle and will, of course, entail a cost to the utility. The Applicants' estimate that this alternative will increase the 1988 cost of nuclear-produced energy by an amount equivalent to 1 mill/kWh or 3 percent of the generating cost. (Applicants' Witnesses Stoller and Muckerheide at 5, 6, 13 following Tr. 4692 and Seery at 22 following Tr. 8207)

300. Staff Witnesses Nash and Fisher place this fuel cycle cost differential between recycle and no recycle at 2.3 mills/kWh. (at 21 following Tr. 8304)

301. The absence of used-fuel reprocessing, *per se,* and of regulatory decision on any alternate ultimate "backend" of the fuel cycle will require temporary accommodation of fuel elements being continually discharged from operating reactors. A typical capacity of the used-fuel storage facility in a reactor complex is one and one-third cores established by a discharge schedule of one-third core per year and a requirement for the capability to receive the core-in-operation were a need for complete unloading to arise. Staff Witness Miller described means of increasing the capacity of existing storage pools by the introduction of neutron-absorbing chemical elements to permit more compact arrays of the fuel elements.[105] The neutron absorber may be included in the structural supports or may be dissolved in the cooling-shielding water. The witness predicted that the capacity of the Unit 2 pool could thereby be doubled to provide adequate space for operation for about 10 years.[106] Additional storage capacity could be provided by newly constructed facilities both at the reactor site or elsewhere. (following Tr. 4778)

302. Applicants' Witnesses Stoller and Muckerheide estimate the effective cost of the "high-density" storage racks to be 0.6 mill/kWh. Their cost, however, is included in the current estimated capital investment. (at 13 following Tr. 4692)

303. The potential impact on the environment of increased and prolonged storage of used fuel at the Pilgrim site through greater exposure of the public was analyzed by Staff Witness Parsont who concluded that an insignficant increment to an already low exposure would occur. Radiation from the pool is comprised of (a) a component from the used fuel and the irradiated structural materials of the elements, and (b) a component from the cooling water arising from activity induced in water itself and in dissolved impurities and from radioactive materials that have leaked from defective fuel tubes. The design capacity of the Unit 2 storage pool is 300

[105]The structures for such configurations are sometimes referred to as "high-density racks."

[106]The availability of the one-core reserve-storage capability during these 10 years is not clear from the testimony. (at 3 following Tr. 4778).

fuel assemblies. The minimum depth of the cooling water above the radioactive portion of the assemblies is 26 feet. Calculations by validated methods have shown the dose rate 3 feet above the surface of the pool to be 10^{-7} mrem/hr from radiation arising in the fuel. By similar means the dose rate at that elevation from the activity in the water is 1.8 mrem/hr. The corresponding dose rate to an individual located 1900 feet above the pool from both sources would be 0.02 mrem/hr.[107] An increase in the inventory of used fuel by several-fold necessary to accommodate the discharge from the reactor over an extended period will increase this impact proportionally at most. (Staff Witness Parsont following Tr. 4906)

304. Applicants' Witness Muckerheide arrived at the same conclusion through considerations of the quantities of thermal energy to be dissipated by the pool coolant as a function of the inventory and its age. There is additional conservatism in the result because the spectrum of the emissions from "old" fuel elements softens with age. Further, the retention of used fuel at the Pilgrim site reduces still further the small radiological environmental effects associated with normal transport. (at 9 through 11 following Tr. 4692)

IV.B. d. Consideration of Alternatives

i. Alternate Energy Sources

305. The witnesses collectively addressed the utilization of fossil and solid-waste fuels, both directly or after conversion, in conventional steam-electric generating plants, and the reliance upon solar, wind, and oceanic temperature-differential energies.

306. Plant operational reliability is an important consideration in determining both absolute and relative costs of power and energy derived from various sources. Reliability was discussed by a number of witnesses. The quantitative measure addressed was the capacity factor defined as the ratio of the actual energy output to the expected net energy output. The availability factor is the ratio of the maximum energy that could have been supplied to the expected net energy output. The capacity factor and the availability factor are equal for a fully loaded base-load generating station.

[107]The residence nearest the proposed storage is 1900 ft distant. (ER at 2-66) Calculation at a vertical distance of 1900 ft is conservative through neglect of radiation shielding effected by structures and other objects encountered in a 1900-ft ground level path. Vertically emitted radiation must be air-scattered to ground level. (Staff Witness Parsont at 2, 3 following Tr. 4906)

307. Applicants' and Staff's testimony[108] on nuclear and coal as alternate sources of energy was first presented to the Board in October 1975. Rapidly changing economic conditions throughout the United States in the ensuing period have made moot those early cost-benefit analyses and many were updated[109] in mid-1977. Accordingly, the Board has attached greater weight to the later testimony.

308. The Commonwealth presented the testimony of three panels of witnesses relating to the economics of electric power generation from nuclear and coal energy. Witnesses Lee and Levy sponsored a report[110] which discusses required generating capacity and the narrow choice between nuclear and coal sources. (bound following Tr. 4962)

309. Commonwealth Witnesses MacDonald and Madden (following Tr. 5690, Tr. 6220) and Boxer (following Tr. 8587) addressed primarily the expected capacity factor of future nuclear generating stations. Both studies derived from a statistical analysis of past performances of generating plants. The former compared anticipated operations of coal and nuclear plants and the latter presented the results of an historical study of nuclear power plant performance.

310. The decision of NEPOOL on the need for power in the immediate future is to expand the base-load generation capacity through the addition of nuclear fueled plants to achieve an approximate 55 percent nuclear component of the total NEPOOL capacity. Although NEPOOL expects its nuclear units to operate at about 70 percent capacity factor,[111] its study shows an economic benefit of nuclear at an operation at even 50 percent capacity factor.[112] Unit 2 is considered to be an integral part of the NEPOOL plan.

311. Applicants' Witness Maroni (at 6 following Tr. 8207) presented comparative generating costs in 1988 by nuclear and coal units. The nuclear-energy costs apply to the 1150 MW Unit 2 plant then expected to be in service for four years and the coal-energy costs apply to two 590 MW

[108]See, for example, the testimony of Applicants' Panel comprised of Messrs. Sten, Weiner, Butler, Maroni, Hechling, Stoller, Godley, Smith, Gerber, Irving and White following Tr. 955. Also Staff's Final Environmental Statement following Tr. 897 and testimony of Witness Vetrano following Tr. 1409.

[109]See Applicants' panel comprised of Messrs. Maroni, Madsen, Leery, Dunlap and Gibbons following Tr. 8207. Also testimony of Staff Witnesses Nash and Fischer following Tr. 8304.

[110]"*The Economics of Nuclear Power: A New England Perspective,*" Energy Policy Office, Commonwealth of Massachusetts (December 1975).

[111]The 30 percent operating penalty in the projected capacity factor results from both scheduled and unexpected down times. For example, a normal 6-week refueling reduces the capacity factor by 12 percent. Additional reductions stem from regulatory actions, equipment malfunctions, and economic dispatch of energy. (Applicants' Witness Henchling at Tr. 1350)

[112]Applicants' Witnesses Sten at 18 following Tr. 955; Maroni at Tr. 1330; Sten at 109 following Tr. 2647; Bigelow at 8 following Tr. 7929.

units expected to have come on line in 1984 and 1985. The nuclear projections are by BECo and its consultants; the coal-plant capital costs are based on a 1977 Bechtel Power Corporation study prepared for the Electric Power Research Institute.[113] The estimated coal-fuel costs were prepared by Boston Edison Company consultants.

312. The projected total generating costs are 50.2 and 68.1 mills/kWh, the 17.9 mills/kWh differential favoring the nuclear plant.[114] A 70 percent capacity factor was applicable in both instances. Further capital investment, comprising 72 percent of the nuclear plant generating cost, is no longer subject to escalation after commencement of operation. The nuclear-coal differential may, accordingly, increase with time.

313. For comparison with the estimates of the Applicants the bases selected by the Staff in recent testimony (Staff Witness Nash at 25 following Tr. 8304) are, as for the Applicants' values, a nominal generating capacity of 1000 MW, no recycle of used nuclear fuel, no requirement for exceptional nuclear-plant site preparation on account of seismic requirements, and a 70 percent capacity factor of both nuclear and coal generating stations. On these bases the projected 1984 electro-nuclear energy cost is 47.3 mills/kWh and that from a coal-fueled plant is 63.6 mills/kWh representing a differential of 16.3 mills/kWh favorable to nuclear comparable to the Applicants' value of 17.9.

314. The estimates of the cost of generating electrical energy is strongly dependent upon the capacity factor expected. The Applicants based the 1984 cost on a value of 70 percent for both the nuclear and coal plants. The Commonwealth challenged that basis through three panels of witnesses as being a typical.

315. Commonwealth Witnesses Lee and Levy testified that nuclear and coal electric power plants will be competitive in the 1980's and justify their conclusion with qualitative statements of expectations at selected capacity factors for both types of plants. (at 38 following Tr. 4962) No numerical cost estimates, or of their construction, are given. Coal is said to be advantageous when coal capacity factors are large (75 percent) and nuclear capacity factors are small (50 percent), and vice versa. These witnesses pointed out that coal capacity factors, in recent times, have become 60 percent and less as plant size has increased. The report concludes that a

[113]Bechtel Power Corp. "Coal-Fired Plant Capital Cost Estimates," EPRI-AF-342, January 1977. (following Tr. 8207) The basic conditions of this report were adjusted to those unique to coal-fired plants proposed by Boston Edison for construction in the south-of-Boston area.

[114]The estimated 50.2 mills/kWh is based on a "throwaway" fuel cycle, i.e., the residual uranium-235 and the byproduct plutonium are not recovered from used fuel, rather the fuel is put into the custody of the Government at a prescribed cost to the owners, the utility, which is included in the estimate.

nuclear choice should be made by a utility provided a "65 or 70 percent" capacity can be realized.

316. Commonwealth Witnesses MacDonald and Madden (following Tr. 5690) offered the results of a statistical analysis purporting to show (1) that an increase in the output of a generating unit, either coal or nuclear, results in a decrease in the cumulative capacity factor; and (2) that, with increasing age, the capacity factor of a nuclear unit decreases while that of a coal plant increases.

317. The study was by the method of regression analysis with the cumulative capacity factor as the dependent variable and the duration of plant operation and its design power as the independent variables. Operational data from 28 nuclear plants and 31 coal plants were analyzed. The design power of the nuclear units, which had operated up to 15 years, ranged from 175 to 1085 MW. The coal-fired plant had operated up to 17 years and ranged in power from 114 to 1150 MW.

318. During cross-examination of these witnesses (Tr. 6220 *et seq.*) the data and results originally presented were, in many instances, altered severely.[115] The results of the analysis were supplemented by addition of confidence intervals at the 95 percent level.[116] The confidence level on values of the expected cumulative capacity factor are observed to span as much as six orders of magnitude.[117]

319. The witnesses calculated[118] expected generating costs for both nuclear and coal units utilizing the above capacity factors. Considering the many combinations of assumed input parameters, no dramatic differences between the two types of fuel are discernible. No confidence limits are stated. Also, no consideration of societal costs of coal operations, such as black-lung disease and fatalities arising from accident, were included. (Tr. 6325)

320. Witness MacDonald seemed not to value highly some of the results for, in response to a query about the initial absence of an assignment of standard errors, he stated "Basically because the number of data points is small, the statistics are not good. I refer to the large uncertainty associated

[115]See, for example, Applicants' Exhibits 11 and 12, identified at Tr. 6226 and 6228. These exhibits were admitted at Tr. 6329.

[116]These data were supplied by the Commonwealth to the Applicants under date of July 26, 1976 and were received and bound into the record on July 1, 1977 following Tr. 8789.

[117]For example, the expected cumulative capacity factor of a 600 MW nuclear plant after 5 years operation is reported by Witnesses MacDonald and Madden to be 72.8 percent within a 95 percent confidence range of 27600 percent to 0.0510 percent. For a 1150 MW plant similar to Unit 2 the expected capacity factor after 15 years operation is 23.6 percent within the range 121.6 percent to 3.7 percent. For a 1200 MW coal-fired plant after 15 years the factor is 61.8 within the range of 159.3 to 23.7 percent. (Tables 3B, 3C and 4b following Tr. 8798)

[118]Table 5 of "Statistical Analysis of the Capacity Factors of Base Load Nuclear and Coal Plants" following Tr. 5690 as amended by Applicants' Exhibit 12 following Tr. 6329.

with the data points, and what I was trying to demonstrate is that given the data as they were, this is the kind of answers that you come up with. I'm not making any supposition that these have - are highly significant in a strict sense, statistical sense." (Tr. 6293 *et seq.*)

321. The Commonwealth added further to its rebuttal of the Applicants' expected 70 percent capacity factor through the testimony of Witness Boxer (following Tr. 8587) which statistically predicted the capacity factor of Unit 2 in 1988 to be in the range 17.63 to 77.87 percent with an expected value equal to 47.75 percent. The limits are at a 95 percent confidence level.[119] This analysis was also by a multiple regression method with plant age and size as the independent variables.

322. The witness applied the analysis which predicted a Unit 2 capacity factor of 47.75 percent to operating pressurized water reactors supplied by CE. The analysis included empirical constants derived from the history of all operating pressurized water reactors. The difference between statistically predicted capacity factors and those actually observed ranged from -33 to + 20 percent. (Tr. 8614)

323. In the analysis by Witness Boxer only 20 percent of the variability is accounted for by plant age and size, the independent variables studied. (Tr. 8693) Correspondingly, 80 percent of the variability is unexplained or unexplainable. Were these unknowns introduced, their effects could influence the expected capacity factor either way.

324. Although the Staff originally utilized a capacity factor in the range 60 to 80 percent in determining benefits from the proposed Unit 2 plant (FES Table 10.1 at 10-6), Staff Witness Nash proposed a value between 55 and 65 percent near an estimated break point below which the total production cost from fossil fuel is more favorable than from a comparable nuclear-fueled plant. This value of the capacity factor was derived from the recorded 1964-1973 experience of the operation in the United States of nearly 900 fossil plants (capacity factor 68.9 percent) and of 20 nuclear plants (factor 64.2 percent). In 1974, 41 nuclear plants operated at a capacity of 57.2 percent. (at 27 following Tr. 3110)

325. Methods of analyzing and evaluating the environmental and societal impacts, including those on man, of the coal-fuel cycle have not matured to the degree characteristic of nuclear effects. Nonetheless, the Staff attempted a comparative evaluation of the health effects of nuclear- and coal-fuel cycles. The absence of data and the shortcomings of its interpretation introduce uncertainties of as much as two orders or magnitude in these conclusions. The results reported are expressed as the

[119]These final results were transmitted to the Board on August 1, 1977 and were accepted into the record as Commonwealth Exhibit 16 by Board Order dated July 14, 1978.

annual mortality within the population in the 1980's attributable to the respective fuels of a 800 MW generating station.[120] The estimated mortality as a consequence of the operation of a coal-fuel system is 30 to 240 times greater than that estimated from the operation of an all-nuclear cycle. (Gotchy, Table 1, following Tr. 8358)

326. Staff Witness Gotchy testified to the relative impacts on the environment of the nuclear and coal cycles with emphasis on the developing recognition of the effects of pollutants from coal, and the attendant uncertainties were energy requirements from that source to increase at the rate anticipated assuming that gas, oil, and nuclear fuels do not meet the demands of the future. On the other hand, the hazard potential of the nuclear cycle has been recognized since its inception and very conservative estimates of its effects have been assigned. Major pollutants from the coal cycle demanding study and evaluation of their environmental effects include:

(a) particulates of respirable size containing metals;
(b) hydrocarbons of carcinogenic character;
(c) oxides of sulfur and nitrogen;
(d) ozone, radon, carbon monoxide and dioxides and other gases;
(e) carbon dioxide[121] in its special consideration as an increasing atmospheric constituent affecting the transmission of radiant energy, the "greenhouse" effect;
(f) acids and acidic solutions as rainout and as discharges from coal mines;
(g) fly-ash and other solid products of combustion, including the SO_2 absorbers. (Gotchy at 3 and 6 to 11 following Tr. 8358)

327. The annual requirement of a fully operating 1180 MWe electric generating plant will be about 4 million tons of coal, 16 million barrels of fuel oil, or 40 tons of nuclear fuel. The annual particulate and gaseous effluents from the three plants are, respectively, 1.4×10^5 tons, 3×10^4 tons, and a negligible amount. (FES at 9-7)

328. In a discussion of the removal and disposition of products of the generation of electricity from a coal source, it was pointed out by Applicant Witness Hechling (Tr. 1318 *et seq.*) that the solid constituent is comprised of pit-ash, fly-ash and the absorbent utilized in the SO_2 cleanup system, usually a calcium salt. About 90 percent of the ash is collected from flue gases. Disposition of solids at present is by landfill, a practice currently

[120]This is a 1000 MW plant operating at 80 percent capacity factor.
[121]A favorable effect of the discharge of carbon containing little ^{14}C is a dilution of that radioisotope.

under review by the Environmental Protection Agency (EPA) with emphasis on seepage of toxic metals naturally present in coal into groundwater supplies. The past casual treatment of this apparent innocuous material is now due for a more careful assessment.

329. The creation and shipment of radioactive materials to and from Unit 2 is less an adverse effect on the environment than the transportation and on-site storage of the large amounts of fuel required for a fossil plant. A half-million tons of coal or a quarter-million tons of oil is required to provide a 60-day reserve. (FES at 9-7)

330. Evidence was received from the Applicants and from the Staff on the availability and viability of other, less common sources of energy. Included were coal gasification, solar, winds, solid wastes from population centers, and oceanic temperature differentials.

331. Successful gasification of coal requires a resource of low-sulfur content and large quantities of water, needs incompatible with their geographic distribution in the United States. That is, low-sulfur coal is mainly in the west with adequate water supplies at the Mississippi River and points east. (Applicants' Witness White at 75 following Tr. 955)

332. The generation of electric energy from solar sources is in development. With an availability factor in New England estimated at 12.5 percent (3 hrs/day), large efficient collector and hugh thermal storage capacities are required. A source of 1100 MW would employ 300 square miles of collector plus storage space. Home heating and home-water heating is a more viable undertaking and, in fact, is underway on a limited scale. Present costs of a solar unit for home heating are relatively high (10 percent of the cost of a 3-bedroom home to supply 50 percent of the average seasonal heating requirements) and a backup system, preferably electric, would be required. Demonstration installations have shown a need for further materials research to cope with observed deterioration of collectors. This use of solar energy would reflect on the demand for oil, the principal heating source, rather than for electricity. (White at 76 *et seq.* following Tr. 955)

333. Commonwealth Witness Converse testified to the technological feasibility of using solar energy for space heating in New England as an economically competitive supply. Since space heating accounts for 40 percent of all the energy requirements in New England and solar is a viable source, he concluded that, with progressive development, solar energy could supply 10 percent of that requirement by the end of this century. In his experience, an experimental solar space heating installation, at a capital cost of $12,000, in a 1.5 story structure occupied by shops and having a floor area of 2300 ft^2, is supplemented by heat pumps and resistance heating. Although some heat panels have lost effectiveness through opacity

of covers and some have been removed from service for other reasons, a not unusual experience in a developing technology, the solar installation provided about 40 percent of the heat required in the winter and spring of 1975. (Commonwealth Exhibit 2 admitted at Tr. 1540; Tr. 1606, 1615, 1544 *et seq.*)

334. In cross-examination it was established that only about 1 percent of the space heating in New England is electrically supplied. (Tr. 1560) Since some of the required backup systems to solar power installations would be electrical, the electrical energy requirement might even increase.

335. Staff Witness Vetrano stated that the production of electricity from steam generated from solar energy has not been shown to be feasible; that production of electricity through photovoltaic action is not presently feasible because of low efficiency and undependable equipment (an energy density of 0.1 W/cm^2 and a 10 percent collector efficiency results in a requirement of 2500 acres for the generation of the Unit 2 design power output); that the potential of solar energy for domestic water heating and for home heating and cooling is viable. He concluded that increasing fuel costs and the expected decrease in the cost of improved solar collectors would make solar home heating and cooling an economical option in the 1985-1990 period; and, with respect to wind power, he estimated that 2000 windmills off the Massachusetts shore would be required to produce the electrical output of Unit 2. This number is derived from an annual average wind power of 800 W/m^2 , 200-ft blades and 20 percent efficiency. (following Tr. 1409)

336. Applicants' Witness White considered the production of useful energy from wind power to be a likely supply from small units with present capacity being limited to about 5 kW/unit. Research projects directed toward a 100 kW unit are underway. (following Tr. 955) In cross-examination Witness White emphasized the variability of the wind, and hence, of power availability and the need for energy storage. He considered wind not to be a source of bulk electrical energy although it has promise for home application. (Tr. 1394)

337. The temperature differential of 20 Celsius degrees required for production of ocean-thermal power is not available in the New England shore areas; the effects on aquatic biota of transporting the required large quantities of seawater and the underwater transmission of the energy are considerations in areas where the method is technically viable; and the estimated cost is 50 mills/kWh in 1974 dollars for a 100 MWe plant not including electrical transmission costs. (Vetrano at 41 following Tr. 1409; White at Tr. 1395)

338. The production of useful energy from burning trash and other solid wastes was discussed by witnesses representing the Applicants, the

Commonwealth and the Staff. Applicants' Witness White stated that solid wastes would be most useful as a supplement to the combustion of fossil fuels. Possibly as much as 100 MWe could be obtained from the waste produced in the Boston area. Even if direct use of waste as a fuel were developed, a generating capacity equal to that of Unit 2 would require a transportation system to collect solids from a population greater than that of Massachusetts. (White at 81 following Tr. 955)

339. Commonwealth Witness Cousins described the beneficial recovery of energy from solid wastes as a developing technology and stated that, by statutory requirement, the Commonwealth is conducting research and development on improved methods of disposal including recycling. A plant consuming 1200 tons of trash per day for the generation of process steam is in operation in Massachusetts. The trash is not mixed with other fuel. A second plant, underway for 1979 operation, is expected to produce 60 MW of electric power from burning 3000 tons of mixed refuse per day, probably without fossil fuel addition. The electrical energy generating potential from combustion of all the solid waste produced in Massachusetts is 470 MW. Further, however, the witness testified that the use of solid waste as a fuel "will not replace Unit 2." (following Tr. 5411, particularly fn. 1 at 3; also Tr. 5463, 5439 and 5452)

340. The commitment of the Commonwealth to support the technology of electric power generation from solid wastes was confirmed by Witness Neely. (Commonwealth Exhibit 17 at 17)

341. Staff Witness Vetrano noted that the expected generating cost at Boston area plants is an attractive 8 mills/kWh with pickup a major cost component. Location of such facilities near urban areas is therefore desirable. Also, the combustion of trash combined with coal is a preferable method. Trash burning is susceptible to problems of particulate and gaseous-contaminant emissions similar to those of coal. Scrubbers or other decontaminating devices are required. (at 35 following Tr. 1409)

342. The Staff considered other processes, all requiring research and/or development to establish their economic or practical applicability. These include pyrolysis and hydrogenation, anaerobic digestion and coal gasification. (at 24 *et seq.* following Tr. 1409)

343. No viable source of geothermal power is known to exist in New England. (FES 9-2)

ii. Alternate Sites for Unit 2

344. As stated in the Preliminary Statement of this Decision (para. 15 *supra*), this Board rejected in a prior Partial Initial Decision the analysis of alternate sites by the Staff. The Board found the Staff's analysis "...to be

couched in generalities." The Board further found "...no record of a careful examination, either physically or by review of proffered descriptions of other than [the Applicants' proposed] Rocky Point." The Board concluded that "...the Staff's evaluation of alternate sites is inadequate, and...this deficiency requires the denial of the Applicants' application for a Limited Work Authorization." [6 NRC 839 at 845 (1977)] In upholding this decision the Appeal Board[122] rejected the Staff's "generalized" review process which led to the elimination of all other potential sites without a detailed examination of specific sites, including site visits.

345. Licensing the construction of a nuclear power plant is a "major federal action" within the meaning of Section 102(2)(C) of the National Environmental Policy Act of 1969 (NEPA). This section of NEPA requires the Commission to "...consider whether reasonable alternatives less harmful to the environment exist before allowing a utility to proceed with construction." (*Id.* at 778) To satisfy NEPA, the agency must identify, study and compare alternative sites for the location of the proposed facility. In determining whether a proposed site is environmentally acceptable, the Board must find that after giving each alternative site a "hard look" none is found "obviously superior" to the one proposed by the Applicant. (*Public Service Company of New Hampshire* (Seabrook Stations Units 1 and 2), CLI-77-8, 5 NRC 503 (1977).)

346. Subsequent to the Board's decision (LBP-77-66) and the Appeal Board's affirmation (ALAB-479), the Staff undertook to remedy the alternate site review deficiencies. To assist the Staff in its review, BECo submitted on January 26, 1978, a draft siting study entitled "Boston Edison Company Siting study for Long-Term Capacity Expansion - 1975 to 2000" [the 1974 siting study, Applicants' Exhibits 14(A), (B) and (C)]. The 1974 siting study was not prepared for the purpose of supporting the construction of Unit 2 at the Rocky Point Site but rather for the purpose of identifying current and future generating options and other sites for the Boston Edison Co. to the year 2000. The study assumed that the Rocky Point Site was planned for three nuclear plants, and further utilization of the site was not considered. The 1974 siting study was updated by BECo on May 30, 1978. (Staff Exhibit 53 at vii bound following Tr. 9852)

347. The 1974 siting study conducted by United Engineers and Constructors, Inc. (UE&C) used a radial approach. The review started with the center of the BECo service district and moved radially outward along resource areas (water bodies) until a decision was made that a sufficient number of sites had been identified. After examination of over 100 parcels

[122] *Boston Edison Company* (Pilgrim Nuclear Generating Station, Unit 2, ALAB-479, 7 NRC 774 (1978).

of land in eastern Massachusetts, a total of 24 fossil and nuclear sites were identified, of which 10 were deemed satisfactory as possible nuclear plant sites. (Applicants' Witness Griffin at 6 following Tr. 9608 and Staff Exhibit 53 at 3-2) The 1974 siting study was limited to eastern portions of Massachusetts and did not include the largest fresh water resource in the state—the Connecticut River. The Staff considered this to be a major deficiency of the 1974 siting study and in its comparison of alternative sites the Staff included Montague as a representative site from the Connecticut River Valley to compare with the Applicants' proposed Rocky Point Site. (Staff Exhibit 53 at 4-1)

348. After its initial evaluation of the 1974 siting study the Staff requested additional information. BECo supplied the requested additional information in 1978, including responses to Staff questions and reconnaissance level information obtained from various sources. The Applicants reviewed and updated the description of nine potential nuclear sites identified in the 1974 site study and provided similar information for existing sites in New England, including Charlestown, Seabrook, Montague, and Millstone. (Applicants Exhibit 15 received at Tr. 9637) The Staff supplemented Applicants' further information with data gathered independently, including site visits by each member of the team responsible for preparing supplements to the FES. (Staff Witness Scaletti at 2 following Tr. 9852)

349. During the course of their review, the Staff visited 19 sites, including among others Millstone, Seabrook, Montague, Pilgrim and each of the 10 candidate nuclear sites listed in the Applicants' 1974 siting study. (*Id.*) Millstone and Seabrook are both located outside the State of Massachusetts and even though Applicants might have legal problems locating power plants outside the Commonwealth, Staff included them in its evaluation. (Staff Exhibit 53 at 4-1)

350. The Staff analysis of the 1974 siting study was divided into (1) an assessment of the site selection process and (2) an assessment of the candidate sites. Their assessment of the site selection process resulted in the addition of three sites (Montague, Millstone and Seabrook) to the list of candidate nuclear sites of the 1974 study. The Staff's final assessment included Montague, Millstone, Seabrook and 9 sites from the 1974 study (3 located within 20 km of the Merrimack River, 4 within the town of Plymouth, and 2 located in the Buzzard Bay area) for a total of 12 alternates for comparison with the Rocky Point Site.

351. Twenty-three characteristics of each of the 12 sites were evaluated and compared directly with Rocky Point. The comparison includes: water availability, terresterial ecology and land use, socioeconomics, demography, hazards, aquatic ecology and water quality, geology and seismology, and

meteorology. Characteristics of the sites were rated as superior, equal or inferior to Rocky Point. (*Id.*)

352. The Staff conclusion after comparing the environmental attributes and site characteristics of each of the 12 alternate sites is that none of the candidate sites is superior to the Rocky Point site. (Id. at viii and 4-60)

353. As regards underground siting, BECo's 1974 siting study addressed the issue and, although indicating certain possible advantages, did not recommend the underground siting concept. (Applicants' Exhibit 14 at III-29, 30 and 72, 73) Both Applicants and Staff testified that underground siting for Unit 2 is not a practical alternative within the current schedule for construction. (Applicants' Witness White at 65 *et seq.* following Tr. 1656; Staff Witness Harbour at 9 following Tr. 1493) No general designs currently exist for plants to be built underground and technical problems such as potential flooding and assuring the stability of the site have to be resolved. (White at 66 following Tr. 1656; Harbour at 4, 5 following Tr. 1493) Underground siting would also entail substantially higher plant costs. (Applicants' Exhibit 14 at III-29; Harbour at 5, 6 following Tr. 1493) Both Applicant and Staff witnesses testified that offshore siting is not a reasonable alternative for Unit 2 at this time since most of the coastal areas of Massachusetts are protected ocean sanctuaries. The design and feasibility of offshore plants are currently under Staff review and no plants have been licensed to date. The record shows that Applicants have evaluated inland sites along the Merrimack, Concord, Nashua and Taunton Rivers. (Applicants Exhibit 14, Figure V-1 and page VI-3) Both Applicants and Staff agree that the only available fresh water source in Eastern Massachusetts capable of supporting a large power plant is the Merrimack River. (*Id.*; Applicants' Witness Griffin at 5 following Tr. 9607; Staff Exhibit 53 at 4-1 and 4-6 through 4-19 bound following Tr. 9952) The Staff also considered the Montague site, an inland site located on the Connecticut River in Western Massachusetts. (Staff Exhibit 53 at 4-44 through 4-52)

iii. Alternate Condenser Cooling

354. The condenser cooling systems considered by the Applicants and reviewed by the Staff included: open-cycle, "once-through" cooling, utilizing Bay water with a temperature increase up to 29°F; cooling ponds; spray canals; mechanical-draft saltwater cooling towers; natural-draft saltwater cooling towers; combinations of various open-cycle/closed-cycle systems; and dry-cooling towers. (ER §10.1.1; FES at 9-7 to -16) The system selected by Applicants based upon technical, economic and environmental consideration is a once-through cooling system with a condenser temperature rise of 22°F across the condensers. The two

alternatives considered feasible for detailed consideration were mechanical-draft and natural-draft cooling towers utilizing saltwater because less than 10 percent of the freshwater required is available. A summary comparison of these three systems is in ER Table 10-1.

355. A mechanical-draft saltwater cooling system would require three towers approximately 50 x 400 by 75 feet high. Compared to a once-through installation, this method would discharge less heat to the Bay and would require 65 percent less cooling water, thereby reducing the loss of aquatic biota through impingement and entrainment.[123] It would, however, result in serious problems of salt drift and noise. The Staff estimates that up to 2.5 tons of salt per day could be deposited on surrounding land resulting in property damage and impacts on terrestrial vegetation. (FES 9-10) Further, noise levels due to operation of motors and fans in the towers would be substantial. Applicants have estimated that fog from towers might adversely affect local boat navigation an average of 160 hours per year. (ER 10-33) Applicants further estimate that a mechanical-towers cooling system would result in an annual average generating capacity 2.5 to 3 percent less than with a once-through system, with a maximum of 5 percent during peak summer temperatures. (ER 10-7)

356. A natural-draft cooling tower system would include a single tower 370 feet high by 310 feet base diameter and would be more costly than mechanical towers. While impacts from salt drift and noise would be slightly less than from mechanical-draft towers, these impacts would, nevertheless, be substantial. Further, the natural-draft tower and its plume would be much more visible to the surrounding area. (ER 10-7; FES 9-11) The water requirements would be approximately 40 percent of the proposed once-through cooling system. Average generating capacity is estimated to be 2.5 percent less than with a once-through system, with maximum of 6 percent during summer peak ambient temperatures. (ER 10-7A)

357. Both Applicants and Staff concluded from their evaluations that none of the alternatives was environmentally preferable to the system or treatment proposed by the Applicants.

IV.B. e. Environmental Monitoring

358. The Applicants' preoperational monitoring for Unit 2 is based upon their experience with Unit 1 and is described at ER § 6.1. Studies were made of the characteristics of Cape Cod Bay and of the marine ecology in

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[123]Applicants presented some evidence on the survival of aquatic organisms upon passage through a condenser cooling system (ER 5-24 *et seq.*) and estimate that perhaps as few as 10 percent, excluding fish eggs and larvae, will be killed whereas essentially none would survive in a cooling tower. Accordingly, the mortality in a once-through cooling system may be less, overall, than in one employing towers. (ER 10-44).

the vicinity of the Pilgrim Station. Meteorological observations include continuous wind velocity, temperature, and vertical temperature gradients. The Staff concurs generally with the proposed Unit 2 preoperational program and recommends the addition and/or intensification of a number of studies on aquatic biota. (FES at 6-4) The Staff also recommends additional determination of concentrations of copper and nickel in nearby seawater and in organisms that concentrate these elements and that chlorine demand studies be made of Unit 1 intake water as a function of temperature and season. (FES at 6-7)

359. A comprehensive preoperational environmental radiological monitoring program for Unit 2 was established through agreement between the Applicants and one of the Intervenors. It was entered into the record as a stipulation.[124]

360. Staff witnesses testified (Tr. 6452) to the acceptability of the monitoring program described in the stipulation and to its adequacy under NRC regulations.[125]

361. Applicants' Witness Wrenn (Tr. 6419) and Staff Witness Bores (Tr. 6457) stated that the requirements of the monitoring program were within the capability of existing techniques and instrumentation.

362. The specifications of the stipulation constituted a revision to the FES, Section 6.1.4.1, thereby becoming the Unit 2 preoperational monitoring program[126] and will also become a part of the technical specifications of Unit 1 (Tr. 6454) and of the Unit 1 operating license. (Staff Exhibit 11C)

IV.B. f. Environmental Effects of Postulated Accidents

363. The environmental effects of postulated accidents have been assessed by the Applicants. (ER § 7) The Staff has reviewed this assessment, has made independent calculations, and has concluded that the environmental risks of the accidents are extremely small. (FES at 7-1) The radiological effects of accidents on the environment have been assessed using the standard assumptions and guidance issued by the Commission as a proposed amendment to Appendix D to 10 CFR Part 50 on December 1,

[124]Massachusetts Wildlife Federation Exhibit MWF-1A and MWF-1B at Tr. 6460. At Tr. 7633 the Board established that agreement existed on the stipulation among the parties, and dismissed the MWF contentions made moot by the stipulation.
[125]At and near Tr. 6450 the stipulation MWF-1B is referenced as Exhibit B.
[126]Staff Exhibit 11C following Tr. 7828.

1971 (36 *FR* 22851).[127] The Applicants' analysis considered accidents of a wide range of severity including some of those of Class 8 in the Commission's assumptions, such as those arising from the ejection of a control rod and from a break in the reactor coolant piping.

364. Both the Applicants' and the Staff's calculations show that the radiological exposures to a member of the public at the site boundary will be no more than 35 percent of the limit specified in 10 CFR Part 20, that the year 2020 population dose within an area of 50-mile radius about the site, will be no more than 2300 man-rem. In general, the exposure resulting from any of the postulated events will be a small fraction of the exposure due to natural background radiation and will be, in fact, well within naturally occurring variations in the natural background.

IV.B. g. Unavoidable Adverse Environmental Effects

365. In an assessment of unavoidable environmental effects the Staff finds (FES at 10-1) that 28 acres of land in addition to that already committed to an existing electric generating station will be removed from productive use for the life of Unit 2.

366. Noise, typical of construction, may be objectional for the public during early stages of the project.

367. The operation of Unit 2 will require the discharge of about 2000 ft^3 of water per second into Cape Cod Bay at a temperature about 22°F above ambient. Accordingly, the quality of the water will have been somewhat degraded by this usage. Additionally, the water may contain dissolved gases to near saturation.

368. Although the thermal and chemical discharges are expected not to have an adverse effect on the biota of the Bay, there will be some loss through entrainment in the water stream and impingement upon the equipment. These losses are within the recuperative capability of the Bay.

369. Amounts of radiation and radioactive materials, insignificant compared to those present from natural sources, will be released to the environment during routine operation of Unit 2.

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[127]On June 9, 1980, the Commission issued a Statement of Interim Policy on Nuclear Power Plant Accident Considerations Under NEPA. This statement withdraws the proposed annex to Appendix D to Part 50 and announces the Commission's "position 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." This policy applies to all cases in which a Final Environmental Impact Statement has not been prepared. Since the FES for Unit 2 had already been prepared and issued, this policy requirement does not apply to Pilgrim Unit 2.

IV.B. h. Relationships Between Local Short-Term Use of Man's Environment and the Maintenance and Enhancement of Long-Term Productivity

370. Long-term competition among such uses of coastal sites as seaports, power plants, industrial facilities, commercial and sport-fishing and commercial and individual housing developments, may be expected to continue and increase. The particular site in question here is, however, already dedicated to a power plant and other uses are either already excluded by virtue of site incompatability or they are not affected by the plant and can continue. The construction and operation of Unit 2 is consistent with the long-term objective of coastal zone management. The presence of Unit 1 has not inhibited commercial or recreational uses of Cape Cod Bay and the addition of Unit 2 will not change that situation.

IV.B. i. Irreversible and Irretrievable Commitments of Resources

371. The Staff identified (FES at 10-3) material and other resources which would be consumed or otherwise lost from the environment, in their present form, as a result of the construction and operation of Unit 2.

372. Included in this identification are the usual materials of construction, constituents of the reactor and other items unique to the generation of electric power from nuclear energy, chemicals consumed in process operations, aquatic biota destroyed when carried in the coolant, uranium transformed into other uranium isotopes, plutonium and fission products. In the opinion of the Staff the consumption of these material resources will have negligible effect on their reserves. (FES at 10-4)

IV.C. Compliance with Federal Water Pollution Control Act (as Amended)

373. The current status of the Federal and the State permits necessary for the discharge of condenser coolant water and other liquids from Unit 2 is not entirely clear to the Board since appeals on the issuance of these permits have been taken. However, the present record shows the following.

374. National Pollutant Discharge Elimination System (NPDES) Permit No. MA 0025 135 was issued to BECo on March 11, 1977.[128]

375. The Massachusetts Division of Water Pollution Control (MDWPC) by letter to the Applicants dated June 20, 1977, (Staff Exhibit 18A) stated that the proposed discharges of coolant from Unit 2 will not violate §§ 301, 306 and 307 of the Federal Water Pollution Control Act

[128]Staff Exhibits 18A-D admitted at Tr. 8801. Exhibit 18A is the 401 State Water Quality Certification; 18B is the letter transmitting the NPDES permit to the Applicants; 18C is the Federal Permit MA 0025 135; 18D is an amendment to that Permit.

(FWPCA) and expressed the Division's intent to issue a certification provided the discharge will be conducted in a manner which will not violate the conditions stated in the NPDES Permit MA 0025 135.

376. The Applicants and the Staff state that NPDES Permit MA 0025 135 constitutes a certification under § 401 of FWPCA and that the MDWPC letter of intent commits the Commonwealth to a certification. The Staff stated that "Both units [Pilgrim 1 and 2] have received the necessary approvals and permits from the Environmental Protection Agency . . ." (At 4-2 Staff Exhibit 53)[129] Staff Witness Lehr supported this FES conclusion. (Tr. 9965)

377. Counsel for Intervenors Cleetons and others have challenged the authority of these permits and have appealed their issuance. (Tr. 9965) This issue has been through an adjudicatory hearing before the EPA.

378. There is no evidence in the record to the effect that the Commonwealth has exercised its authority to stay the certification represented by the permits during the appeal [30A Mass. Ann. Laws §14(3)].

IV.D. Cost-Benefit Analysis

379. The Staff has carried out a cost-benefit analysis of the construction and operation of the proposed Unit 2 by an established methodology and through the application of judgmental factors. The analysis has led to environmental and monetary costs which are compared with the benefits to be gained. (FES § 10.4)

380. The basis of the Staff's analysis, in addition to the impact of the generating station itself, included the impact of the uranium fuel cycle as specified in 10 CFR 51.20(E) and set forth in Table S-3.[130] It also included the environmental effects of transportation of fuel and of solid radioactive wastes to and from the facility as specified in 10 CFR § 51.20(g) and in Table S-4.

381. The principal benefit of the plant is the production of electrical energy to fulfill the requirements of the Applicants' customers and to replace presently operating oil-fired generating stations. The consequences of these replacements have been extensively discussed in ¶ 184 *et seq. supra.* Based on a 60 percent capacity factor, the generation of electricity will be approximately 6.2 billion kilowatt hours per year. (FES § 10.4.2, revised in Staff Exhibit 13 following Tr. 8308)

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[129]Staff Exhibit 53 is the Final Supplement to the Final Environmental Statement received at Tr. 9852, bound following Tr. 9952.

[130]See also Section IV.B.c.v. *supra* which summarizes the Board's findings on the impact of radon-222 and technetium-99 which were also considered.

382. The economic costs for constructing, operating and maintaining Unit 2 over its 30-year projected life will be $2.227 billion. (FES Table 10.2, revised in Staff Exhibit 13 following Tr. 8308)

383. The principal environmental costs identified are those which have been described previously in this Decision and include impacts during construction and operation of Unit 2, minor radiological exposures to the population from both Unit 2 and the segment of the national nuclear fuel cycle attributable to Unit 2, and a small risk potential in the transport of radioactive materials.

384. The costs and benefits of emergency planning and TMI-related issues have not been factored into this cost-benefit analysis. After evidentiary hearings on those issues are completed the Board will reassess its cost-benefit balance.

V. CONTENTIONS ADDRESSED IN THESE PROCEEDINGS

385. A number of issues were introduced into these proceedings through contentions filed by the Intervenors and accepted[131] by the Board in a Memorandum and Order dated February 18, 1975. Each Party was offered the opportunity to present evidence on the surviving contentions. The findings by the Board appear in Parts II, III and IV *supra*. This Part contains a statement of each contention and its disposition by the Board based on the findings.

A. Need for the Power to be Generated by Unit 2

386. Statement of Contentions:

> **Commonwealth 6** "The need for the electrical generating capacity of Pilgrim 2 has not been properly established because the Applicants have not developed a model adequately considering the effects of the following on demand:
>
> (a) Voluntary curtailment of consumption of electricity by the public;
> (b) Elasticity of demand;
> (c) Peak load pricing to flatten demand; and
> (d) New standards for improved building insulation, heating, lighting and air conditioning."
>
> **Cleeton H** "Applicants and Staff have not adequately demonstrated the need for additional power in that the projected needs are inaccurate and conservation has not been seriously examined."

[131]Of the contentions originally filed, a number were rejected by the Board before the onset of the hearing. Some of those accepted were later withdrawn by their originators and some were modified. These actions are summarized in the Preliminary Statement, ¶ 4, *supra*.

387. At Part IV.B.a. *supra* (¶ 180 through ¶ 230) detailed findings of fact by this Board on the need for Unit 2 have been made. These findings, based on the evidence submitted by the Parties, lead the Board to conclude that while it is difficult to predict future need with any degree of certainty, the record clearly shows that a positive growth rate in the electrical requirements of the New England region exists. The electric utilities have a continuing legal duty to provide an adequate supply of electrical power to their customers and this requirement dictates conservative planning. The Board therefore concludes that Unit 2 is needed to meet these future requirements. The Board also finds that the operation of Unit 2 will be in regional and national interests by substituting for the consumption of oil which has been and may be in short supply or better utilized for other purposes. Further, the Board opines that substitution alone constitutes an adequate basis for Unit 2.

V. B. Overstatement of Production of Electrical Energy

388. Statement of Contention:

> **Commonwealth 8** "The benefits of the proposed facility have been overstated with respect to projected production of electrical energy."

389. This contention is addressed in large degree in the more broad topics of Part V.A. *supra* and in Part V.E. *infra*, the discussion of contentions of the need for power and of alternate [to Unit 2] sources of energy, respectively. The relevant findings of fact on those issues are at least implicitly in Parts IV.B.a. (¶ 180 through ¶ 230) and IV.B.d.i. (¶ 305 through ¶ 343). Those broad citations can be reduced to ¶ 305 through ¶ 324 as the principal reference to the issue. The basic concept here is the reliability of Unit 2 once operating and is measured by the capacity factor. The values of the capacity factor predicted for Unit 2 vary widely among the several witnesses and the uncertainties attached to those values have an even greater compass. The absence of statistical reliability in the testimony of witnesses presented by Intervenor Commonwealth leaves the Board unpersuaded by their arguments. Accordingly the Board sees no reason not to accept the expected performance envisaged by the Applicants and the Staff. It is observed that the Staff reduced its prediction during the course of this hearing. Translation into the context of this contention leads to the judgment that it lacks foundation.

V. C. Financial Qualifications Of Applicants

390. Statement of Contention:

Commonwealth 5 "The Applicants are not financially qualified to design and construct the proposed facility."

391. Part II.D.c. *supra* (¶ 66 through ¶ 79) are findings of fact by the Board on the financial qualifications of the Applicants. Based upon these findings the Board concludes that the Applicants have made an adequate showing of their financial ability to construct the proposed facility.

V. D. Technical Qualifications

392. Statement of Contention:

Commonwealth 10 "The Applicants and their architect engineer, Bechtel Corporation, and nuclear steam system supplier, Combustion Engineering, are not technically qualified to engage in the proposed activities and cannot provide an adequate quality assurance program based upon their previous records in similar ventures."

393. At Part II.C. *supra* (¶ 44 through ¶ 65) are findings of fact by the Board on the technical qualifications of the Applicants and their principal contractors. No evidence was presented by the Commonwealth in support of this contention other than cross-examination of Applicants and Staff witnesses. Based upon the findings, the Board concludes that the Applicants and their principal contractors are qualified to construct the proposed facility.

V. E. Alternate Energy Sources

394. Statement of Contentions:

Commonwealth 3 "The Applicants and Staff have not given adequate or accurate consideration to solar power, wind power, the use of fossil fuels, the use of fuel derived from solid waste, or the high temperature gas-cooled reactor as alternative sources of power."[132]

Cleeton I "Applicants have not adequately considered alternate sources of power in that they have not considered: methods of thermonuclear fusion; wind power; solar energy, utilization of ocean temperature differences; gasification of coal; production of low sulfur oil from garbage, animal waste and coal; or cultivation of high energy algae for conversion to methane or for direct power plant combustion."

[132]On motion by the Commonwealth dated September 25, 1975, its originally stated contention was modified by the Board to include solid waste as a potential fuel. (Tr. 832).

395. At IV.B.d. *supra* (¶ 306 through ¶ 343) are findings of fact by the Board on possible alternative methods of producing energy by the use of other types of fuel or other technologies. The Commonwealth presented a number of witnesses supporting its contention and cross-examined the Applicants' and the Staff witnesses. Intervenors Cleeton offered no witnesses but did cross-examine Applicants' and Staff witnesses. Upon consideration of the environmental, economic, technological and practical factors presented by the various witnesses, the Board concludes that Unit 2 will be a source of electrical energy, over the span of its expected life, superior to other proposed sources. The choice between nuclear and coal cycles is tipped toward nuclear by present-day economics and the uncertainty in the future effect of the restrictions and regulations expected to be applied to reduce the environmental impact of the combustion of coal. Evaluations of that impact are only now beginning. Their consequence is expected to further worsen the economy of the coal fuel cycle. The Board expresses confidence and faith in the technical community in its pursuit of acceptable alternates. Of greater promise among those considered, it seems to the Board, are solar energy and the combustion of municipal solid wastes. The first is strikingly free of pollutants; the second, though fraught with the decontamination problems of other combustion processes, will provide a sorely needed reduction in a continuing and aggravating disposal problem and, at the same time, will recover energy otherwise destined for landfills at an ever-increasing rate. Additional sources show promise but their perfection will occur on a schedule well beyond the requirement Unit 2 is intended to fill. The Board concludes that there are at present no viable alternative energy sources.

V. F. Alternate Sites for Unit 2

396. Statement of the Contentions:

> **Commonwealth 4** "The Applicants and the Staff have not given adequate consideration to underground siting, off-shore siting and inland siting using closed-cycle cooling systems, as alternate types of sites."
>
> **Commonwealth 12** "Neither Applicants nor Staff have adequately considered the alternative of locating the proposed plant at a site more suitable from a population density and environmental standpoint."
>
> **MWF 1(b)** "To the extent that the practicability of such additional or alternative means [of complying with ALARA standards] is site-dependent, including without limitation factors relating to transportation, the Applicants and Staff have failed to consider adequately

alternate sites in light of the desirability of such additional or alternative means."

397. At Part III. and Part IV.B.d.ii. *supra* (¶ 127 through ¶ 178 and ¶ 344 through ¶ 353) are findings of fact by this Board on the existence of obviously superior alternate sites for the proposed facility. The Board finds that Applicants and Staff have adequately considered the alternatives of underground siting, offshore siting and inland siting, including those which would employ closed cycle cooling systems. Based on the record the Board concludes that the Staff has adequately evaluated and compared in detail a sufficient number of diverse and potentially licensable alternate sites to meet the requirements of NEPA and the Commission's regulations. The Board concludes that the population density estimated for the area contiguous to the site proposed for the Unit 2 nuclear generating station throughout its projected life is within guides established by the Commission and, accordingly, that the projected density is not cause, in itself, for selecting other sites. The Board concurs with the conclusions of both Applicants and Staff in that none of the alternate sites considered in this proceeding is "obviously superior" to Rocky Point, the Applicants' preferred site. The Board finds that the predicted radiological effects of the gaseous and liquid discharges from Unit 2 are so small as to make any comparisons with other sites on the basis of estimated population dose unnecessary.[133] With respect to the suitability of Rocky Point, Board findings (¶ 127 through ¶ 178 *supra*) indicate that the site is suitable. Specifically, the Board finds that the population center selected by Staff establishes an LPZ conforming to the Commission's regulations and, from geographic and population viewpoints, the proposed Unit 2 site is suitable for the location of a nuclear plant of the general type and size proposed by the Applicants. The Board further finds that there are no nearby industrial, military, or commercial facilities which would cause the site to be unsuitable. The atmospheric dispersion conditions at the proposed Unit 2 Site are better than at 70 percent of some 80 sites which have been proposed

[133]Subsequent to the admission of MWF contention 1(b), the Commission promulgated (40 *Fed. Reg.* 19439 dated May 5, 1975) Appendix I to 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents." Compliance with Appendix I is discussed in Part II.G.c. *supra.* For the purposes of assessing the validity of MWF's contention one need only take notice of the calculated population doses from liquid and gaseous releases (summarized in ¶ 125 *supra*). In accordance with the $1000 per man-rem criterion established by Appendix I § II.D. the maximum expenditures that could be required for a radwaste system augment are less than $1000 for liquids and $3400 for gases assuming that the augments would reduce the discharge to zero. The Board considers this potentially required expenditure by the Applicants to be *de minimis* and as such would not be a factor upon which an alternative site would be selected.

for other reactors throughout the country. The Board finds the site suitable based on meteorological considerations. The Board also finds the Site suitable from hydrologic, geologic and seismic viewpoints. The Category I structures shall be designed to withstand a horizontal seismic-induced ground acceleration of 0.20 g and the existing soil is suitable for that purpose.

V. G. Impact of Aircraft on Pilgrim Site

398. Statement of Contention:

> **Cleeton C** "Neither Applicants nor Staff have adequately considered the health, safety and environmental risks originating as a consequence of locating Pilgrim Unit 2 in the proximity of a major descending flight path to Logan Airport and the potential impact of descending aircraft on the Unit 2 site. A consideration of such risks would lead to the selection of a site more suitable from a health, safety and/or environmental viewpoint."

399. At Part III.C. *supra* (¶ 142 through ¶ 151) are findings of fact by the Board on the potential impact of aircraft on the Pilgrim Site. The Board concludes that the probability of an impact on vulnerable portions of the site is so small as not to be credible.

V. H. Alternate Condenser Cooling

400. Statement of Contention:

> **Commonwealth 2** "Alternative cooling systems employing towers at the proposed site are available but have not been adequately assessed by the Applicants or Staff."

401. Part IV.B.d.iii. *supra* (¶ 354 through ¶ 357) are findings of fact on this contention. The Board finds that Applicants and Staff have adequately evaluated alternative condenser cooling methods including their intake and discharge systems and further finds that no significant environmental advantages would be realized by the use of either mechanical or natural draft saltwater cooling towers over the proposed Unit 2 once-through cooling system.

V. I. Adverse Effects of Unit 2 on Cape Cod Bay

402. Statement of Contention:

Commonwealth 1 "The Applicants and the Staff have not adequately or accurately considered the potential adverse effects on the Cape Cod Bay ecosystem of:

a. Entrapment and impingement of fish and other biota in the intake structure.
b. Entrainment of ichthyoplankton and larvae in the condenser cooling system.
c. Supersaturation of ambient atmospheric gas entrained in cooling water with possible resultant fish mortality.
d. The loss of Irish moss vegetation due to entrainment of spores, thermal effects and bottom scouring.
e. The use of biocides as an anti-fouling mechanism.
f. The use of heat treatment to control mussels.
g. The attraction of certain fish, including menhaden and pollock, to the heated water in the vicinity of the discharge.
h. The recirculation of heated water into the cooling system."

403. At Part IV.B.b.ii. *supra* (¶ 238 through ¶ 240) and in IV.B.c.ii *supra* (¶ 253 through ¶ 263) the impact on Cape Cod Bay from both construction and operation are the subject of findings of fact by the Board. Commonwealth offered no testimony on this contention but cross-examined Applicants' and Staff witnesses. Based on the Board's findings cited above it is concluded that the Applicants and Staff have adequately considered the potential adverse effects of the proposed facility on the Cape Cod ecosystem and found them to be within acceptable limits.

V. J. Environmental Impact of Routine Releases of Radioactive Material

404. Statement of Contention:

Cleeton E "The routine discharges of radioactive materials and/or attendant routine doses of radiation caused by the operation of Pilgrim Unit 2 constitute an unreasonable threat to the health and safety of the Intervenors' family."

405. At Part IV.B.c.iv *supra* (¶ 265 through ¶ 286) findings of fact by the Board appear. Although this contention was admitted only for the limited purpose of "...[permitting] the Cleetons to demonstrate, if they could, the specific environmental impact, if any, on the health and safety of Intervenors' family by routine releases of radioactive materials caused by the operation of Pilgrim Unit 2," (Board Order dated April 2, 1975), the Applicants, the Staff and the Cleetons presented extensive evidence. The

contention was not admitted as a generic item or as a challenge to Commission regulations. The testimony of the Cleetons' witnesses failed to show unusual circumstances whereby the Cleeton family is inordinately susceptible to the effects of radiation. Absent a showing of such circumstances, Commission rules and regulations governing releases of radioactive material and radiation exposures apply. The record shows that Unit 2 is designed to operate in conformance with these rules. Accordingly the Board accepts the testimony of witnesses of the Applicants and the Staff stating that radiologic effects of routine releases from Unit 2 will be small compared to those attributable to background sources and hence negligible.

V. K. Theft and Sabotage

406. Statement of Contention:

> **Commonwealth 9** "The Applicants and the Staff overstate the advantage of the nuclear option as opposed to alternative methods of electrical generation by understating the risk of theft and sabotage attendant on nuclear generation, the costs of which, if considered in the cost-benefit analysis for Pilgrim 2 would cause the overall costs of the facility to outweigh its benefits."

407. At Part II.E. *supra* (¶ 81 through ¶ 93) are findings of fact by the Board on the risks of theft and sabotage of the proposed facility. The Board finds that the potential risk of sabotage and of theft of radioactive materials, including used fuel, from within Unit 2 or in transport, by unarmed persons is sufficiently small in the overall cost-benefit analysis not to affect a conclusion that alternative generation options are inferior to nuclear energy.

V. L. Transportation Risks

408. Statement of Contention:

> **Cleeton B** "Applicants and Staff have not properly assessed the radiological risk to Intervenors' health and safety caused by possible future accidents of vehicles used in the transportation of nuclear fuels and nuclear wastes to and from the Pilgrim 2 site."

409. At Part IV.B.d. *supra* (¶ 287 through ¶ 290) are findings of fact by the Board on the question of risks in the transportation of nuclear fuels and wastes. The Intervenors presented no direct testimony in support of their contention on the risks arising from accidents during such transport. The Intervenors cross-examined witnesses presented by the Applicants and the Staff. Upon review of this record discussing accidents involving vehicles

transporting nuclear materials, the Board observes no evidence of unusual traffic risks. Accordingly, the Board concludes that the transport of nuclear materials to and from Unit 2 does not constitute an unacceptable risk to the health and safety of the public or of the Intervenors in excess of that engendered by day-by-day commercial activity on highways and railroads.

V. M. Effect of Unavailability of Reprocessing and Waste Disposal Facilities on Costs and the Environmental Assessment of Increased Spent Fuel Storage

410. Statement of Contention:

> **Cleeton K** "The delay in the licensing of reprocessing facilities and in the availability of long term waste disposal and storage facilities will preclude the availability of sufficient fuel for Pilgrim 2. In addition, it will require longer storage of spent fuel at the Pilgrim 2 site, thereby increasing the radiological environmental impact of the facility. These factors will tend to increase the cost of fresh fuel and cause additional storage expenses for the Applicants, which will make the nuclear option more expensive than comparable fossil facilities. Proper consideration of these matters in this proceeding would cause the cost-benefit balance to shift in favor of alternatives to nuclear fueled generation capacity for Pilgrim 2."

411. At Part IV.B.c.vi. *supra* (¶ 295 through ¶ 304) are findings of fact by the Board on this contention. The Board concludes that sufficient fuel for Unit 2 is assured throughout its lifetime even absent the recovery of resources by reprocessing used fuel from operating nuclear reactors. Further, the additional fuel costs entailed by that absence is acceptably small. The impact on the environment of storage of used fuel at the Pilgrim site is solely an increase in the nearby radiation field. That increase, however, is negligible compared to the natural radiation background.

V. N. Compliance with ALARA Standards

412. Statement of Contention:

> **MWF 1(a)** "The Applicants' plant design does not comply with the Commission's 'as low as practicable' standards since the releases of radioactive materials in liquid and gaseous effluents may be further reduced through the use of alternative or additional means such as, for example, additional solidification and filtration systems."

413. At Part II.G.c. *supra* (¶ 117 through ¶ 126) are findings of fact by the Board on this contention. The Board finds that Applicants' plant design

is in compliance with the Commission's regulations imposed to assure releases of radioactive materials in liquid and gaseous effluents are "as low as reasonably achievable."

V.O. Adequacy of Regulatory Staff Inspection Practices

414. Statement of Contention:

> **Commonwealth 11** "The Nuclear Regulatory Commission Regulatory Staff has not demonstrated that its inspection practices are adequate in terms of the frequency and scope of inspection to monitor the quality assurance programs of nuclear power plant manufacturers. Absent more stringent inspection of such quality assurance programs, the issuance of a construction permit for the proposed Pilgrim 2 facility will be inimical to the health and safety of the public."

415. At Part II.G.b. *supra* (¶ 114 through ¶ 116) are findings of fact by the Board on this contention. The Board agrees that the adequacy of the Staff inspection program must be measured in conjunction with the overall inspection and quality assurance effort which is applied to the manufacture of nuclear power plants and individual plant components. Viewed in that context the Board finds that the inspection practices of the NRC Staff are adequate in terms of frequency and scope to monitor the quality assurance programs of nuclear power plant manufacturers and that the health and safety of the public are adequately protected by such actions.

V. P. Steam Generator Tube Integrity

416. Statement of Issue:

> **Board Issue** "Evidence regarding the overall integrity of the proposed steam generator tubes will be taken."

417. At Part II.G.a. *supra* (¶ 99 through ¶ 116) are the Board's findings on steam generator tube integrity. The Board finds that there exists the requisite reasonable assurance that the public health and safety will not be endangered as a consequence of tube failure during the operation of Unit 2.

VI. CONCLUSIONS OF LAW

418. The Board makes the following conclusions of law based upon the entire record and all the evidence in this proceeding, including our consideration and evaluation of the Application for Permit and supporting documents submitted by Applicants, the Staff's Safety Evaluation Report and Final Environmental Statement; the written and oral testimony of all of the witnesses; the exhibits admitted into evidence; the Rules and

Regulations of the Commission; the Atomic Energy Act of 1954, as amended; the National Environmental Policy Act, as amended; and relevant NRC decisions and case law.

1. In accordance with the provisions of 10 CFR 50.35(a):

   (a) the Applicants have described the proposed design of the facilities including, but not limited to, the principal architectural and engineering criteria for the design, and have identified the major features or components incorporated therein for the protection of the health and safety of the public;
   (b) such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;
   (c) safety features or components, if any, which require research and development have been described by the Applicants. Further the Applicants have identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and
   (d) on the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at Rocky Point without undue risk to the health and safety of the public.

2. The Applicants are technically qualified to design and construct the proposed facility.

3. The Applicants are financially qualified to design and construct the proposed facility.

4. The issuance of a permit for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

5. The provisions of Section 102(2)(C) and (E) of NEPA and 10 CFR Part 51 of the Commission's regulations have been complied with in this proceeding. In particular the Board has independently considered the benefits and costs of the proposed facility and concludes that the benefits to be derived from Unit 2 outweigh its costs. This consideration has included the impacts of construction and operation on both the terrestrial and

aquatic environments as well as of the uranium fuel cycle including transportation of fuel and other radioactive materials to and from the site.

VII. ORDER

419. The record of these proceedings includes a number of items derived during discussions among the Parties. They constitute commitments by the Applicants to effect various actions. These actions, listed below, are made conditions to any construction permit issued as a result of this Order.

1. The FES, at v, enumerates five conditions designed to protect the environment. These include: a) twenty-one items detailed at FES § 4.5.1 and 4.5.2 intended to minimize the environmental impacts of construction; b) a preoperational monitoring program, described in FES § 6.1 and in MWF Exhibits 1(A) and 1(B), comprised of ecological, water, meteorological, and radiological observations and evaluations; c) the establishment and maintenance of a control program to review conformance of the construction with the conditions set forth in the permit; d) communication to the Commission on construction activities deviating from the conditions of the permit; and e) transmittal to the Staff of analyses of and solutions for any unexpected harmful effects detected during construction.
2. The Applicants shall establish written procedures and instructions to control all construction activities prescribed in the FES and in this Decision and shall provide periodic management audits to determine that all conditions are implemented. They shall maintain records showing compliance with all of the environment-related conditions.
3. The Applicants shall possess and shall prudently exercise their authority to control at any time all activities within the exclusion area except on Rocky Hill Road. This authority shall include the exclusion of personnel and property. During any emergency the Applicants shall additionally have control of Rocky Hill Road through appropriate law enforcement officials.
4. To minimize exposures of aquatic biota to cold shock the Applicants shall make every effort to avoid simultaneous shutdown of both Units 1 and 2 during winter months.

420. A construction permit to build Pilgrim Unit 2 should be issued subject to the above conditions and subject to the favorable completion of hearings on emergency planning and Three Mile Island 2 related issues.

In accordance with 10 CFR 2.760, 2.762, 2.785 and 2.786, this Partial Initial Decision shall constitute the final action of the Commission on the matters considered herein thirty (30) days after issuance, subject to any review pursuant to the above-cited Rules of Practice.[134] Exceptions to this Partial Initial Decision may be filed by any Party within ten (10) days after its service. A brief in support of the exceptions shall be filed within thirty (30) days thereafter, forty (40) days in the case of the NRC Staff. Within thirty (30) days of the filing and service of the brief of the appellant, forty (40) days in the case of the NRC Staff, any other Party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

A. Dixon Callihan
Administrative Judge

Richard F. Cole
Administrative Judge

Andrew C. Goodhope, Chairman
Administrative Judge

Dated at Bethesda, Maryland
this 2nd day of February 1981.

[The Appendix has been deleted from this publication but is available at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.]

[134]In accordance with 10 CFR Part 2, Appendix B, the Board has reviewed the issues decided herein and has determined that none presents serious, close questions which may be crucial to whether a license should become effective before full appellate review is completed. Further, the Board has found no issues on which prompt Commission policy guidance is required.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Charles Bechhoefer, Chairman
Dr. Frederick P. Cowan
Gustave A. Linenberger

In the Matter of | **Docket Nos. 50-329 OM**
50-330 OM
Docket Nos. 50-329 OL
50-330 OL

CONSUMERS POWER COMPANY
(Midland Plant, Units 1 and 2) | **February 12, 1981**

The Licensing Board confirms its earlier bench ruling granting the applicant's motions to compel the depositions of three named NRC staff members, subject to certain limitations. The Board denies the staff's motion for reconsideration of the earlier ruling with respect to one of those members but grants its alternative motion to refer that aspect of the ruling to the Appeal Board. The Board also confirms its earlier denial of a staff motion for a protective order to prevent the further deposition of a fourth staff member.

RULES OF PRACTICE: DISCOVERY (AGAINST NRC STAFF)

Depositions of named NRC staff members may be required only upon a showing of exceptional circumstances. 10 CFR 2.720(h) (2).

MEMORANDUM AND ORDER

(Concerning Depositions of NRC Staff Members)

A. Consumers Power Co. (Applicant) has filed three motions to compel the depositions of named NRC Staff members: (1) a motion dated

January 15, 1981, to depose Kamalaker Naidu (Office of Inspection and Enforcement, Region III); (2) a motion dated January 23, 1981 to compel the deposition of Harold Thornburg (Office of Nuclear Reactor Regulation, Bethesda); and (3) a motion, also dated January 23, 1981, to compel the deposition of Gaston Fiorelli (I&E, Region III). In each case, the Applicant sought not only to take the requested depostion but also the assessment of certain costs against the NRC Staff. The NRC Staff opposed all three motions: it filed a written response dated January 27, 1981 with respect to Mr. Naidu, and it addressed all three motions at the prehearing conference commencing on January 28, 1981.

On December 4, 1980, the NRC Staff filed a motion for a protective order to prevent the further deposition of Joseph Kane (Office of Nuclear Reactor Regulation, Bethesda). The Applicant opposed this motion by reply dated January 9, 1981. With our permission, the Staff on January 27, 1981 filed a response to the Applicant's reply.

The Board heard oral argument on all four motions at the prehearing conference on Jauary 28-29, 1981.[1]

With respect to Messrs. Naidu, Fiorelli, and Thornburg, the Staff generally took the position that, under NRC rules, the Staff could select any of its members to be deposed and that another party could not second-guess the Staff as to the choice of a witness. 10 CFR §2.720(h)(2). Although recognizing an exception for "exceptional circumstances," the Staff asserted that those circumstances had not been demonstrated. On the other hand, the Applicant pointed to specific areas of inquiry which Staff-designated witnesses were unable to address, as well as information tending to indicate that the named Staff witnesses had knowledge in such areas. As for Mr. Kane, the Staff claimed that his deposition to date had been unduly lengthy and repetitive, to the extent that further questioning would amount to harassment. The Applicant claimed that Mr. Kane had been evasive or non-responsive during much of his deposition and that there were particular areas in which Mr. Kane had knowledge which the deposition had not yet reached.

At the January 29, 1981 session of the prehearing conference, the Board rendered the following ruling on these motions from the bench (Tr. 699-704):[2]

> The Board has decided to grant the motions of the Applicant to compel the depositions of Messrs. Naidu, Fiorelli, and Thornburg. We

[1]See Tr. 422-479, 631-634 (Naidu deposition); Tr. 485-525 (Fiorelli deposition); Tr. 537-546 (Thornburg deposition); Tr. 551-611 (Kane deposition). The Applicant's requests for fees were considered at Tr. 612-630, 634-647.

[2]The transcript language has been modified slightly for clarity.

have also decided to deny the Staff's request for a protective order with respect to Mr. Kane, subject to certain requirements. We find that in each case the Applicant has demonstrated exceptional circumstances, within the meaning of 10 CFR §2.720(h)(2), to warrant the deposition or further deposition of the named individuals. Specifically, the Applicant has demonstrated, as contemplated by the foregoing regulation, that the named NRC employees have direct personal knowledge of material facts not known to the deponents heretofore made available by the Staff. In particular:

(1) The Board agrees that the Applicant should be able to question Mr. Naidu about the adequacy of the current QA program. Mr. Keppler, made available by the Staff, expressed no detailed knowledge of this subject (see, *e.g.*, Tr. 462-465) but identified Mr. Gallagher and Mr. Naidu as having knowledge of different aspects of this area. Mr. Gallagher was apparently unable to address certain matters about which he was questioned. The deposition of Mr. Naidu may include (a) whether the QA program has been adequately modified, and will be adequately implemented, to prevent QA deficiencies of the type which have heretofore occurred, and (b) whether the particular QA deficiencies which have arisen with respect to the soils settlement problem have been adequately resolved.

(2) Mr. Keppler also identified Mr. Fiorelli as the individual having knowledge of other QA matters. The Applicant should be able to question Mr. Fiorelli about (a) the SALP board meeting concerning the appraisal of the Consumers Power Company QA program for the Midland facility; (b) the Region III I&E review of non-conformance reports submitted in accordance with ALAB-106 (including the extent to which the NCR's reflect upon the Staff's QA questions which are at issue here); (c) Mr. Fiorelli's discussions or communications with Mr. Keppler on these matters; and (d) matters arising out of Exhibit 2 of the Gallagher deposition.

(3) Mr. Shewmaker, who was made available by the Staff, identified Mr. Thornburg as having particular knowledge in certain areas which Mr. Shewmaker did not possess. Mr. Thornburg should be made available to address (a) a meeting he attended on November 28, 1979, which the parties referred to in their oral argument before this Board, and (b) information he provided to, or discussions he had with, Mr. Stello and/or Mr. Case during the period between that meeting and the issuance of the December 6, 1979 modification order. Specifically, Mr. Thornburg may be questioned about whether, and if so in what

respects, the Staff changed its position concerning remedial actions proposed by the Applicant to ameliorate the soils settlement problem.

(4) Although the depostion of Mr. Kane has been lengthy, we find no evidence of harassment by the Applicant or bad faith by the Staff or Mr. Kane. Mr. Kane plays a significant part in this proceeding, concerning some very technical and complex areas. Mr. Kane should be made available for further questioning concerning (a) cracks in the concrete ring foundation for the borated water storage tanks; (b) the underground piping matter; (c) amendment 85 to the FSAR (at such time as Mr. Kane is prepared to address this subject); and (d) the line of questions which the Applicant attempted to commence at the conclusion of the deposition on December 4, 1980 (Volume VI, p. 403). In addition, Mr. Kane may be asked (for the record) sufficient questions to determine whether he has significant knowledge of the other subjects mentioned by the Applicant at the prehearing conference. If he does not, he need not attempt to answer questions on those subjects. (He also should then *not* be used as a Staff witness on those subjects.) With respect to the change of position reflected in the letter from R.L. Tedesco to the Applicant dated January 8, 1981, Mr. Kane may be asked whether he merely participated in that matter as a conduit or whether he had any substantive input. If the latter situation is the case, he may be questioned concerning that input.

The Board has decided to disallow the claim of the Applicant for costs and expenses. Although we are essentially rejecting the position of the Staff on the various motions, we find no bad faith in the Staff's asserting these positions. In addition, we find that the filing of motions to compel is the usual way contemplated by the Rules of Practice to obtain the testimony of particular Staff witnesses, and nothing in our telephone conference call changed that for this case. (We had hoped, however, to avoid this procedure if possible.) Moreover, the Applicant had indicated that it will take the deposition of Mr. Gilray in Bethesda during the next two or three weeks; to take two other depositions at that time would not seem to inconvenience it unduly. For that reason, we direct the Staff to make available Messrs. Kane and Thornburg at that time in Bethesda (if sought by the Applicant). Otherwise, the depositions of Messrs. Kane and Thornburg shall be taken in Bethesda at a time mutually agreed by the Staff and Applicant. The depositions of Messrs. Naidu and Fiorelli shall be taken in Glen Ellyn, Illinois at a time mutually agreed by the Applicant and Staff.

The depositions shall be limited to the subjects indicated. We urge the parties to attempt to work out any differences of opinion amicably; if they cannot do so, they can ask us to resolve disputes. In doing so, we will be guided by our desire to permit parties to obtain all material information which they may need to develop their cases. We add, however, that we will not countenance open-ended interrogation of Staff witnesses.

B. On February 5, 1981, the Applicant initiated a telephone conference call to resolve a dispute which had arisen between it and the NRC Staff concerning the scheduling of the depositions which we had ordered. Participating were Mr. Ronald Zamarin for the Applicant and Mr. William Paton for the Staff. The Chairman was the sole Board member who was involved, since the other members were unavailable at the time. (See 10 CFR §2.721(d).)

The Applicant asserted that the NRC Staff was refusing to agree upon definite schedules for the depositions and had offered no explanation. The Applicant sought a definite schedule so that it could arrange its own schedule and make travel plans as necessary. The Staff explained (although it apparently had not previously informed the Applicant) that it was planning to file a motion for reconsideration of the earlier Board order.

The Board Chairman ruled that the Staff should schedule the depositions in question but that no depositions were to be taken until the Board had ruled on the reconsideration motion. The Board Chairman also stated that the Applicant need not respond to the Staff motion unless asked by the Board to do so, and that any response requested would be through the medium of a telephone conference call (given the expedited discovery schedule which had been contemplated by the Board's discovery order).[3]

C. On February 9, 1981, the Staff filed its "Motion for Reconsideration or Referral of Licensing Board's Rulings of January 29, 1981." In that document, the Staff sought reconsideration only of our ruling with respect to Mr. Thornburg. By limiting its motion in that respect, the Staff is leaving in effect our rulings compelling the depositions of Messrs. Naidu and Fiorelli and the further deposition of Mr. Kane. (The Staff has reserved the right to contest these rulings later on appeal, a course of action which will provide appellate review only after the depositions have taken place.) The Staff also asked us to refer our ruling to the Appeal Board should we determine to deny its reconsideration motion.

Both the Board Chairman and the Applicant received the Staff's motion the afternoon of February 9, 1981. Because the Board determined that the arguments of the Staff raised questions concerning certain aspects of our earlier order, we requested the Staff to arrange a telephone

[3] Dr. Cowan concurs with this ruling.

conference call on February 10, 1981 to discuss these questions. The Staff did so.

Judges Bechhoefer and Cowan participated in this call. Representing various parties were Messrs. Michael I. Miller and Ronald Zamarin for the Applicant, Mr. William Olmstead for the NRC Staff, Ms. Sharon Warren, *pro se,* and Mr. Wendell Marshall, for the Mapleton Intervenors.

In its reconsideration motion, the Staff challenged our earlier finding of "exceptional circumstances" with respect to Mr. Thornburg; it characterized the Applicant's attempt to depose Mr. Thornburg as a "fishing expedition" barred by 10 CFR Part 2, Appendix A, Part IV. The Staff also claimed that, under the regulations, the Applicant must first attempt to obtain information from the Staff through documents, next through interrogatories, and only if those attempts fail through depositions. The Staff would require a showing that *"no other individual"* made available by the Staff could provide the desire information "*and* that the information is *material*" (emphasis in original). In that connection, the Staff took the position that the only information possessed by Mr. Thornburg which might be material was subject to executive privilege and hence should not be discovered. The Staff also observed that the information as to which we found "exceptional circumstances" was known to other witnesses made available by the Staff; put another way, it asserted that "Consumers * * * [has] not established that *none* of the numerous witnesses made available to [it] had the desired information."

We will not at this point treat whether the Staff's understanding of the Rules of Practice accords with our own. For, subject to its undertaking to ascertain whether two other witnesses already designated by the Staff possess knowledge of the matters concerning which the Applicant wishes to inquire, the Applicant, in our view, has demonstrated "exceptional circumstances," in the context of the discovery arrangements being followed in this proceeding, to warrant the desposition of Mr. Thornburg.

With its motion to compel Mr. Thornburg's deposition, the Applicant supplied documents which indicated, it claimed, that during the period between November 28, 1979 and November 30, 1979 the Staff may have changed its opinion with respect to whether the modification order should be issued. Notes of a meeting on November 28, 1979 involving several ranking NRC employees who were engaged in resolving the soils settlement question indicate, according to the Applicant, a general consensus that CPC's "proposed fixes are such that, if they are implemented properly they should be adequate" (Shewmaker dep., exhibit 13, attached to Applicant's motion to compel). Among the persons who apparently attended that meeting were Messrs. Shewmaker, Hood, Keppler, and Rinaldi, all of whom had been made available for desposition, and Messrs. Fiorelli and

Thornburg, whose depositions we have ordered. Notwithstanding the alleged consensus, however, drafts of a proposed modification order circulated two days later (with the order itself issuing eight days after the meeting). The Applicant also produced a meeting log which indicated that Mr. Thornburg had met on November 28 and 29, respectively, with Mr. Case and Mr. Stello, the officials who signed the modification order. The Applicant wishes to discover any factual information communicated to Mr. Case or Mr. Stello which may have led to the modification order and which may indicate a shift in position of Staff members.

This information in our view is material—indeed essential—to a proper evaluation of the soils settlement question. As all parties seem to agree, the surfacing of differing professional opinions within the Staff (if any) will assist us in reaching an informed decision on this question. Mr. Thornburg appears to have information not possessed by others made available by the Staff. In order to confine the deposition to information demonstrated by the Applicant to be not otherwise available, we limited the subjects of the deposition. See p. 218 *infra*, and Tr. 701-702.

We also find that the Applicant has made sufficient attempts (except as described below) to obtain the information from other sources to warrant our finding of "exceptional circumstances" with respect to Mr. Thornburg (subject to procedural requirements hereinafter outlined). There is a public interest reason for completing discovery, as well as the entire proceeding, as expeditiously as possible—if only because the Applicant is free to continue plant construction in areas impacted by the soils settlement condition despite questions by the Staff as to whether the soils settlement questions have been adequately resolved. For that reason, the Applicant and Staff have informally agreed to utilize depositions as the primary discovery methodology. We agree that the use of depositions in this context is desirable and, hence, we decline to require that the Applicant first attempt to obtain the information through documents or interrogatories.[4]

The Applicant did attempt to obtain the requested information from witnesses produced by the Staff who had attended the November 28, 1979 meeting and who might have had knowledge of facts later communicated to Mr. Case or Mr. Stello (Tr. 540-41). The Applicant and Staff disagree on whether Mr. Darl Hood, the Project Manager, was asked the proper questions on his deposition. And Mr. Fiorelli has not yet been deposed.[5] As

[4]It appears, however, that the Applicant has sought to obtain certain documents from the Staff; we express no opinion whether its requests were specific enough to have obtained documents (if any) containing the information sought from Mr. Thornburg.

[5]His deposition is currently scheduled for February 17, 1981. See Notice of Deposition dated February 3, 1981. We were advised in the February 10, 1981 conference call that Mr. Thornburg's deposition is currently scheduled for February 20, 1981.

a condition for the deposition of Mr. Thornburg, and consistent with the scheme in the NRC Rules of Practice, we have modified our earlier order to require the Applicant first to question Messrs. Fiorelli and Hood about the matters on which it seeks to question Mr. Thornburg; only if they cannot respond properly to the Applicant's questions is Mr. Thornburg to be made available.

With respect to the Staff's substantive objections, the potentially privileged nature of the information sought by the Applicant (*i.e.*, deliberations leading to the modification order) was the primary reason we called for responses (by telephone) to the Staff's reconsideration motion. The Staff cites *Consumers Power Co.* (Palisades Nuclear Power Facility), ALJ-80-1, 12 NRC 117 (1980) as authority for the proposition that interrogation concerning the deliberative processes of the NRC Staff is privileged from discovery, under the executive privilege. However, we understand that decision as holding only that, in that proceeding, the party seeking discovery had not demonstrated "exceptional circumstances" and could not obtain the requested information absent a showing that it had done so. In particular, that party had not demonstrated the safety significance of the data sought. 12 NRC at 126. The ruling also left open the possibility that the data might eventually have to be revealed. *Id.* at 128.

More pertinent, in our view, is the decision of the Commission in *Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), CLI-74-16, 7 AEC 313 (1974). There, the Licensing Board ordered production of portions of documents which included information bearing upon the deliberative and policy making functions of the Advisory Committee on Reactor Safeguards (ACRS), on the basis that disclosure of the information was "necessary to a proper decision in this particular proceeding" and "the information is not reasonably obtainable from another source, *in view of the need to expedite the proceeding* and the stipulated tight schedule for discovery." *Id.* at 314, emphasis supplied. The Commission approved this release of information, citing in addition the following factors:

> This proceeding involves a safety issue * * * not discovered until after issuance of the construction permits * * * . This potential problem required issuance of an Order to Show Cause. Moreover, there were allegations—sufficient to warrant an investigation—that the licensee had intentionally withheld [pertinent] information * * * from the agency for several years. Under these circumstances, we [believe] it imperative that all information concerning [the question at issue] be made public. The policy considerations underlying the Committee's decision to delete deliberative passages from its records should not be

permitted to prevent disclosure of the safety-related information contained in the records here in issue.

Id. at 315 (fns. omitted).

We note that the Applicant has disclaimed any intent of inquiring into deliberative information (Tr. 544-545; also, telephone conference on February 10, 1981). To clarify our earlier ruling, we have limited the scope of Mr. Thornburg's deposition (insofar as communications with Messrs. Case or Stello are involved) to facts; recommendations are excluded. Taking that limitation into account, and given the similarity of circumstances between this proceeding and the situation described in *North Anna*, we hold that the Applicant has demonstrated sufficient "exceptional circumstances" to warrant the deposition of Mr. Thornburg (subject to the preliminary procedural requirements we have imposed).

D. The Staff asked us to refer this ruling to the Appeal Board, on the basis that later appeal would not correct the injury it would sustain if deliberative material were revealed. We agreed to do so. *Cf. Kansas Gas and Electric Co.* (Wolf Creek Nuclear Generating Station, Unit No. 1), ALAB-327, 3 NRC 408, 413 (1976).[6] Although we strongly support the conclusion we have reached with respect to Mr. Thornburg, we also recognize that, should the Staff's assessment of the situation be accepted, our ruling might have public interest implications, within the contemplation of 10 CFR §2.730(f). We denied the request to delay the deposition of Mr. Thornburg until after the Appeal Board ruling; the Appeal Board can, of course, stay our order if it believes that course of action is appropriate.

We note that the only ruling we are referring to the Appeal Board is that with respect to Mr. Thornburg. We perceive no persuasive reasons for early review of the other rulings included herein.

For the reasons stated, and subject to the limitations which we have described, the Applicant's motions to compel the depositions of Messrs. Naidu, Fiorelli, and Thornburg are *granted*. The Staff's motion for a protective order with respect to Mr. Kane is *denied* (subject to the limitations on questioning which we have described). The Staff's motion for reconsideration of our ruling concerning Mr. Thornburg is *denied*, subject to the Applicant's taking the additional procedural steps outlined in this opinion. The Staff's motion to refer our ruling with respect to Mr. Thornburg to the Appeal Board is *granted*.

[6] See, generally, *Public Service Co. of Indiana, Inc.* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977).

It is so ordered this 12th day of February, 1981.

FOR THE ATOMIC SAFTEY
AND LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Judge Linenburger took no part in the consideration or disposition of the matters dealt with in Sections B, C and D of this opinion.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Elizabeth S. Bowers, Chairman
Glenn O. Bright
Dr. Jerry R. Kline

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

PACIFIC GAS & ELECTRIC COMPANY
(Diablo Canyon Nuclear Power Plant, Units 1 & 2) **February 13, 1981**

In a prehearing conference order in a proceeding to consider a motion by applicant for authority to load fuel and conduct low power testing (pursuant to 10 CFR 50.57(c)), the Licensing Board: (1) confirms its previous rulings denying motions by the Governor of California seeking reconsideration of the Board's earlier denial of his request to stay the proceeding pending the preparation of an EIS or EIA dealing with fuel loading, testing, and low power operation and for an order directing the staff to prepare either an EIS or EIA; (2) describes the criteria under which it will admit contentions for litigation in the proceeding in light of NUREG-0737 (the Commission's revised TMI-related requirements for new operating licenses) and CLI-80-42 (the Commission's revised policy statement for implementation of the TMI-related requirements into the licensing process); (3) denies a request by an intervenor group that the Board certify to the Commission the question of the criteria for deciding the admissibility of contentions in this proceeding; (4) rules on the contentions submitted by the intervenor group; (5) rules on the subjects on which the Governor of California may participate in the proceeding as the representative of an interested state under 10 CFR 2.715(c); and (6) adopts the schedule for prehearing activities stipulated by the parties.

NEPA: NEED FOR ENVIRONMENTAL REVIEW

Where the environmental impacts of full-term, full-power operation have already been evaluated in an environmental impact statement, a licensing action for limited operation that would result in lesser impacts need not be accompanied by either an additional impact statement or an impact appraisal.

RULES OF PRACTICE: REOPENING OF PROCEEDINGS

New regulatory requirements establish good cause for reopening a record or admitting new contentions on matters related to the new requirement.

RULES OF PRACTICE: PARTICIPATION BY AN INTERESTED STATE

A representative of an interested state participating under 10 CFR 2.715(c) is not required to submit contentions of his own in order to participate in a proceeding, but if the representative wishes to raise specific issues not otherwise accepted by a board, he must comply with the requirements of 10 CFR 2.714(b) for acceptable contentions. *Gulf State Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).

PREHEARING CONFERENCE ORDER

At the time of the Three Mile Island (TMI) accident, the record in this proceeding was complete. The occurrence of that accident prompted a motion (from the Joint Intervenors) on May 29, 1979, to reopen the record. The Staff urged that the Board defer ruling on that motion until the Staff could investigate the accident and report its conclusions as to the implications for the Diablo facility to the Board and the parties. In an Order of June 5, 1979, the Board granted the Staff request.

On June 20, 1980, the Nuclear Regulatory Commission (NRC) issued a "Statement of Policy for Further Commission Guidance for Power Reactor Operating Licenses," 45 Fed. Reg. 41738. That statement adopted as both necessary and sufficient for responding to the TMI accident insofar as new operating licenses are concerned the requirements contained in NUREG-0694, "TMI-Related Requirements for New Operating Licenses."

The existence of this guidance from the Commission prompted the filing by Applicant of a motion seeking authority to load fuel and conduct low power testing (a so-called §50.57(c) motion). The motion seeks an operating license authorizing (i) loading of fuel; (ii) proceeding to initial criticality; (iii) performing startup testing at zero power; and (iv) testing at power levels not in excess of 5% of rated power with respect to each unit.

The Applicant's motion prompted further proceedings which are the subject of this prehearing conference order. To some extent, the Joint Intervenors' motion to reopen is also necessarily involved in these proceedings and will be dealt with herein as appropriate.

Applicant's motion was filed on July 14, 1980. On August 4, 1980, Joint Intervenors responded, asserting that prior to the grant of any such license outstanding issues pertaining to seismic design, security planning, quality assurance, and emergency planning had to be resolved. Further, Joint Intervenors asserted the need for a hearing at which they would contest Applicant's conclusions with respect to fuel loading and low power testing.

Also on August 4, Governor Brown filed in opposition to Applicant's motion, asserting that the motion did not comply with the Rules of Practice.

On August 6, the Staff responded to the motion asserting, *inter alia,* that the motion application for 50.57(c) license appeared to be adequate and suggested that the Board should proceed. Attached to the Staff's response was Supplement 10 to the SER which served as its evaluation of the impact of TMI on the sought-for license.

After calling for further response from the parties, the Board issued an Order accepting the Applicant's motion as sufficiently complete to commence the proceeding and setting October 27, 1980, as the date for the filing of contentions. This date was subsequently adjusted to December 3, 1980 because of the parties involvement with Appeal Board matters. The Board also approved the Staff's identification of the issues remaining in the proceeding on which Board findings were still required and concurred in the Staff's judgment that a decision on Joint Intervenors' motion to reopen the record on so-called "Class 9" accidents should await the conclusion of proceedings on the seismic issue currently underway before an Appeal Board.

In compliance with the Board's Order, Joint Intervenors filed contentions and the Governor filed a list of "subjects" on which he wishes to participate. The Applicant and Staff have filed responses, and a two-day prehearing conference was held on January 28 and 29, 1981, in Bethesda, Maryland, in which the contentions and "subjects" were discussed.

Before dealing in detail with the positions of the parties with regard to the contentions and "subjects," at the prehearing conference a related matter was considered. On December 8, 1980, the Governor filed a motion to stay the proceeding pending preparation of an environmental impact statement (or alternatively an appraisal) dealing with fuel loading, testing, and low power operation. This motion was supported by Joint Intervenors on December 18 and opposed by Applicant and Staff on December 23, since the motion did not even address the criteria for a stay. The motion was argued at the prehearing conference and orally denied by the Board on the basis that the Governor did not address the criteria in 10 CFR 2.788(e) which must be met in order for the motion to be granted. The Board also stated the motion would not prevail on the merits since an EIS was issued and a PID on environmental matters was issued. (Tr. 33-35) Subsequently, the Governor orally moved for reconsideration and requested the Board to direct the Staff to prepare an environmental impact appraisal in order to determine whether an environmental impact statement is necessary under 10 CFR 51.5(b)(3), 51.5(c)(1), 51.7 and 51.5(c)(1) prior to issuance of any fuel loading and low power testing license. The motion was denied and the Board stated the rationale would be detailed in its Order. While these motions seek somewhat different results, the Governor's rationale in support of them and the Applicant's and Staff's rationale in opposition are essentially identical. So are the Board's rulings. Therefore, both motions will be discussed together.

The Governor relies for support on the provisions of 10 CFR §§51.5(b) and (c). Subsection (b) of this section lists certain licensing actions which may or may not require an environmental impact statement. One of these is "[i]ssuance of a license to operate a power reactor...at less than full power...." (§51.5(b)(3).) Subsection (c) states that in the event that an environmental impact statement is not prepared, a negative declaration and environmental impact appraisal will be prepared, unless the Commission determines otherwise, with respect to the licensing action listed in subsection (b). The Governor seeks an order directing the Staff to prepare an environmental impact appraisal as a first step in determining whether an impact statement must be undertaken. The motion to stay sought a halt in these proceedings pending the preparation of the statement or, alternatively, of the appraisal. The Governor cites several cases for the proposition that NEPA requires agencies to take a hard look at environmental matters.

Both Applicant and Staff raised procedural objections to the motion to stay the proceedings, the principal of which was the lack of any showing having been made under §2.788.

Further, both have pointed to the fact that an environmental impact statement and a supplement thereto have been prepared in regard to the full-term full-power operation. Hearings have been held on the environmental issues and a Partial Initial Decision issued. (LBP-78-19, 7 NRC 989 [1978].) The Applicant and Staff rely on *Maine Yankee Atomic Power Company* (Maine Yankee Atomic Power Station) ALAB-161, 6 AEC 1003 [1973]; *aff'd sub nom., Citizens for Safe Power, Inc. v. NRC*, 524 F.2d 1291

(D.C. Cir. 1975) for the proposition that, in this situation, there is no need to consider the environmental impact of something less than full-power, full-term operation. The Staff also cites *Portland General Electric Co.* (Trojan Nuclear Plant) LBP-78-40, 8 NRC 717, at 744 (1978); *aff'd:* ALAB- 534, 9 NRC 287 (1979) for the same proposition. Additionally, the Staff points out that, absent some showing that the §50.57(c) license would entail some impacts which were not considered in the earlier environmental impact statement, supplement thereto, and hearings and decision thereon, there is no need to undertake a fresh environmental study. The latter, obviously, would only rehash earlier considerations. For this proposition, the Staff cites *Georgia Power Company* (Vogtle Units 1 & 2), ALAB-291, 2 NRC 404 (1975); *Detroit Edison Company* (Enrico Fermi Unit 2), LBP-78-11, 7 NRC 381 at 393 (1978); and *Northern States Power Company* (Prairie Island Units 1 & 2), ALAB-455, 7 NRC 41 at 46 n. 4 (1978).

The Governor has assumed and the Applicant and Staff have not challenged the proposition that §51.5(b)(3) includes the license here sought. Section 51.5(b)(3) includes licenses to *operate* at less than full power, while Applicant seeks a license to, *inter alia, test* at less than 5% of rated power. The Board believes that a meaningful distinction may exist between testing and operation which would raise the question whether §51.5(b)(3) applies to this proceeding.

Be that as it may, following the assumption that §51.5(b)(3) is applicable, the Board notes that the Staff has correctly stated the law. The Governor's attempt to postulate a situation not covered in the earlier environmental proceedings (issuance of §50.57(c) license, followed by denial of a full-term, full-power license) simply does not hold water. As pointed out in *Maine Yankee, supra,* any licensee faces the possibility of restriction or cancellation of his license as a result of regulatory developments. Clearly the environmental impacts of full-term, full-power operation are greater than the impacts of the limited testing here sought. To consider these limited impacts after the comprehensive review already undertaken would serve no useful purpose.

Consequently, it follows that both the Governor's motions must be denied; the motion to stay because the Governor cannot make the required showing that he is likely to prevail on the merits, etc.; and the oral motion to require preparation of an environmental impact appraisal because the Governor has not prevailed on the merits.

Next, it is necessary to address the positions taken by the parties with respect to the standards to be employed in determining which "contentions" and "subjects" are admissible. It would be an understatement to say that the discussion of this subject at the prehearing conference was characterized by some confusion. Nonetheless, the Board has carefully reviewed the transcript and has set down the positions of the parties as it understands them.

Joint Intervenors' position is most easily understood. The Joint Intervenors maintain that all contentions which were timely filed (by December 3) and which have a nexus to the application for the testing license are admissible. Contentions, of course, must meet the specificity requirements of 10 CFR §2.714 (Tr. 68, 82-84). Joint Intervenors base their position on their reading of the Commission's "Further Commission Guidance for Power Reactor Operating Licenses: Revised Statement of Policy" of December 18, 1980 (45 Fed. Reg. 85236, Dec. 24, 1980). Joint Intervenors believe that the fact that the revised policy statement removed the limitation in the policy statement as to litigation of the sufficiency of additional regulatory requirements (those which constitute new requirements as opposed to those which constitute refinement of existing regulations) means that contentions may propose additional requirements beyond those addressed in NUREG-0737 (Tr. 340). Joint Intervenors take the position that their proposed contentions fall into two categories; they propose issues over and above those issues contained in NUREG-0737 and challenge the sufficiency of issues addressed in NUREG-0737.

Applicant's position, as stated in its response to contentions and subjects of December 18, is clear. Applicant believes that the revised policy statement, reiterating as it does the traditional standards for reopening records and admitting late contentions, does not provide any authority to deviate from those standards. Thus, absent a showing of good cause under the applicable standard, a showing which intervenors have not attempted to make, no contentions are admissible. At the prehearing conference Applicant took the position that the paragraph at the bottom of page 8 of the policy statement prohibits new contentions.[1]

Staff's position as set forth in the transcript of the prehearing conference adopts a position not far from Applicant's. Staff agrees that good cause must be shown in order to reopen the record or admit a new contention at this stage. Staff correctly points out that the policy statement is not a rule and that therefore preexisting rules must be followed. (Tr. 89) Staff parts company with Applicant, however, in that it views the policy statement and NUREG-0737 as constituting good cause to reopen the record on *preexisting contentions* impacted by NUREG-0737 as meeting the "nexus" requirement. (Tr. 89) Staff does not similarly view NUREG-0737 as constituting good cause for filing new contentions based on its

[1]"The Commission believes that where the time for filing contentions has expired in a given case, no new TMI-related contentions should be accepted absent a showing of good cause and balancing of the factors in 10 CFR 2.714(a)(1). The Commission expects adherence to its regulations in this regard."

requirements. Staff's reasons for this dichotomy are not entirely clear. (Tr. 91, 93-94)

Governor Brown's position is complicated by the fact that he is participating under 10 CFR §2.715(c) as opposed to §2.714, and by the timing of his entrance into the proceeding after the record was complete. The Governor's position is basically the same as the intervenors: he may participate on any subject which he timely filed (by December 3) and which relates to the testing license application. The Governor stipulates that his "subjects" must meet the specificity and bases requirements of 10 CFR §2.714. (Tr. 117-8) Applicant maintains that, pursuant to §2.715(c), the Governor may only participate on issues raised by the parties or by the Board and may not raise issues on his own. (Applicants response of December 18) The Governor, needless to say, takes sharp issue with this position. (Tr. 111-4) Staff's position with respect to the Governor appears to be the same as its position with respect to the Intervenors. That position, however, has a much more dramatic effect on the Governor because he did not participate when the original record was compiled and hence cannot reopen the record on matters which concern him. (Tr. 118-9) Thus the Governor would be limited to participating on any intervenor contentions and Board questions admitted, unless, in the Staff's view, he can make a showing of good cause to admit a new contention at this time.

While these are interesting arguments, we have found it unnecessary to confront them. As set forth below, we have viewed the Governor's "subjects" in the same light as contentions put forward by Joint Intervenors in those instances where an admitted contention did not exist.

With this background, it is appropriate to set forth the Board's rulings with respect to the above matters, followed by rulings on specific contentions. Because of the nature of the application here in question, this discussion must begin with 10 CFR §50.57(c).

Section 50.57(c) provides that, in any contested proceeding on an operating license application, the Applicant may request a "...license authorizing low power testing (operation of not more than 1 percent of full power for the purpose of testing the facility) and further operations short of full power operation." The presiding officer is to act on the motion "...with due regard for the rights of the parties..., including the right of any party to be heard to the extent that his contentions are relevant to the activity to be authorized." To the extent that the motion is contested, the presiding officer is to make findings of fact and conclusions of law. Findings and conclusions on matters not in contest are to be made by the Director of Nuclear Reactor Regulation.

Historically, §50.57(c) motions have usually been made prior to the closing of the record in operating license proceedings, but after the completion of the record on any contentions which are relevant to the sought-for testing license. This timing permitted the presiding officers to make the necessary findings and conclusions with respect to the testing license prior to the completion of the record on all contentions.

For purposes of the §50.57(c) motion, the contentions were those previously allowed in the proceeding. Contentions were considered "relevant" to the motion to the extent that they needed to be resolved prior to criticality. Thus, for example, a contention which asserted that the control rod drives were defective would have to be heard and decided prior to the grant of a testing license. To the extent that matters not raised by contentions were "relevant" to the motion, §50.57(c) contemplates that the Director of Nuclear Reactor Regulation would make the necessary findings. The filing of the motion was not deemed to provide an opportunity to file new contentions. Acceptance of new contentions remained governed by the provisions of §2.714.

Some recent developments must be taken into consideration against this background. An Appeal Board has laid down rules under which unresolved safety issues are to be considered (in the absence of controversy) in construction permit cases (*Gulf States Utilities Co.* [River Bend Station, Units 1 & 2] ALAB-444, 6 NRC 760 at 775 [1977]) and another Appeal Board has applied these rules, to a limited extent, to operating license cases (*Virginia Electric & Power Co.* [North Anna Nuclear Power Station, Units 1 & 2] ALAB-491, 8 NRC 245, 248 [1978]). More importantly, the Commission has adopted measures it considers both necessary and sufficient to adequately protect the public health and safety for new operating licenses (NUREG-0737) along with a revised policy statement to govern consideration of these measures in licensing proceedings. Further, the Commission has recently adopted new rules governing emergency planning.

These developments must be considered in passing on the relevance of contentions to the motion for a testing license.

NUREG-0737 and the rule on emergency planning constitute new regulatory requirements. New regulatory requirements have always been viewed as establishing good cause for reopening a record or admitting new contentions. The Board does not agree with the Staff that there is a basis for treating NUREG-0737 as establishing good cause to reopen the record on old contentions while reaching an opposite conclusion with respect to the filing of new contentions. On the contrary, the whole purpose of the revised policy statement is to open the door to litigation of all NUREG-0737 requirements. If NUREG-0737 is not to constitute good cause for both reopening the record and filing new contentions, the revised policy statement becomes largely meaningless. The Board interprets the "nexus" requirement as nexus to Diablo Canyon facility not "nexus" to a contention previously admitted in this proceeding. Further, the appeal board's *North Anna* ruling means that we cannot totally leave to the Staff for resolution those items which are not clearly contemplated by a relevant contention.

Applying the above to the instant proceeding, the Board will:

1. Make findings on all relevant preexisting contentions if no findings have been made previously.
2. Reopen the record on all relevant preexistng contentions to the extent necessary to properly take into account NUREG-0737 and the new rule on emergency planning.
3. Admit new relevant contentions with respect to the new rule on emergency planning and NUREG-0737. With respect to NUREG-0737, the Board will:

   a. deny any contention which is not directly related to NUREG-0737 requirements. Contrary to Joint Intervenors view, we believe the Commission's intent as set forth in the policy statement was not changed by the subsequent revision. Both the policy statement (p. 6) and the revised policy statement (p. 7) contain similar paragraphs which set forth three reasons why NUREG-0694 as clarified by NUREG-0737 should be the principal basis for consideration of the new requirements in adjudicatory hearings. These are: *first,* the effort expended by the Staff and Commission to deal with a large number of issues (the statement notes that this process cannot be duplicated in adjudicatory hearings); *second,* the lack of NRC resources to litigate the Action Plan in individual proceedings; and *third,* the fact that many decisions involve policy issues better dealt with through less formal means than adjudication. Further, under the heading "Commission Decision" on page 6 of the revised policy statement, the following appears:

   > Based upon its extensive review and consideration of the issues arising as a result of the Three Mile Island accident — a review that is still continuing — the Commission has concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0737 can provide a basis for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured by the NRC Staff against the regulations, as augmented by these requirements.[9] In general, the remaining items of the Action Plan should be addressed through the normal process for

development and adoption of new requirements rather than through immediate imposition on pending applications.

[9] Consideration of applications for an operating license should include the entire list of requirements unless an Applicant specifically requests an operating license with limited authorization (e.g., fuel loading and low-power testing).

A similar statement appears at page 5 of the policy statement. In view of the above, the Board does not believe it reasonable to interpret the provision permitting the challenge of the sufficiency of new regulatory requirements as permitting the addition of requirements not contained in NUREG-0737.

b. admit contentions which are based on category one requirements (those which refine existing regulations). These contentions may challenge both the necessity and sufficiency of the refinement within the limits imposed by the regulation; and

c. admit contentions which are based on category two requirements (those which supplement existing regulations). Similarly, these contentions may challenge both the necessity and sufficiency of a requirement. In considering these contentions, the Board will pay particular attention to the nexus of the contention to the TMI accident, the significance of the issue raised by the contention, and the differences in the rationale underlying the contention and the NUREG-0737 requirement; and

4. Require the Staff to place on the record its conclusions regarding any issues which the Board, *sua sponte,* considers relevant and significant to the instant motion.

The Joint Intervenors requested the Board to certify the following question to the Commission:

"What requirements, other than relevancy to low-power operation, sufficient specificity and an adequate statement of the basis for the contention must be met for a contention to be admitted for litigation in this period." (Tr. 331)

The Board has interpreted the Commission's Revised Policy Statement and applicable regulations more in support of Joint Intervenors position that the position of either Applicant or Staff. We have accepted NUREG-0737 as good cause for admitting new contentions if there is nexus to Diablo and if they are significant. While we do not accept Joint Intervenors position that the sufficiency of NUREG-0737 can be challenged on matters not included, our interpretation opens this proceeding to a wide range of Joint Intervenors contentions. In light of the provisions of the Revised Policy Statement discussed above, we have determined that a sufficient reason does not exist to certify this question to the Commission and we decline to certify.

The Board notes that neither the Governor nor the Joint Intervenors sought to establish good cause for admitting new contentions or reopening the record on old contentions aside from their reliance on NUREG-0737. Therefore, the contention and subjects are viewed only in the context of NUREG-0737.

A. Joint Intervenors Contentions

> **Contention 1.** No final decision has been rendered by the Commission as to the Applicant's compliance at Diablo Canyon with 10 CFR Part 100 Appendix A regarding seismic safety. Because of the exceptional nature of the seismic danger associated with the Diablo Canyon facility such a definitive determination by the Commission must be issued prior to fuel loading.

> **Contention 2.** No final decision has been rendered by the Commission as to the Applicant's Compliance at Diablo Canyon with 10 CFR Part 73, regarding physical protection of nuclear plants and materials. Such a definitive determination by the Commission must be issued prior to fuel loading.

These Contentions are legal arguments advanced by Joint Intervenors to the effect that there must be a final Commission decision with respect to seismic and security matters prior to fuel loading. Both of these matters are currently the subject of further proceedings before the Appeal Board.

At the prehearing conference, the parties agreed to discuss the possibility of a stipulation relating to these contentions and report their progress to the Board. (Tr. 168-170) No report was forthcoming.

Because these contentions do not present any factual issues, the Board will defer any further action on them until the Initial Decision. Therefore, the parties are requested to advise the Board of their respective positions on these contentions (or of any agreement they have been able to reach) in their proposed findings submitted following closing of the record, taking into account any Appeal Board decisions which may have been rendered in the interim.

> **Contention 3.** The Applicant has failed to demonstrate compliance at Diablo Canyon with 10 CFR Part 50 Appendix B, regarding quality assurance.

Joint Intervenors did not take advantage of an opportunity to be heard on quality assurance matters in hearings raised by the Board on October 18-19, 1977. They have not demonstrated in their filings or oral argument a specific relationship between this contention and the additional requirements for fuel loading and low power testing arising from the accident at TMI as specified by the Commission in NUREG-0737. (Tr. 178) For these reasons and in accordance with the Commission Revised Statement of Policy of December 18, 1980 (at page 8) contention 3 is *denied.*

> **Contention 4.** Numerous studies arising out of the accident of TMI recognized the necessity of upgrading emergency response planning. Based upon these studies, the Commission has promulgated revised emergency planning regulations effective November 3, 1980. The Applicant has failed to demonstrate that the combined Applicant, state, and local emergency response plans for Diablo Canyon comply with those revised regulations ("Final Regulations on Emergency Planning," 45 Fed. Reg. 55402 (August 19, 1980)).

> **Contention 5.** The Applicant has failed to demonstrate that the combined Applicant, state and local emergency response plans for Diablo Canyon comply with the requirements of Sections III.A.1.1 and III.A.1.2 of NUREG-0694.

The Board has stated that it will admit new relevant contentions with respect to the new rule on emergency planning and NUREG-0737. Contention 4 specifically identifies requirements of the new rule on emergency planning which must be complied with (new Appendix E to Part 50). Contention 5 identifies requirements of NUREG-0694 (which was later issued and approved by the Commission as NUREG-0737 with changes and clarification) which must be complied with prior to the issuance of a license for fuel loading and low power testing. The requirements are stated in NUREG-0737 Enclosure 2, however, the text gives no additional clarification for Items. III.A.1.1. and III.A.1.2.) These contentions are relevant and specific to matters which must be resolved prior to issuance of the requested license. Contentions 4 and 5 are, therefore, *admitted* insofar as they pertain to issues related to fuel loading and low power testing.

> **Contention 6.** The Applicant has failed to demonstrate that the containment at Diablo Canyon can withstand pressures resulting from the combustion of hydrogen likely to be generated by the reaction of zirconium cladding with water during a loss of coolant accident at the facility.

Joint Intervenors in oral argument pointed to requirement II.E.4.1 of NUREG-0737 which deals with dedicated hydrogen penetrations when called upon to show how contention 6 is related to new TMI requirements. They conceded, however, that this requirement does not specifically contain a requirement which "meets" contention 6. (Tr. 212) They argue instead that the NUREG-0737 requirement is insufficient. (Tr. 212) The Board interprets this as a demand for a new item not now contained in NUREG-0737. The Board has stated that we would reject such contentions as being inconsistent with the Commission's Revised Statement of Policy.

Contention 6 is therefore *denied*.

> **Contention 7.** The Applicant has failed to address adequately safety considerations designated as high priority and/or high risk in Table B.2 of NUREG-0660 TMI Action Plan.

The Commission in its Revised Statement of Policy has decided that current operating license applications should be measured by the NRC Staff against the regulations as augmented by these requirements contained in NUREG-0737, not NUREG-0660. The Revised Statement of Policy states:

> "In general the remaining items of the Action Plan should be addressed through the normal process for development and adoption of new requirements rather than through immediate imposition on pending applications."

Items appearing in NUREG-0660 but not in NUREG-0737 are, therefore, not to be imposed on pending applications. Joint Intervenors assert, however, that under their right to challenge sufficiency of NUREG-0737 requirements 12 additional items taken from NUREG-0660 should be made a part of NUREG-0737. (Tr. 224) Little rationale for the adoption of the newly enumerated items was given in oral argument however.

Contention 7 is *denied*.

> **Contention 8.** The accident at TMI Unit 2 demonstrated that reliance on natural circulation to remove decay heat is inadequate. During the accident it was necessary to operate at least one reactor coolant pump to provide forced cooling of the fuel. However, the Applicant's testing program does not demonstrate a reliable method for forced cooling of the reactor in the event of a small loss-of-coolant accident ("LOCA") particularly with regard to two-phase flow and with voids such as occurred at TMI-2. This is a threat to health and safety and a violation of both General Design Criterion ("GDC") 34 and GDC 35 of 10 CFR Part 50 Appendix A.

In the prehearing conference Joint Intervenors asserted only a remote relationship between this contention and the augmented requirements for licensing contained in NUREG-0737. They asserted instead a right to go beyond the requirements of NUREG-0737 (Tr. 234) (i.e., to challenge their sufficiency under the Commission Revised Statement of Policy).

Therefore, this contention is *denied.*

> **Contention 9.** Using existing equipment at Diablo Canyon, there are three principal ways of providing forced cooling of the reactor: (1) the reactor coolant pumps; (2) the residual heat removal system; and (3) the emergency core cooling system in a "bleed and feed" mode. None of these methods meets the NRC's regulations applicable to systems important to safety and is sufficiently reliable to protect public health and safety.
>
> a. The reactor coolant pumps do not have an adequate on-site power supply (GDC 17), their controls do not meet IEEE 279 (10 CFR 50.55a(h)) and they are not adequately qualified (GDC 2 and 4).
> b. The residual heat removal system is incapable of being utilized at the design pressure of the primary system.
> c. The emergency core cooling system cannot be operated in the "bleed and feed" mode for the necessary period of time because of inadequate capacity and radiation shielding for the storage of the radioactive water bled from the primary coolant system.

In the prehearing conference Joint Intervenors asserted only a remote relationship between this contention and the augmented requirements for licensing contained in NUREG-0737. They assert instead a right to go beyond the requirements of NUREG-0737. (Tr. 234) (i.e., to challenge their sufficiency under the Commission's Revised Statement of Policy)

For these reasons this contention is *denied.*

> **Contention 10.** The Staff recognizes that pressurizer heaters and associated controls are necessary to maintain natural circulation at hot stand-by conditions. Therefore, this equipment should be classified as "components important to safety" and required to meet all applicable safety-grade design criteria, including but not limited to diversity (GDC 22), seismic and environmental qualification (GDC 2 and 4), automatic initiation (GDC 20), separation and independence (GDC 3 and 22), quality assurance (GDC 1), adequate, reliable on-site power supplies (GDC 17) and the single failure criterion. The Applicant's proposal to connect two out of four of the heater groups to the present on-site emergency power supplies does not provide an equivalent or acceptable level of protection.

Joint Intervenors point to item II.E.3.1 in enclosure 2 of NUREG-0737 which addresses emergency power for pressurizer heaters as a new TMI-related requirement justifying admission of this contention (Tr. 242) Item II.E.3.1 does, however, not require that pressurizer heaters be classified as "components important to safety" a fact conceded by intervenors (Tr. 242) Intervenors challenge the sufficiency of this requirement (Tr. 242) (i.e., that they ought to be so classified). We do not believe that they have sufficiently tied this contention to the requirements of NUREG-0737 for it to be admitted, nor has it been demonstrated what a bearing this has on fuel loading and low power testing at Diablo Canyon.

The contention is *denied.*

> **Contention 11.** The Applicant has proposed simply to add the pressurizer heaters to the on-site emergency power supplies. It has not been demonstrated that this will not degrade the capacity, capability and reliability of these power supplies in violation of GDC 17. Such a demonstration is required to assure protection of public health and safety.

Joint Intervenors cited item II.E.3.1 of NUREG-0737 as a new Commission requirement for licensing arising from the accident at TMI. (Tr. 242) Item II.E.3.1 deals specifically with requirements of emergency power supplies to pressurizer heaters. Its requriments must be met 4 months prior to issuance of the SER according to enclosure 2 of NUREG-0737 (p. 2-6). This contention is, therefore, relevant to this proceeding and specifically related to a new requirement for licensing. It is, therefore, *admitted.*

> **Contention 12.** Proper operation of power operated relief valves, associated block valves and the instruments and controls for these

valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria.

This contention does not specifically identify an item in NUREG-0737 which has not been complied with nor has a showing been made that any item is insufficient. The contention is therefore *denied*.

Contention 13. NRC regulations require instrumentation to monitor variables as appropriate to ensure adequate safety (GDC 13) and that the instrumentation shall directly measure the desired variable. IEEE 279, §4.8, as incorporated in 10 CFR 50.55a(h)), states that:

> "To the extent feasible and practical protection system inputs shall be derived from signals which are direct measures of the desired variables."

Diablo Canyon has no capability to directly measure the water level in the fuel assemblies. The absence of such instrumentation delayed recognition of a low-water level condition in the reactor for a long period of time. Nothing proposed by the Staff would require a direct measure of water level or provide an equivalent level of protection. The absence of such instrumentation poses a threat to public health and safety.

This contention raises an issue which is clearly TMI-related, and is included in NUREG-0737 (II.F.2) as an action item. As presented, the contention lacks specificity, as there is no argument among the parties that a water level indication will be required. During discussion of the contention (Tr. 258-262) it was revealed that the Intervenor's concern was that installation of the indicator would not be required until 1/1/82, rather than before fuel loading and low power testing. With that understanding the Board *accepts* contention #13 as a litigable issue.

Contention 14. 10 CFR 50.46 requires analysis of ECCS performance "for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered." For the spectrum of LOCAs, specific parameters are not to be exceeded. At TMI, certain of these were exceeded. For example, the peak cladding temperature exceeded 2200° fahrenheit (50.45(b)(1)), and more than 1% of the cladding reacted with water or steam to produce hydrogen (50.46(b)(3)). The measures proposed by the Staff address primarily the very specific case of a struck-open power operated relief valve.

However, any other small LOCA could lead to the same consequences. Additional analyses to show that there is adequate protection for the entire spectrum of small break locations for the Diablo Canyon design have not been performed. Therefore, there is no basis for finding compliance with 10 CFR 50.46 and GDC 35. None of the corrective actions to date have fully addressed the demonstrated inadequacy of protection against small LOCAs.

The contention appears to have a very tenuous relationship with NUREG-0737; specifically to I.C.1. I.C.1, however, appears to lead toward off-normal occurrence analysis with the view of developing procedures to be used in operator training, rather than ECCS performance, *per se.* In any event, 10 CFR 50.46 sets *limits* on clad temperature and oxidation, and does not lay down input parameters to be used in analysis. As phrased in the contention and further laid out in discussion (Tr. 262-268), the contention lacks the necessary basis and specificity to be accepted. The contention is therefore *denied.*

Contention 15. The accident at TMI-2 was substantially aggravated by the fact that the plant was operated with a safety system inoperable, to wit: two auxiliary feedwater system valves were closed which should have been open. The principal reason why this condition existed was that TMI does not have an adequate system to inform the operator that a safety system has been deliberately disabled. To adequately protect the health and safety of the public, a system meeting the Regulatory Position of Reg. Guide 1.47 or providing equivalent protection is required.

Review of the contention as presented and the pertinent discussion (Tr. 268-270) indicates that there exists a very fragile connection with the requirement of NUREG-0737 at best. In any event, a broad allegation that the requirements of NUREG-0737 are insufficient does not supply the requisite specificity to define an issue to be placed in litigation in this proceeding. The contention is *denied.*

Contention 16. The design of the safety systems at TMI was such that the operator could prevent the completion of a safety function which was initiated automatically; to wit: the operator could (and did) shut off the emergency core cooling system prematurely. This violated §4.16 of IEEE 279 as incorporated in 10 CFR 50.55(a)(h) which states:

> "The protection system shall be so designed that, once initiated, a protection system action shall go to completion."

> The Diablo Canyon design is similar to that at TMI and must be modified so that no operator action can prevent the completion of a safety function once initiated.

The Board could find no connection between this contention and the requirements of NUREG-0737. The contention is *denied.*

> **Contention 17.** The design of the hydrogen control system at TMI was based upon the assumption that the amount of fuel cladding that could react chemically to produce hydrogen would, under all circumstances, be limited to less than 5%. The accident demonstrated both that this assumption is not justified and that it is not conservative to assume anything less than the worst case. Therefore, the Diablo Canyon hydrogen control systems should be designed on the assumption that 100% of the cladding reacts to produce hydrogen.

This contention was considered in conjunction with contention 6. (Tr. 209-222). For the same reason set forth above the contention is *denied.*

> **Contention 18.** The TMI-2 accident demonstrated that the severity of the environment in which equipment important to safety must operate was underestimated and that equipment previously deemed to be environmentally qualified failed. One example was the pressurizer level instruments. The environmental qualification of safety-related equipment at TMI is deficient in three respects: (1) the parameters of the relevant accident environment have not been identified; (2) the length of time the equipment must operate in the environment has been underestimated; and (3) the methods used to qualify the equipment are not adequate to give reasonable assurances that the equipment will remain operable. Diablo Canyon should not be permitted to load fuel until all safety-related equipment has been demonstrated to be qualified to operate as required by GDC 4. The criteria for determining qualification should be those set forth in Regulatory Guide 1.89 or equivalent.

NUREG-0737, at II.B.2, considers added requirements for shielding against and qualification tests for the radiation to be expected in a TMI-2 situation. To this extent the contention appears to be related to a NUREG-0737 requirement. However, the stated contention, as well as the discussion which took place at the Prehearing Conference (Tr. 272-74) is totally lacking in any specific issues which might be litigated in this proceeding. Even the three defects in environmental qualifications at TMI were not shown to connect in any recognizable way with Diablo Canyon, and even if so alleged, are too diffuse to constitute a litigable issue. The contention is *denied.*

> **Contention 19.** Neither the Applicant nor the NRC Staff has presented an accurate assessment of the risks posed by operation of Diablo Canyon, contrary to the requirements of 10 CFR 51.20(a) and 51.20(d). The design of Diablo Canyon 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-2 falls within that classification. Therefore, there is not reasonable assurance that Diablo Canyon can be operated without endangering the health and safety of the public.

Without going to the merits of the contention, as presented, the Board will defer consideration of this issue until the Appeal Board has ruled on the Diablo Canyon seismic issue which is now before it.

> **Contention 20.** The TMI-2 accident demonstrated that there are systems and components presently classified as non-safety-related which can have an adverse effect on the integrity of the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity. This issue is discussed at length in Section 3.2, "System Design Requirements," of NUREG-0578, the TMI-2 Lessons Learned Task Force Report (Short Term). The following quote from page 18 of the report describes the problem:
>
> > There is another perspective on this question provided by the TMI-2 accident. At TMI-2, operational problems with the condensate purification system led to a loss of feedwater and initiated the sequence of events that eventually resulted in damage to the core. Several nonsafety systems were used at various times in the mitigation of the accident in ways not considered in the safety analysis; for example, long-term maintenance of core flow and cooling with the steam generators and the reactor coolant pumps. The present classification system does not adequately recognize either of these kinds of effects that nonsafety systems can have on the safety of the plant. Thus, requirements for nonsafety systems may be needed to reduce the frequency of occurrence of events that initiate or adversely affect transients and accidents, and other requirements may be needed to improve the current capability for use of nonsafety systems during transient or accident situations. In its work in this area, the Task Force will include

> a more realistic assessment of the interaction between operators and systems.

> The Staff proposes to study the problem further. This is not a sufficient answer. All systems and components which can either cause or aggravate an accident or can be called upon to mitigate an accident must be identified and classified as components important to safety and required to meet all safety-grade design criteria.

There is not cognizable relationship between this contention and the requirements in NUREG-0737, as confirmed by Intervenor (Tr. 280). The contention is *denied.*

> **Contention 21.** The accident at TMI-2 was caused or aggravated by factors which are the subject of Regulatory Guides not used in the design of TMI. For example, the absence of an automatic indication system as required by Regulatory Guide 1.47 contributed to operation of the plant with the auxiliary feedwater system completely disabled. The public health and safety require that this record demonstrate conformance with or document deviations from the Commission's regulations and each Regulatory Guide presently applicable to the plant.

The Intervenor has agreed that there is no NUREG-0737 requirement which is related to this contention (Tr. 284). *Denied.*

> **Contention 22.** Withdrawn (Tr. 286)

> **Contention 23.** The accident at TMI-2 was a multiple failure accident involving independent and dependent failures. The multiple failure sequences exceeded the single failure criterion utilized in the Diablo Canyon design basis accident assessment. Therefore, comprehensive studies of the interaction of nonsafety grade components, equipment, systems, and structures with safety systems and the effect of these interactions during normal operation, transients, and accidents need to be made by the Diablo Canyon Applicant in order to assure that the plant can be operated without endangering the health and safety of the public.[2]

This contention was considered to be on the same subject as contention 20. For the same reasons the contention is *denied.*

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[2]On February 11, 1981, the Joint Intervenors submitted two (2) documents referenced in the prehearing conference. The Board had prior knowledge of these documents.

Contention 24. Reactor coolant system relief and safety valves form part of the reactor coolant system pressure boundary. Appropriate qualification testing has not been done to verify the capabilities of these valves to function during normal, transient and accident conditions. In the absence of such testing and verification, compliance with GDC 1, 14, 15 and 30 cannot be found and public health and safety are endangered.

NUREG-0737, at II.D.1, sets out test schedules for relief valve, safety valve and block valve tests. The RV and SV tests must be completed before fuel load. However, the block valve tests completion schedule is for before fuel loading *or* 7/1/82, whichever is later. Intervenors believe that all these tests should be completed prior to fuel loading, and that the NUREG-0737 requirements are not sufficient in this manner. (Tr. 250-258) With this understanding by the Board, the contention is *accepted.*

Contention 25. Withdrawn (Tr. 286)

Contention 26. Withdrawn (Tr. 286)

Contention 27. Withdrawn (Tr. 286)

In the prehearing conference, at such times when the Applicant or Staff criticized a contention of the Joint Intervenors as inadequate, the Joint Intervenors would volunteer that they could improve the specificity of a contention after meeting with their technical consultants (e.g., Tr. 185 or 193). The Joint Intervenors had several months to develop their contentions. They are represented by knowledgeable, experienced counsel. The Board has ruled on the contentions as submitted and as clarified at the prehearing conference. The Joint Intervenors will not be granted additional time to revise and resubmit those contentions not admitted by the Board.

The Joint Intervenors in their filing of January 8, 1981 and in the prehearing conference (Tr. 116) want to adopt Governor Brown's subjects as their contentions. The only subjects admitted were bootstrapped to the Joint Intervenors contentions. There are no separately admitted subjects from Governor Brown. The question is academic.

B. Governor Brown's Subjects

Governor Brown's timely-filed petition to participate as the representative of an interested state under 10 CFR 2.715(c) set forth "subjects on which Governor Edmund G. Brown, Jr., intends to participate" in this proceeding. No contentions, *per se,* were presented. As a representative of an interested state participating under 10 CFR 2.715(c) Governor Brown is not required to submit contentions of his own, but is free to fully participate in the litigation of any contentions which are otherwise accepted by the Board. However, if the Governor wishes to raise specific issues not otherwise accepted by the Board he must comply with the requirements of 10 CFR 2.714(b) for acceptable contentions, just as any other party must. [See *Gulf State Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).] The Governor agrees to this proposition (Tr. 117-18). To determine the admissability of Governor Brown's subjects as issues in this proceeding (as limited by the Board's Order of October 2, 1980) they will be considered individually as contentions and subjected to the same tests as have been applied to the contentions submitted by Joint Intervenors.

Subjects 1 and 2. Withdrawn (Tr. 169-71)

Subject 3. Whether the emergency plans of PG&E, the State, and the local jurisdiction are satisfactory for issuance of the requested licenses.

A. Whether further steps, including those set forth in the NRC's Final Rule on Emergency Planning, 45 Fed. Reg. 55402 (August 19, 1980), must be accomplished before the licenses may be issued.

This subject is closely related to Joint Intervenor's contentions 4 and 5, which the Board has accepted. Governor Brown may thus participate in litigation of this issue.

Subject 4. Whether PG&E, as alleged in its Motion, has complied with or will comply with the requirements of NUREG-0694 prior to loading fuel (Motion, p. 2), including the following matters specified in the Safety Evaluation Report ("SER"), Supplement 10, which the NRC Staff has examined but which, as of publication of Supplement 10, were not complete:

(a) Adequacy of the training, experience and procedures for shift technical advisors. (SER, Suppl. 10, p. I.A.-2)
(b) Results of cold license examinations for the 21 candidates who were to take examinations in August 1980, and results of examinations for other licensed personnel. (*Id.* I.A.-6)
(c) Adequacy of procedures for accident mitigation and recovery. (*Id.* I.B-3)
(d) Adequacy of the reorganization of PG&E's operating organization for both routine and emergency operations and adequacy of PG&E's agreements with other organizations and utilities to pool resources in the event of an emergency. (*Id.*)

(e) Adequacy of PG&E's guidelines and procedures for emergency core cooling and small break LOCAs.
(f) Adequacy of PG&E's startup test procedures. (*Id.* I.C.-7)
(g) Adequacy of PG&E's measures to deal with human factors-related deficiencies (*Id.* IV. 1-2 and 3).

There is no issue presented here. All parties and the Board agree that these matters must be resolved before a license can issue. (Tr. 288-295) The Board and the parties should be kept informed on the status of Applicant's compliance. *Denied.*

Subject 5. Whether the seven tests proposed by PG&E in its Motion are a complete list of necessary tests.

A. Whether, in addition to the seven stated tests, there must be tests designed to demonstrate 2-phase natural circulation cooling capability that are representative of actual accident conditions.

This subject is not contained in the NUREG-0737 requirements, nor does it have the basis and specificity to qualify as a contention on its own. (Tr. 235-39). *Denied.*

Subject 6. Whether the activities sought by PG&E to be authorized under the licenses are "vital to demonstrate the effectiveness of the augmented reactor operation training program, improved management organization and operating procedures and controls, and certain changes in design and equipment implemented by PG&E to meet the NTOL Requirements." (Motion, p. 2)

This subject has no colorable relationship with this proceeding, because PG&E reasons for desiring to undertake the testing program at this time are irrelevant to Section 50.57(c) requirements. Further, the subject lacks any connection to NUREG-0737 and lacks sufficient basis and specificity to qualify as a contention on its own. (Tr. 295-307) *Denied.*

Subject 7. Whether the requested licenses and the activities authorized thereby "will provide meaningful technical information beyond that obtained in the normal startup test program." (Motion, p. 2)

This subject has no colorable relationship with this proceeding, because PG&E reasons for desiring to undertake the testing program at this time are irrelevant to Section 50.57(c) requirements. Further, the subject lacks any connection to NUREG-0737 and lacks sufficient basis and specificity to qualify as a contention on its own. (Tr. 295-307) *Denied.*

Subject 8. Whether the requested licenses and the activities authorized thereunder "will not pose an undue risk to the health and safety of the public" (Motion, p. 2), particularly since PG&E has not submitted safety analyses related to these activities and the NRC's risk assessment is unsupported by plant-specific analyses. (SER, Supp. 10, p. I.G.-5)

This subject lacks the requisite basis and specificity to qualify as a contention. (Tr. 295-307) *Denied.*

Subject 9. Whether-the requested licenses will result in radiation levels within the plant that would preclude or impede implementation of any later changes ordered by the NRC. (Ref. Motion, p. 2)

A. Whether these levels would expose workers to unacceptable exposures beyond ALARA levels.

This subject lacks the necessary basis and specificity to be accepted as a contention. (Tr. 308-09). *Denied.*

Subject 10. Whether the requested licenses and the activities authorized thereunder "will provide significant supplemental operator training." (Motion, p. 2).

A. Whether there are other means, including training on simulators and at other facilities, to obtain such supplemental operator training.

This subject has no colorable relationship with this proceeding, because PG&E reasons for desiring to undertake the testing program at this time are irrelevant to Section 50.57(c) requirements. Further, the subject lacks any connection to NUREG-0737 and lacks sufficient basis and specificity to qualify as a contention on its own. (Tr. 295- 307) *Denied.*

Subject 11. Whether early operation of Diablo Canyon Units 1 and 2 will contribute in any meaningful way toward the national objective of reducing dependence on imported oil and/or reduce in any meaningful way the risks or consequences to the public of inadequate generating resources and/or allow generation of power using less expensive fuels. (Ref. Motion, p. 3).

This subject has no colorable relationship with this proceeding, because PG&E reasons for desiring to undertake the testing program at this time are irrelevant to Section 50.57(c) requirements. Further, the subject lacks any connection to NUREG-0737 and lacks sufficient basis and specificity to qualify as a contention on its own. (Tr. 295-307) *Denied.*

> **Subject 12.** Whether the small break loss of coolant accident analyses and tests, including computer code verification, required for Westinghouse PWRs are sufficiently complete and accurate to permit issuance of the requested licenses.

This subject lacks the necessary basis and specificity to qualify as a contention and does not relate to an admitted contention. (Tr. 263-268) *Denied.*

> **Subject 13.** Whether the licenses should issue prior to installation of PG&E of a reliable and unambiguous method of measuring reactor vessel water level.

A. Whether PG&E's proposed system to measure water level in the reactor vessel is adequate for all conditions, including level swell, 2-phase flow, flow blockage and system dynamics. (SER, Supp. 10, p. II.F-9)

Although lacking the basis and specificity required for an allowable contention, the subject is essentially the same as Joint Intervenors contention 13 which the Board has accepted. Governor Brown may, therefore, participate in litigation of this issue in the form in which the Joint Intervenor's contention was accepted.

> **Subject 14.** Whether the licenses should issue prior to completion of qualification tests and analyses on relief and safety valves.

Although this subject lacks the specificity and basis necessary to being accepted as a contention, it is essentially the same as Joint Intervenor's Contention 24, and Governor Brown may participate in this litigation.

> **Subject 15.** Whether PG&E has established adequate procedures for dissemination of operating experience, obtained from operation of both Diablo Canyon and other nuclear plants, to PG&E personnel. (SER, Supp. 10, p. I.C.-7)

The subject lacks the necessary basis and specificity to qualify as a contention. (Tr. 309-12) *Denied.*

> **Subject 16.** Whether additional TMI Action Plan items should be completed before the licenses are issued, including:
>
> (a) NRC audit of emergency procedures (NUREG-0660, p. I.C.-7)
> (b) Withdrawn (Tr. 313)
> (c) Withdrawn (Tr. 313)

(d) Withdrawn (Tr. 313)
(e) Withdrawn (Tr. 313)
(f) Completion of upgraded training and qualification requirements. (*Id.* I.A.2-1)
(g) Completion of reevaluation of AFW reliability. (*Id.* II.E.1-1)

This subject, as stated, lacks the requisite basis and specificity to be accepted as a contention. Further, as discussed above, the Action Plan items are not appropriate for litigation unless contained in NUREG-0737. (Tr. 312-17) *Denied.*

Subject 17. Whether the NRC and PG&E have complied with all obligations under the National Environmental Policy Act, the regulations of the Council on Environmental Quality, and the NRC's regulations in Part 51.

A. Whether an environmental impact statement, or at the very minimum, an environmental impact appraisal must be prepared.

This subject has no relationship to any allowable issue in this proceeding and also lacks the basis and specificity necessary for it to be accepted as a contention. (Tr. 317-28. See, also, Board rulings on Governor Brown's Motion to Stay and Governor Brown's oral motion for ruling in Tr. 321-23 made previously in this Order). *Denied.*

At the prehearing conference, the parties stipulated to the following schedule (Tr. 367):

Assuming Board Order Issues	February 13, 1981
Close of Discovery	March 25, 1981
Motions for Summary Disposition	April 1, 1981
Prepared Direct Testimony Filed	May 8, 1981
Hearing Commences	May 19, 1981

The Board accepted the schedule. (Tr. 370)

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Elizabeth S. Bowers, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 13th day of February 1981.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Herbert Grossman, Chairman
Glenn O. Bright
Dr. Richard F. Cole

In the Matter of **Docket No. 50-367 CPA (Construction Permit Extension)**

NORTHERN INDIANA PUBLIC SERVICE COMPANY (Bailly Generating Station, Nuclear 1) **February 20, 1981**

The Licensing Board in this construction permit extension proceeding denies a motion seeking reconsideration of the Board's earlier denial of certain contentions and declines to certify or refer the denial to the Appeal Board or Commission.

RULES OF PRACTICE: ADMISSIBILITY OF CONTENTION (CONSTRUCTION PERMIT EXTENSION PROCEEDING)

Contentions that have no discernible relationship to the construction permit extension are inadmissible in a permit extension proceeding. For such contentions, a show-cause proceeding under 10 CFR 2.206 is the exclusive remedy. *Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear 1), ALAB-619, 12 NRC 558 (1980).

MEMORANDUM AND ORDER

(Denying Motions for Reconsideration and for Certification or Referral)

MEMORANDUM

On December 24, 1980, following the issuance of ALAB-619 (Nov. 20, 1980) in this proceeding, which further defined the scope of a construction permit extension proceeding originally delimited in *Indiana and Michigan Electric Co.* (Donald C. Cook Nuclear Plant, Units 1 and 2), ALAB-129, 6 AEC 414 (1973), this Board denied certain "newly-filed" contentions and contentions involving the merits of the "short pilings issue". The Board's denial of these contentions was based upon the two-pronged test for admitting contentions in an extension proceeding first enunciated in *Cook,* ALAB-129, *supra,* and reaffirmed in *Bailly,* ALAB-619, *supra* that matters which do not directly cause a delay in construction will be heard if they (1) arise from the reasons assigned for the construction permit extension and (2) cannot appropriately abide the operating license proceeding. We denied the newly-filed contentions because they failed the first test of not arising from the reasons assigned for the construction permit extension and denied the short pilings contentions because they failed the second test by being able to abide the event of the operating license proceeding.

The Porter County Chapter Intervenors (PCCI) and the State of Illinois intervenor object to our denial of these contentions and have filed motions requesting a reconsideration and reversal of our rulings or, in the alternative, an immediate certification or referral of these rulings to the Appeal Board. For the reasons stated below, we deny those motions.

Newly-Filed Contentions

PCCI and the State of Illinois base their objection to the Board's denial of the newly-filed contentions on the ground that the Board misinterpreted *Bailly,* ALAB-619, as requiring the two-pronged test in all construction permit extension proceedings. They construe ALAB-619 and *Cook,* ALAB-129, *supra,* as merely tailoring the two-pronged test to the particular facts of those respective cases.

While ALAB-619 may have disclaimed the two-pronged test as "an inflexible mold" (p. 22) for judging every contention in all permit extension proceedings, it clearly mandated the dismissal of any contention that has "no discernable relationship" to the permit extension (p. 23), as is the case with the newly-filed contentions. For those contentions, 10 C.F.R. §2.206 is the exclusive remedy. Ibid. Whether or not that formulation in ALAB-619 is merely a rephrasing of the first prong of the *Cook* test as we see it, intervenors have offered no substantial reason for reversing our ruling.

The State of Illinois' further observation (Motion, Fn., p. 3), that *Cook*, ALAB-129, *supra* did not confront the question of whether matters unrelated to the extension requests are within the scope of a good cause proceeding, is inapposite. As we made clear in our December 24, 1980 Memorandum and Order (pp. 3-4), we did not deny the newly-filed contentions on the basis of the *Cook* decision but, rather, on ALAB-619's "further guidance" with regard to the *Cook* determination

Short Pilings Issue

In objecting to the Board's denial of the short pilings issue, PCCI and the State of Illinois rely upon *Bailly*, ALAB-619, *supra; Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542 (1980); and *Mississippi Power and Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 AEC 423, 426 (1973); to assert that the Board improperly prejudged the "merits" of the isssue to determine that it could abide the operating license stage.

Intervenors' assertion is without foundation. They confuse their position on the merits of the issue, which the Board accepted for the purpose of determining admissibility, with their value judgment on whether this issue could abide the operating license stage, which the Board rejected. The latter determination is one that only the Board can make under *Cook*, ALAB-129, *supra*, and *Bailly*, ALAB-619, *supra*, subject to reversal if it is erroneous. For the purpose of determining admissibility, the Board assumed that NIPSCO's short pilings proposal might not meet applicable health and safety or environmental standards. However, taking into account the present posture of that design proposal, the uncertainties surrounding the unknown quantity of the sub-soil composition, the testing program already instituted, the close scrutiny given to the proposal by the Staff and Commission, and the Commission's determination in CLI-79-11, 10 NRC 733 (1979) that the short-pilings proposal could have abided the operating license stage in its posture at the construction permit proceeding, we determined that the issue could abide the operating license stage. Nothing submitted by intervenors has persuaded us that we erred in exercising our judgment to deny that issue.

ORDER

For all of the foregoing reasons and based upon a consideration of the entire record in this matter, it is, this 20th day of February, 1981

ORDERED

That PCCI' and State of Illinois' objections to, and motions for reconsideration of, the Board's Order of December 24, 1980 denying the newly-filed contentions and short-pilings issue *are denied.*

In view of the failure of intervenors' motions to demonstrate that a certification or referral to the Appeal Board or Commission is necessary to prevent detriment to the public interest or unusual delay or expense, intervenors' alternative motions for certification or referral *are also denied.*

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Herbert Grossman, Chairman
ADMINISTRATIVE JUDGE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Charles Bechhoefer, Chairman
Dr. George C. Anderson
Ralph S. Decker

In the Matter of

Docket No. 50-409-SC
Prov. Op. Lic. DPR-45
(Show-Cause Proceeding)

DAIRYLAND POWER COOPERATIVE
(La Crosse Boiling Water Reactor)

February 24, 1981

Upon conclusion of an evidentiary hearing on the risk of allowing the La Crosse facility to remain in operation without a site dewatering system pending the Licensing Board's final determination of the size of the safe shutdown earthquake (SSE) for the facility, the Licensing Board in this show-cause proceeding determined: (1) no warrant exists for installation of a dewatering system to preclude liquefaction in the event of an earthquake producing peak ground acceleration at the site of 0.12g (the Licensee's and Staff's assumed SSE); and (2) there is no undue risk to the public health and safety in permitting operation *pendente lite* without installation of a site dewatering system.

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING (BURDEN OF PROOF)

In any show-cause proceeding arising after the grant of a construction permit but prior to the award of a full-term operating license, the licensee must bear the "burden of proving compliance with Commission safety regulations." *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-315, 3 NRC 101 (1976).

TECHNICAL ISSUE DISCUSSED:

Earthquake induced liquefaction, seismic hazard analysis (recurrence frequency).

APPEARANCES

Messrs. O. S. Hiestand and **Kevin P. Gallen,** Washington, D.C., for Dairyland Power Cooperative, Licensee.

Ms. Anne K. Morse, La Crosse, Wisconsin, for the Coulee Region Energy Coalition, Intervenor.

Mr. Frederick M. Olsen, III, La Crosse, Wisconsin, Intervenor, *pro se.*

Mr. Stephen G. Burns and **Ms. Karen D. Cyr,** for the Nuclear Regulatory Commission Staff.

INDEX

PARTIAL INITIAL DECISION

(Permitting Continued Reactor Operation Without Site Dewatering System)

I. PROCEDURAL BACKGROUND

On February 25, 1980, the Director, Office of Nuclear Reactor Regulation, of the Nuclear Regulatory Commission issued an Order to Show Cause under which Dairyland Power Cooperative (DPC or Licensee), the holder of Provisional Operating License No. DPR-45 for the La Crosse Boiling Water Reactor (LACBWR), was required to show cause why it should not

> "1. As soon as possible, but no later than May 27, 1980, submit a detailed design proposal for a site dewatering system to preclude the occurrence of liquefaction in the event of an earthquake with peak ground surface accelerations of 0.12g or less;" and
> 2. As soon as possible after NRC approval of the aforesaid site dewatering system, "but no later than February 25, 1981, make such system operational, or place the LACBWR in a safe cold shutdown condition."

The order provided an opportunity for a hearing for the Licensee or "any other person whose interest may be affected" by the order.[1]

In response, the Licensee submitted a study which, it claimed, showed cause why it should not be required to design and make operational the site dewatering system. DPC sought a hearing if the NRC Staff should disagree with the conclusions reached in that study. Hearing requests were also filed by Ms. Anne K. Morse,[2] on behalf of the Coulee Region Energy Coalition (CREC), and by Mr. Frederick M. Olsen, III, *pro se.*

By Order dated July 29, 1980, the Commission delegated to this Licensing Board the authority to rule on the requests for a hearing and, if we determined a hearing were required, to conduct an adjudicatory hearing "solely on contentions within the scope of the issues identified in the

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[1]The order was published at 45 Fed. Reg. 13850 (March 3, 1980).

[2]The show-cause order had been issued in partial response to a request for such an order filed on May 21, 1979, by Ms. Morse. The request had sought to suspend the LACWBR operating license for several discrete reasons. Except with respect to the liquefaction question which became the subject of the show-cause order, the request for license suspension was denied; with respect to liquefaction, the show-cause order permitted operation to continue for one year. See "Director's Decision Under 10 CFR 2.206," DD-80-9, 11 NRC 392 (February 29, 1980).

February 25, 1980, Order: (1) whether the licensee should submit a detailed design proposal for a site dewatering system; and (2) whether the licensee should make operational such a dewatering system as soon as possible after NRC approval of the system, but no later than February 25, 1981, or place the LACBWR in a safe cold shutdown condition."[3]

Following this delegation, we invited the Licensee and NRC Staff to file responses to the petitions of CREC and Mr. Olsen. Memorandum and Order dated August 5, 1980.[4] We also scheduled a prehearing conference to consider the various hearing requests.[5] Thereafter, in a Safety Evaluation Report (SER) dated August 29, 1980, the Staff changed its earlier position and determined that a site dewatering system need not be installed, based upon further studies by the Licensee and review by the Staff and its consultant (WES), discussed *infra.*

At the prehearing conference we announced that we were granting the hearing requests of CREC and Mr. Olsen[6] and were consolidating the two intervenors for purposes of participation in this proceeding. We established a schedule for discovery and for the filing of motions for summary disposition. We also announced our conclusion that the size of the safe shutdown earthquake (SSE) was an issue we thought should be explored in this show-cause proceeding.[7] Because the Licensee (and the Staff during the prehearing conference) disagreed with this conclusion, we agreed to certify the question of our authority to consider this issue to the Appeal Board. These rulings are memorialized in our Prehearing Conference Order Granting Requests for a Hearing and Certifying Question to Appeal Board, LBP-80-26, 12 NRC 367 (September 30, 1980).[8] On September 30, 1980, we

[3]This Order of the Commission was published at 45 Fed. Reg. 52290 (August 6, 1980).

[4]We noted that no response to the Licensee's conditional hearing request was warranted inasmuch as, if the Staff were to continue to believe that a site dewatering system should be installed, the Licensee would have a right to a hearing under 10 CFR §2.202(c).

[5]The conference was scheduled for September 11, 1980, in La Crosse, Wisconsin. See Prehearing Conference Order dated August 22, 1980, published at 45 Fed. Reg. 57613 (August 28, 1980).

[6]Previously, Mr. Olsen had moved to disqualify the entire membership of this Licensing Board. We denied this motion in our Memorandum and Order dated September 19, 1980. In accordance with 10 CFR §2.704(c), we referred this denial to the Appeal Board. On September 24, 1980, the Appeal Board summarily affirmed our ruling. ALAB-614, 12 NRC 347.

[7]In its full-term operating license application, dated October 9, 1974, DPC proposed 0.12g as the peak ground acceleration resulting from the SSE. Show-cause order, p. 2, 45 Fed. Reg. 13850. The Staff has not yet completed its evaluation of this proposal. In LBP-80-26, we described why we believed that we should determine the acceptability of this proposal rather than assuming its validity for the purposes of this proceeding.

[8]Before the Appeal Board, the Staff changed its earlier tentative position and agreed we were empowered to consider, and should consider, the magnitude of the SSE. The Appeal Board affirmed our earlier ruling. ALAB-618, 12 NRC 551 (November 17, 1980). On December 1,

issued a Notice of Hearing.[9]

In our Memorandum and Order Scheduling Evidentiary Hearing and Prehearing Conference, dated November 12, 1980, we stated that, were we to find that the installation of a dewatering system as proposed by the show-cause order was required, the Licensee could not design such a system, have it approved by the NRC Staff, and install it by February 25, 1981, the cut-off date specified in the show-cause order. We also expressed some reservation whether we would be able to render an initial decision prior to February 25, 1981.[10] We further observed that the one year period of operation which the show-cause order permitted had been based on a hazard analysis performed by the Staff and that any further extension of time (absent a final decision in this proceeding) would require an additional hazard analysis. We therefore scheduled an evidentiary hearing on the risks of continued operation *pendente lite* (including 5 particular areas of inquiry which we identified.)[11]

On November 14, 1980, the Licensee and NRC Staff each filed motions for summary disposition, covering all issues except the Board-raised question of the size of the SSE. The Intervenors failed to respond to either motion. (The Staff on December 9, 1980 supported the Licensee's motion except with respect to one limited question relating to the soils beneath the stack of the adjoining Genoa-3 coal fired plant. The Staff had not reviewed these studies but deemed them extraneous to our eventual determination.)

On December 5, 1980, we issued a Memorandum which posed the substantive questions to which we sought answers prior to any final determination in this proceeding (see fn. 10, *supra.*) We noted that at the forthcoming prehearing conference we would discuss the manner in which these questions would be addressed; as it turned out, the Licensee and NRC Staff were able to respond to the questions adequately at the evidentiary hearing.

On December 16 and 17, 1980, we conducted an evidentiary hearing on the risk of extending the February 25, 1981 cut-off date. Testimony was presented by the Licensee and the NRC Staff. The witnesses for those

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1980, the Licensee sought Commission review of these rulings. The Commission has not yet acted on that request.

[9]This Notice was published at 45 Fed. Reg. 66537 (October 7, 1980).

[10]Among other matters, we noted that we had some substantive questions which we intended to pose to the parties in the near future and that we expected to have answers to those questions prior to any final determination in this proceeding.

[11]We also scheduled a prehearing conference to consider any summary disposition motions which might be filed, the manner in which the parties proposed to respond to Board questions (see fn. 10, *supra*), and further scheduling. A Notice of Prehearing Conference and Evidentiary Hearing was published at 45 Fed. Reg. 76557 (November 19, 1980). The schedule for the prehearing conference and evidentiary hearing was slightly modified by our Order of November 25, 1980, published at 45 Fed. Reg. 79954 (December 2, 1980).

parties also offered to address the substantive questions raised by the Board in the Memorandum of December 5, 1980, as well as certain questions arising from the affidavits provided in support of the Licensee's and Staff's motions for summary disposition. As a result of the Intervenors' failure to respond to those motions, and because the questions also were pertinent to the direct case presented by the Licensee and Staff concerning the risk (or lack of risk) of extending the cut-off date, we found no due-process objection to the witnesses' addressing those matters without having submitted prepared testimony on the particular questions. During the course of the hearing, the witnesses responded to our questions. Because the Staff had earlier indicated that it would not be prepared to address the SSE question for at least 6 months, we limited the inquiry at the December hearing to the risk of liquefaction arising from the Licensee's and Staff's assumed SSE, producing peak ground acceleration of 0.12g at the site. Determination of the SSE, and additional liquefaction potential which might attend an SSE producing greater than 0.12g ground acceleration at the site, was accordingly deferred to a later date.

At the prehearing conference following the evidentiary hearing on December 17, 1980, we established a schedule for the submission of proposed findings of fact and conclusions of law (as well as for discovery on the remaining SSE question). See Prehearing Conference Memorandum dated January 6, 1981. Proposed findings and conclusions were submitted by the Licensee and Staff. The Intervenors did not do so. (The Licensee also responded to the Staff's proposed findings and conclusions, offering no objection to them.)

In preparing the following findings of fact and conclusions of law, we have reviewed and considered the entire record of this proceeding, including the affidavits submitted in support of the motions for summary disposition and the findings of fact and conclusions of law proposed by the parties. Those proposed findings not incorporated directly or inferentially in this Partial Initial Decision are rejected as being unsupported by the record or as being unnecessary to the rendering of this Decision. We have also considered the limited appearance statements from members of the public received during the evidentiary hearing sessions.

For the reasons which follow, we find no warrant for the installation of a dewatering system to preclude liquefaction in the event of an earthquake producing peak ground acceleration at the site of 0.12g. We also reiterate our conclusion of the necessity of definitively ascertaining the size of the SSE for this facility prior to any final determination of the need for a site dewatering system. Finally, we conclude that there is no undue risk to the public health and safety in permitting operation *pendente lite,* without installation of a site dewatering system.

II. BURDEN OF PROOF

At the outset of the evidentiary hearing (Tr. 73-77), the Licensee took the position that the burden of proof (as well as of going forward with evidence) should be placed on the Intervenors. DPC reasoned that, because of the Staff's change in position, the Intervenors were the only remaining parties seeking imposition of the remedies sought by the show-cause order; and, for that reason, became the "proponents" of the order within the meaning of 10 CFR §2.732 and Section 7 of the Administrative Procedure Act, 5 U.S.C. §556(d). We ruled that DPC had the burden of proof (Tr. 77).

This ruling was premised upon our understanding of the decisions of the Appeal Board in *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-283, 2 NRC 11 (1975), *clarified on reconsideration*, ALAB-315, 3 NRC 101 (1976). There, in a show-cause proceeding arising after the grant of a construction permit but prior to the award of an operating license, the Appeal Board held that the Atomic Energy Act modified the normal rule of the Administrative Procedure Act by placing the burden of proof on safety matters on the construction permit holder (the licensee). We interpret these decisions as according the burden of proof to the licensee, at least until award of a full-term operating license. The intercession of a provisional operating license does not alter this statutory scheme. As the Appeal Board pointed out, a utility must bear the "burden of proving compliance with Commission safety regulations" at the beginning and end of the licensing process, as well as in the interim (ALAB-315, 3 NRC at 104); since a provisional operating license has only limited life and a holder of such a license still must prove its entitlement to a full-term license, the licensing process has not yet ended for the holder of a provisional license. As a result:

> * * * the Atomic Energy Act intends the party seeking to * * * operate a nuclear reactor to bear the burden of proof in any Commission proceeding bearing on its application to do so, including a "show cause" proceeding.

Id. at 105.[12]

The subject matter of this show-cause proceeding clearly bears upon DPC's application for a full-term operating license. This issue could readily be litigated in that context, and DPC would have the burden of proof. 10 CFR §2.732. The circumstance that the liquefaction issue was brought to

[12]The Appeal Board expressly reserved judgment with respect to the burden of proof in a show-cause proceeding involving potential withdrawal of an operating license. ALAB-315, 3 NRC at 109, n. 20. Our decision here, of course, involves only a provisional and not a full-term operating license.

light prior to the completion of the Staff's review in the operating license proceeding should not, and in our opinion does not, shift the burden of proof from the Licensee. *Cf.* ALAB-283, *supra,* 2 NRC at 17.

DPC asserted two reasons why, in its opinion, the *Midland* decisions do not govern the instant show-cause proceeding. First, it claimed that the decisions do not address the situation, as here, where the Staff is no longer advocating the original position taken in the show-cause order (Tr. 74). *Midland,* however, involved an analogous situation: during the hearing, the show-cause order did not enjoy the support of any active party, since the intervenors were no longer participating and the Staff no longer favored the order. ALAB-283, *supra,* 2 NRC at 15.

Second, DPC argued that the *Midland* burden-of-proof interpretation is about to be modified by the Commission. It referred to the NRC's publication of a proposed rule which would lead to that result. 42 Fed. Reg. 37406 (July 21, 1977). We understand, however, that the Commission is about to withdraw the proposed rule and accordingly allow the *Midland* ruling to remain in effect. The Staff confirmed this understanding (Tr. 75-76). In any event, we are bound by *Midland* (to the extent applicable here) unless and until it is modified or abrogated by the Commission.

In sum, we hold that the burden of proof in this proceeding falls on DPC.[13] And although under *Midland* the Intervenors may have the burden of initially demonstrating evidence sufficient to cause us to inquire further (ALAB-315, 3 NRC at 112), their reliance on a 1978 study of the U.S. Army Engineer Waterways Experiment Station (see pp. 267, 270 *infra*) amply satisfies this requirement.

III. FINDINGS OF FACT

Introduction

1. The Licensee's direct testimony was sponsored by a witness panel consisting of Dr. Robin K. McGuire, a Senior Engineer with Dames & Moore specializing in risk analysis, earthquake engineering, and decision analysis; Dr. Mysore S. Nataraja, a Senior Engineer at Dames & Moore specializing in soil mechanics, earthquake engineering, and geotechnical instrumentation; and Mr. John D. Parkyn, the Assistant Superintendent of LACBWR. This testimony ("DPC panel testimony") was admitted into evidence at Tr. 306.[14] Messrs. Nataraja and Parkyn also sponsored

[13]The circumstance whereby we permitted the Staff to present a portion of its testimony prior to the Licensee's testimony resulted from schedule conflicts of one Staff witness and does not in any way stem from the allocation of the burden of proof or burden of going forward.

[14]All transcript references are to the transcript of the evidentiary hearings held on December 16 and 17, 1980.

affidavits in support of the Licensee's motion for summary disposition. (References to those affidavits will appear as "Nataraja aff." or "Parkyn aff.".) The Licensee also presented testimony by Mr. Richard Shimshak, LACBWR Superintendent (Tr. 384-85). The Staff presented testimony by Dr. Leon Reiter, Leader of the Seismology Section in the Geosciences Branch, Division of Engineering, Office of Nuclear Reactor Regulation, NRC (Reiter testimony, fol. Tr. 85) and by Mr. Howard A. Levin, Technical Assistant to the Director, Division of Engineering (Levin testimony, fol. Tr. 90). The Staff's motion for summary disposition was supported by the affidavit of Mr. John T. Greeves, a Geotechnical Engineer in the Hydrologic and Geotechnical Engineering Branch of the Division of Engineering ("Greeves aff."). Mr. Greeves appeared as a witness but did not sponsor direct testimony (except to the extent that he had participated in preparation of the Staff's SER).[15] His professional qualifications were admitted into evidence and appear fol. Tr. 92. A list of exhibits appears in the Appendix to this decision (*infra*, p. 281).

Background

2. The La Crosse Boiling Water Reactor (LACBWR) is located on the east bank of the Mississippi River near Genoa, Wisconsin. SER, p. 1. It is currently permitted to operate under Provisional Operating License DPR-45. DPC has applied for a full-term operating license; that application is currently under review by the Commission. The review of safety issues for LACBWR has been performed in the context of the Commission's Systematic Evaluation Program (SEP), a program under which NRC is reevaluating the safety margins of a number of older reactors, including LACBWR. During the course of that review, the question of liquefaction potential at the La Crosse site surfaced, leading to issuance of the February 25, 1980 show-cause order.

3. Soil at the site consists primarily of sand deposits and emplaced hydraulic fill material extending to a depth of approximately 130 feet, down to bedrock. SER, p. 1; DPC panel testimony, p. 3. The reactor containment building, the turbine building and the stack are all pile supported structures. The crib house and associated underground piping which are also important to safety are not pile supported. DPC panel testimony, p. 6; Parkyn aff., para. 2.

4. Loose to medium sands of the type found at LACBWR tend to compact when subjected to cyclic loadings such as those induced by an

[15] "SER" refers to the "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Liquefaction Potential at the La Crosse Site," dated August 29, 1980, identified as Staff Exhibit 5 and admitted into evidence at Tr. 96.

earthquake. If the sand is saturated, sufficiently strong vibratory motion can cause reduction in its shear strength so that the sand is unable to resist the imposed shear stresses. Under these circumstances, the sand can behave much like a liquid, *i.e.*, liquefaction can occur. SER, p. 2; DPC panel testimony, p. 3. Generally, the stronger the earth shaking, the longer it lasts; and the wetter and less dense the sand, the more likely it is that liquefaction will occur.

Events Leading to the Show-Cause Order

5. To assist in evaluating the liquefaction potential at LACBWR, Dairyland engaged the services of Dames & Moore, an engineering consulting firm which took soil samples on several occasions, performed liquefaction analyses, reported its findings, and advised the Licensee. Similarly, the NRC Staff called upon the Geotechnical Laboratory, U.S. Army Engineer Waterways Experiment Station (WES), to analyze available information, prepare a report, *Liquefaction Analysis for LaCross [sic] Nuclear Power Station*, December, 1978 (hereinafter WES Rpt.), and provide advice to the Staff. As shall be seen, the analysis and conclusions of the WES Rpt. provided a foundation for the Staff's issuance of the 1980 show-cause order.

6. Two methods, commonly referred to as the analysis/testing method[16] and the empirical method, were used by both Dames & Moore and WES to evaluate the liquefaction potential at LACBWR. DPC panel testimony, pp. 3-4; SER, p. 2. The analysis/testing method consists of taking soil samples at the site in question, modeling the site soil condition in the laboratory, conducting a series of cyclic, tri-axial tests on the samples, establishing the cyclic shear strength for the soil over a range of confining pressures, and comparing shear strength with dynamic shear stresses. DPC panel testimony, p. 4; SER, p. 2. The empirical method utilizes Standard Penetration Tests (SPT), in which a rod of standard shape is driven into the soil by standard impacts and the number of blows required to drive the rod through a unit depth is counted. These data are then compared with the blow counts measured at many other sites where soil performance under actual earthquake conditions has been recorded. *Id.* The more blows required to drive the rod a unit depth, the denser is the soil and the more the soil is able to resist liquefaction. Both methods require a specification of expected vibratory ground motion and a knowledge of soil properties.

7. In describing earthquake induced vibratory ground motion at the LACBWR site, the Staff, the Licensee, and their consultants have all used

[16]This method is also referred to as the laboratory/analytical approach.

peak ground surface acceleration as the principal parameter. Other descriptive parameters such as magnitude, intensity and duration of ground shaking can be related to peak acceleration. For example, the show-cause order states (at pp. 4, 6, 45 Fed. Reg. at 13850, 13851):

> * * * that if sustained strong ground motion with peak accelerations of .12g or higher occurs (normally associated with a magnitude 5 or greater earthquake within 10 km of the site)* * * and
>
> This peak acceleration (.12g) is equivalent to Intensity VII when using the relationship of Trifunac and Brady (1975).

Moreover, the duration of ground shaking, which is often expressed in terms of uniform cycles for soil testing and analysis purposes, is also related to peak acceleration, earthquake magnitude and intensity. In this case, the Staff states in its SER (at pp. 2-3):[17]

> There has been general agreement between the staff and the licensee that the earthquake loading at the La Crosse site can be conservatively characterized as a magnitude 5 to 5 1/2 event at a distance of less than 25 KM with a peak ground acceleration of 0.12g* and an equivalent duration of 5 cycles.
>
> *g is acceleration of gravity, i.e. 32.2 ft/sec/sec"

8. In this phase of the show-cause proceeding, peak ground surface acceleration is assumed to be 0.12g as specified in the show-cause order.[18] As indicated earlier, however, the Board believes that a substantial question exists concerning the adequacy of this assumption and that this question must be resolved prior to a final determination in this proceeding of whether a dewatering system need be installed to preclude liquefaction at the LACBWR site. See LBP-80-26, *supra,* 12 NRC at 376-379; also p. 261 of this Decision.

9. The initial soils investigation at the La Crosse site was conducted in 1962 by Raymond International. Additional borings were made in 1973 under Dames & Moore supervision. SER para. 2.0 and Fig. 1. These data

[17]See also WES Rpt., Figures 9 and 10; Tr. 184-185.

[18]By way of background, it is interesting to note that WES used a more severe specification of ground motion in its liquefaction analysis appearing in its 1978 report. For that study, Staff specified peak accelerations of 0.12g and 0.20g. However, WES assumed an Intensity IX or magnitude 6.6 earthquake with a duration of 10 cycles. WES Rpt., para. 17, pp. 8-9. In assessing the WES analysis, the Staff concluded, among other things, that the assessing the WES analysis, the Staff concluded, among other things, that the assumptions regarding seismicity at the site were conservative. Show-cause order, p. 2 (45 Fed. Reg. at 13850).

were included in DPC's 1974 full-term operating license application and were utilized by WES in its liquefaction analysis.[19] The NRC Staff's assessment of the WES analysis indicated that the then-available soils data were inadequate to estimate accurately the liquefaction resistance. See show-cause order, p. 3 (45 Fed. Reg. at 13850). Consequently the Licensee undertook an additional soils investigation program which was reviewed in advance and approved by the Staff in April, 1979. Based on analyses of the data obtained in the five additional borings made in the LACBWR yard area in May, 1979, Dames & Moore concluded that, for the postulated earthquake with a 0.12g peak ground acceleration, the laboratory/analytical approach yielded a minimum safety factor[20] of 1.5 and the empirical approach a minimum value of 1.0. However, after review by WES and itself, the Staff concluded that the laboratory shear strength curves used in the Dames & Moore analysis were not adequately conservative and that foundation soils down to a depth of approximately 40 feet were not safe against liquefaction if subjected to sustained ground motion with a peak acceleration of 0.12g or higher (which the Staff associated with a magnitude 5 or greater earthquake within 10 km. of the site). Because of this Staff conclusion, the Licensee proposed a plan for dewatering the site to lower the groundwater level. However, Dairyland subsequently withdrew the proposal. As a result, the show-cause order was issued by NRC on February 25, 1980. SER, para. 3.0; show-cause order, p. 4 (45 Fed. Reg. at 13850).

Events Following the Show-Cause Order

10. In response to the Order to Show Cause, Dairyland cited the improved density which could be expected due to vibration during the driving of piles, surveyed pertinent case histories, and subsequently made

[19]To understand the evolution of knowledge of soil properties at LACBWR and the results of the several liquefaction analyses performed, it is again instructive to recognize the limited information available to WES in the conduct of its 1978 evaluation. WES reviewed the data from the 1962 and 1973 borings but was unsure whether penetration tests were conducted by standardized methods. WES Rpt., para. 7. WES also noted that some soil samples were taken using one technique and others by a different one. WES Rpt. para. 8. WES was aware that the tubine and reactor containment buildings and the stack were pile supported. However, it was uncertain as to whether some of the borings were made prior to or following pile driving. WES Rpt. para. 8. WES estimated the increase in soil density under pile supported buildings based solely on the reduction in void ratio which would occur assuming the soil displaced by the pile went entirely into taking up the voids of the adjacent soil. The increase in soil density so estimated was minor and WES did not consider it to be significant. WES recognized that additional density could have occurred due to vibration during pile driving but data were not then available to support this hypothesis. WES Rpt., para. 11.

[20]WES states that "[t]he factor of safety against 10 percent double-amplitude strain has been defined as the dynamic shear strength divided by the average dynamic shear stress." WES Rpt. para. 19. The higher the safety factor, the more resistance there is to liquefaction.

four additional test borings (which were monitored by Mr. Greeves of the Staff, Tr. 110). These borings were made through the turbine building floor and stack foundation into soils directly below these structures, in contrast with previous borings made in the yard area or free field. These borings clearly indicated significantly greater soil densities under the pile-supported structures than in the free field. Analysis and evaluation of these results were reported by Dames & Moore in July, 1980. Upon review of the logs of these new borings and the data reported by Dames & Moore and the Licensee, the Staff and WES concluded that the soils below the reactor containment building, the turbine building and the stack are adequately safe against liquefaction effects for an earthquake up to a magnitude 5.5[21] with a peak acceleration of 0.12g.[22] The Staff also concluded that mitigative measures (e.g., the dewatering system) to increase the margin of safety against liquefaction for these structures are not needed. However, liquefaction remains a concern for the crib house and associated underground piping, which provide one of the sources of emergency core cooling water to the reactor containment building and which are not pile supported. SER, para. 3.0; DPC panel testimony, p. 6. Therefore, the Licensee is installing a redundant dedicated safe shutdown system to be housed inside the turbine building to provide additional emergency cooling water and preclude reliance on the crib house and buried piping in the event of an earthquake. Licensee is required by Staff to have this system operational by February 25, 1981. DPC panel testimony, p. 6; SER, para. 3.0; Tr. 312, 356, 377.

11. As we have described earlier (*supra*, pp. 261-263), both the Licensee and the Staff, having reached agreement that a dewatering system is unnecessary, entered motions for summary disposition, to which the Intervenors did not respond. As part of its motion, through the Greeves affidavit, the Staff responded to certain concerns expressed by the Intervenors during discovery, as well as at the September, 1980 prehearing conference.

12. For example, the Intervenors disagreed with the Staff's position as expressed in the SER that borings under the turbine building and stack foundation are representative of adjacent pile supported structures. In his affidavit, Mr. Greeves responded that borings through the turbine building floor were carefully selected to represent the lower end of the range of soil

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[21] In response to questioning by the Intervenor, Staff witness Greeves stated that liquefaction has not been experienced and recorded for earthquakes of magnitudes less than 5.5. Tr. 113. Dr. McGuire, in response to a Board inquiry, stated that the magnitude of the smallest earthquake of which he was aware which did cause liquefaction was approximately 5.3. Tr. 374.

[22] Corps of Engineers, Letter from F. R. Brown to J. P. Knight (NRC), July 25, 1980, Ref. 5 to SER.

density for pile supported structures and that turbine building piles are more widely spaced relative to the piles under the stack and reactor building. Moreover, soil density generally increases with distance from the river bank and borings under the turbine building were closer to the river than those under the stack foundation. The reactor building is located still further from the river. Mr. Greeves therefore concluded that borings under the turbine building are in fact a conservative representation of soil conditions under other pile-supported foundations on the LACBWR site and that borings under the stack foundation are a conservative representation of conditions under the reactor. Greeves affidavit, pp. 3-5.

13. The Intervenors also noted that piles are not supported by bedrock. Mr. Greeves responded that the Staff does not rely on the support of the piles themselves to resist liquefaction but on the increased soil density resulting from the driving and presence of the piles. Greeves affidavit, pp. 2-3; Tr. 286.

14. The Intervenors observed that shallow voids were found under the turbine building during borings and raised the concern that voids might also exist under the reactor building. Mr. Greeves responded that the soil under the turbine building and stack are hydraulic fill and that voids were not found under the stack at the same elevation.[23] Moreover, pile spacing is closer under the reactor building than under the turbine building. Greeves affidavit, pp. 4-5; Tr. 377-382. Under direct examination, Mr. Richard Shimshak, LACBWR Superintendent, also testified that voids might well be expected under the turbine building due to turbine vibrations, a condition not present under the reactor building or under the stack. Tr. 384-385. In any case, Dames & Moore witness Nataraja testified that the presence or absence of voids has no bearing on his conclusion regarding liquefaction. Tr. 382. The voids under the turbine building have nevertheless been filled. DPC panel testimony, p. 6; Tr. 378; Parkyn affidavit, p. 2.

15. The Intervenors noted that the driving of piles could also increase pore pressure, a condition which accompanies liquefaction. The Staff agreed but pointed out that any increased pore pressure dissipates shortly after the pile is driven and has no further significance. Greeves affidavit, p. 5.

16. Mr. Greeves also addressed the question of whether dense soils such as those beneath pile supported structures would remain stable even if adjacent soils in free field areas undergo liquefaction. He expressed his view that denser soils would remain stable and stated that this had been

[23]The reactor building does not rest on hydraulic fill. Rather its base is 29 feet below grade and rests on the denser soil existing there before the surface grade was raised with fill material. SER para. 2.0; Dames & Moore "Response to NRC Review Questions" transmitted by letter of July 14, 1980 from Frank Linder, DPC to Dennis Crutchfield, NRC, p. II-13; Tr. 365.

demonstrated in response to actual events. In particular, he cited and attached a study which reported that pile supported oil tanks at Ishinomaki, Japan, remained stable even though surrounding soils liquefied extensively as a result of the Miyagiken-oki earthquake of June 12, 1978. During questioning by the Board, Mr. Greeves testified that the situation at LACBWR and the oil tanks at Ishinomaki were comparable.[24] He also pointed out that the oil tank site was subjected to a magnitude 7.4 earthquake with ground surface accelerations around 0.18g up to 0.25g and a duration of 20 equivalent cycles. Mr. Greeves opined that such a magnitude involves effectively 100 times[25] the energy of a magnitude 5.5 earthquake. Tr. 271.

Responses to Board Questions

17. After the summary disposition motions had been filed, the Board evaluated the total record to date, including the affidavits supporting the motions. Because we believed the record to be still somewhat incomplete in a few areas, we posed several questions which were provided to the parties by our Memorandum of December 5, 1980. As described earlier (*supra*, p. 262), these questions were responded to by Licensee and Staff witnesses during the evidentiary hearings on December 16 and 17, 1980. A summary of the questions and responses follows.

18. Our first question related to the effects of Mississippi River water level on the below grade water table, and the consequent effects on soil properties (and thus on liquefaction potential). Witnesses for both the Staff and Licensee agreed that a rise in Mississippi River water level would be accompanied by a rise in the water table, that soils below the higher water table would become saturated, that resistance to liquefaction would be reduced, but that even at flood stage where the water table rose to the surface, soils under pile supported structures are sufficiently dense to resist liquefaction should an earthquake producing a peak acceleration of 0.12g occur. Tr. 248-252, 391-392. The Licensee tesified that the probability that a flood and an SSE producing 0.12g ground acceleration would occur simultaneously is very low. Tr. 389-390, 392.

19. We then asked if there would be a density gradient radially outward from each piling, where samples were taken radially from pilings, and whether the gradient was considered in the liquefaction analyses. Again,

[24]In response to questions by the Intervenors, Mr. Greeves acknowledged that the piles under the oil tanks and those under the LACBWR structures were different in nature. Tr. 274-275. But he testified that the degree of densification produced in each case is the significant factor and that the densification at Ishinomaki and LACBWR was comparable—indeed, the LACBWR soils reflected slightly higher densification. Tr. 281, 285.

[25]On the basis of its own estimates, the Board believes that this figure is low.

both Staff and Dairyland witnesses testified that some density gradient would be expected laterally from a single piling and also laterally from a group of pilings into the surrounding free field. Such density gradients would be expected to persist with time. Moreover, in order to avoid striking pilings during borings for samples under the turbine building, the boring was made as close to the midpoint between pilings as possible so that the samples obtained should be of lower density than had they been obtained immediately adjacent to a piling. Tr. 120-124, 361-364. The differences would not be significant. Tr. 364.

20. We also asked if a dewatering system designed to protect against liquefaction at 0.12g would be effective at 0.20g as well. Dames & Moore witness Nataraja testified that the preliminary dewatering system design so far considered would not preclude liquefaction at 0.20g unless the factors of safety were already sufficiently large. Larger diameter wells, longer screens or higher capacity pumps would likely be required to lower the water table still further. Dr. Nataraja also pointed to certain detrimental effects of dewatering systems and indicated that they would be greater if a higher capacity system were installed. Tr. 393-396.

21. Since the show-cause order was not explicit as to the duration of ground shaking, we asked for clarification. Dr. Nataraja testified that liquefaction potential did depend on the duration of ground shaking as well as magnitude and acceleration, that these parameters were related, that Dames & Moore had used five equivalent uniform cycles, and that that number was conservative. He referred to Figure 10 of the WES report which is a plot of equivalent number of cycles vs. earthquake magnitude and indicates the degree of conservatism associated with the selection of five cycles for purpose of liquefaction analysis. Tr. 396-399.

22. In his affidavit supporting Licensee's motion for summary disposition (para. 9, 10), Dr. Nataraja stated that the factor of safety against liquefaction is "greater than the normally accepted minimum factor of safety." However, he did not elucidate on the term "normally accepted minimum factor of safety." We asked for clarification. Dr. Nataraja testified that the factors of safety determined by Dames & Moore were higher than 1.25, that there is no minimum factor of safety for all cases but that, in his opinion, the factors of safety obtained are adequate by all standards. Tr. 371-374.[26] Staff witness Greeves agreed that a safety factor of 1.25 would be a reasonable guide if it were properly derived considering investigation and analytical techniques and experience level. Tr. 256-257.

[26]The WES Rpt. states (para. 20) that a factor of safety of 1.25 "is often considered reasonable for safety in the type of analysis performed herein."

Board Finding as to Liquefaction

23. Upon careful consideration of all the evidence summarized above, the Board finds that there is reasonable assurance that soils under pile supported structures will not liquefy if subjected to a 5.0 to 5.5 magnitude earthquake with peak ground acceleration of 0.12g or less. There is also reasonable assurance that additional emergency cooling water can be provided for safe reactor shutdown using a dedicated safe shutdown system which is required to be installed by February 25, 1981. We therefore find that the installation of a dewatering system is not required to protect against liquefaction in the event of the foregoing earthquake.

Risk to Public Health and Safety of Continued Operation of LACBWR without a Dewatering System Pending Final Determination

24. When we scheduled the December 16-17, 1980 evidentiary hearings, we were concerned that we might not be able to reach and publish a decision before the February 25, 1981 deadline specified in the show-cause order on the limited question of whether a dewatering system would be needed to protect against liquefaction resulting from a 0.12g earthquake. Moreover, should we have found a dewatering system necessary, installation could not possibly have been completed by the cut-off date.[27] Further, we estimated that approximately another year would be required to complete litigation of the broader question of what the safe shutdown earthquake (SSE) should be and whether a dewatering system would be required should that SSE result in peak ground surface accelerations greater than 0.12g at the site. Consequently, we ordered evidentiary hearings on the risk involved in continuing operation without a dewatering system and identified several areas of inquiry to be addressed with respect thereto.

25. Both the Licensee and Staff took the position that soils under pile supported structures important to safety would remain stable and not liquefy if subjected to an earthquake resulting in earth shaking with a peak ground surface acceleration of 0.12g. Therefore, both concluded that continued operation of LACBWR without a dewatering system would pose no risk to public health and safety even if the site were subjected to an earthquake producing ground acceleration of up to 0.12g.

26. For the reasons summarized in paragraph 23 of these findings, we agree with this risk analysis. Nonetheless, as we ruled in LBP-80-26 (see p. 261, *supra*), determination of an SSE and the liquefaction potential of that SSE are both necessary in order to reach an informed decision on the

[27] Dairyland estimates that it would take approximately six months to design and install a dewatering system. DPC panel testimony, footnote 3, p. 9. Staff estimates a period of two months to review a proposed dewatering system design. Levin testimony, at p. 4.

necessity for a site dewatering system. We have not yet been able to complete our consideration of this matter—discovery is still in progress, the Staff has not yet furnished its analysis of this question, and hearings have not yet been held. If we should later determine that the SSE is above the liquefaction threshold, operation without a dewatering system *pendente lite* would involve some seismic risk.

27. When it issued its show-cause order, the NRC Staff concluded that the return period for an earthquake producing a peak ground acceleration of 0.12g would be at least 1,000 years—*i.e.,* the probability of occurrence of such an earthquake would be 10^{-3} or less. (The Staff noted that, while this value should not be interpreted as an absolute minimum, the actual return period could be an order of magnitude larger.) As a result, the Staff concluded "that the general level of seismic hazard at the LACBWR site is sufficiently low that operation of the plant for the next twelve months would not endanger the health and safety of the public." Show-cause order, pp. 6-7; 45 Fed. Reg. at 13851. The Commission itself declined to review this decision of the Staff (as well as other portions of the Director's Decision (DD-80-9) which declined to adopt further relief, including immediate license suspension, requested by Ms. Morse). See Memorandum dated April 17, 1980 from Samuel J. Chilk, Secretary, to Leonard Bickwit, Jr., General Counsel, subject: Secy-A-80-43 (enclosure 3 to NRC Staff's Response to Requests for Hearing, dated August 29, 1980).

28. In undertaking the foregoing analysis, the Staff was assuming that liquefaction could occur in the event of an earthquake producing 0.12g ground acceleration at the site. Although we have found that assumption not to be valid, we find that, prior to our resolving the SSE question, we must evaluate the risk, if any, of continued operation *pendente lite,* and that the risk criteria utilized by the Staff in the show-cause order are appropriate guidelines for us to use in evaluating the risk of liquefaction in the event of an earthquake producing greater than 0.12g ground acceleration.

29. The seismic hazard giving rise to such risk is an earthquake producing ground acceleration at or near the site of greater than 0.12g. In response to our specific questions, Licensee and Staff both presented estimates, discussed more fully below, of the "seismic hazard," *i.e.,* the annual probability of occurrence (or its reciprocal, the return period) of a 0.12g or greater earthquake. We note here that, as Staff testified, "seismic hazard" is only one element of "seismic risk." To estimate seismic risk involves combining the probability of occurrence of the causal event with the probability that harmful effects to public health and safety will eventuate. Neither Staff nor Licensee had undertaken a quantitative seismic

risk analysis over a range of peak ground surface accelerations over 0.12g.[28] Factors which would have to be considered in such a seismic risk analysis include the extent to which pilings would continue to support buildings even if underlying soils liquefied, means and probabilities of successful core cooling, breach of primary coolant boundary probabilities, extent and probability of containment, type and degree of release of radioactive material, probabilities of a range of meteorological conditions prevailing at the time of release, population locations, densities, and probability of successful evacuation or other protective measures. Tr. 223, 406-409.

Seismic Hazard, Earthquakes of 0.12g or Higher

30. Both the NRC Staff and the Licensee presented estimates of "seismic hazard"—*i.e.,* the annual probability of occurrence of an earthquake resulting in ground acceleration at LACBWR of 0.12g or greater (or its reciprocal, the return period).

31. The Staff's estimates stemmed from a review of results thus far available from the Site Specific Spectra Program (SSSP), an effort conducted for the Staff by the Lawrence Livermore Laboratory and the TERA Corporation. In the SSSP, return periods were calculated for peak ground accelerations, peak velocities and response spectra at eastern and central U.S. sites including the LACBWR site. Since there is insufficient historical earthquake experience in the central U.S. to conduct seismic hazard analyses solely on empirical data, judgment must be used in the selection and limitations of certain parameters and empirically derived relationships. In the LLL-TERA studies, experts on eastern seismicity were polled with respect to seismic zonation, frequency of earthquake occurrences, upper magnitude cutoff, and characterization and attenuation of ground motion. Specific earthquakes such as those at Anna, Ohio, and New Madrid, Missouri, were taken into account.[29] The attenuation model used in the return period estimation for La Crosse is that proposed by Gupta and Nuttli (1976). This is one of the more conservative models for the U.S. east

[28]Staff witness Levin testified that quantification of these factors for La Crosse is not planned and is beyond the state-of-design at this time. However, the NRC Office of Research is currently conducting a probabilistically based generic program called the Seismic Safety Margins Research Program which is attempting for the first time to quantify seismic margins rigorously. Levin testimony, pp. 3-4; Tr. 405.

[29]The July, 1980 Maysville, Kentucky earthquake occurred after polling of the experts and could not therefore have been taken into account. However, based upon an examination of individual experts' assumed zone configurations and the size and frequency of earthquake distributions, it appears that inclusion of this earthquake into considerations would have minimal, if any, effect. Reiter testimony, pp. 3-4. The magnitude of the Maysville earthquake was in the range of 5.0 to 5.5, which is the value being considered by the Staff for LACBWR. Tr. 131.

of the Rocky Mountains. Calculations were also made using several other attenuation models. For La Crosse, peak accelerations for a given return period were similar. The LLL-TERA evaluations do not yet include site specific factors for amplification or deamplification based on geological structure at La Crosse. Due to compensating factors, inclusion of this refinement would not necessarily translate into significant changes in the probability analysis. Upon review of the LLL-TERA results to date, the Staff concluded that the return period for a 0.12g or greater earthquake is of the order of 1000 to 10,000 years. More likely than not, the true value is closer to the higher end of this range. The Staff does not believe that more rigorous "confidence limits" can be specified. Reiter testimony, pp. 1-6; Tr. 124-126, 206; NUREG/CR-1582, Volume 2, Staff Exhibit 6, *passim.*

32. Dames & Moore conducted a similar seismic hazard analysis for the Licensee.[30] Historical data and expert opinion were also employed and calculations were made for various attenuation models and seismogenic zones. Earthquakes were assumed to occur randomly in time and space. Magnitude-frequency distributions were truncated at 5.0 on the lower end since soil liquefaction had never been observed at lower values. See also fn. 21, *supra.* At the upper end, calculations were made using both truncated and non-truncated magnitude-frequency distributions. Again, specific factors to account for possible ground motion amplification and/or deamplification were not included. The analysis concluded that the return period for a 0.12g or greater earthquake is approximately 10,000 years. Using a reasonable range of variation in input assumptions, the results varied from 6,000 to 15,000 years. Differences between the Dames & Moore and LLL-TERA results stem from differences in truncation points and other input parameters. However, the results of both analyses overlap, at least in part. DPC panel testimony, pp. 7-9; Tr. 125, 126, 207, 247, 339, 344, 358, 360-361, 365-370, 374, 376, 387-388.

33. Based on the foregoing, we conclude that the annual probability of an earthquake producing ground acceleration of 0.12g or higher at the LACBWR site is in the range of 10^{-3} to 10^{-4}.

34. In addition to calculations at 0.12g or greater, Dames & Moore made calculations relating return period to a range of peak ground surface accelerations. Dr. McGuire, who conducted the Dames & Moore seismic hazard analyses, testified that, as a rule of thumb, if the acceleration is doubled, the return period increases by a factor of approximately ten. Thus,

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[30]This analysis is reported in the Appendix to the Dames & Moore "Response to Review Question," transmitted to the Director of Nuclear Reactor Regulation by letter from DPC dated July 14, 1980.

if one went from an acceleration of 0.11g or higher to 0.22g or higher, the return period would increase from 10,000 years to about 100,000 years.[31] Tr. 388; see also Reiter testimony, p. 3. It follows that, as the acceleration produced by an earthquake increases, the chance that such hazard would occur decreases.

35. Taking into account our earlier finding that liquefaction under pile supported structures at LACBWR is not likely to result from earthquakes producing 0.12g ground acceleration, it also follows that the hazard of liquefaction which might arise from a larger earthquake is lower than that accepted by the Staff (and the Commission) in the show-cause order as satisfactory for continued operation for a limited time period. See p. 275, *supra.*

Reconciliation of Earthquake Hazard and Standard Review Plan Criteria

36. In response to a specific Board question regarding an apparent departure from a 10^{-6} or 10^{-7} criterion accepted by the Staff with respect to other external events, Staff witness Levin testified that the 10^{-6} or 10^{-7} probabilities specified in the Standard Review Plan[32] are used as guidelines for identifying design basis events such as earthquakes. Events with lower probabilities need not be considered in specifying the design bases. Events with higher probabilities, such as possible seismic events at LACBWR, require further evaluation and actions such as preventative and/or mitigative design features. As a rule, these are evaluated deterministically rather than probabilistically. The subject of these proceedings is a case in point. Having found the probability of the causal earthquake to be higher than the guideline, deterministic procedures were employed to establish whether liquefaction would occur in the event of the assumed earthquake. Levin testimony, pp. 3-4; Tr. 410-419.[33]

[31]Dames & Moore estimated that the threshold liquefaction resistance level for the LACBWR site corresponds to an earthquake producing an acceleration between 0.18g and 0.20g at the ground surface. Show-cause order, p. 4, 45 Fed. Reg. at 13850. This estimate may be low since it was made prior to borings taken in 1980 beneath pile supported structures which showed significantly greater soil densification than in the free field. Applying the above rule of thumb, return time for a 0.18g or higher earthquake would be about 3-6 times longer than for acceleration of 0.12g or greater (Board estimate).

[32]See sections 2.2.3 and 3.5.1.6 of the Standard Review Plan; see also SECY-80-409, Table H-2, September 4, 1980.

[33]As a basis for assuming a size for the SSE for the purpose of its liquefaction analysis, however, the Staff employed a probabilistic rather than a deterministic approach. The possibly larger earthquake which might result from a deterministic approach was one reason for our raising the SSE issue.

Finding as to Risk of Continued Operation

37. Based on the foregoing, we find that the seismic hazard of an earthquake at LACBWR which might produce liquefaction is within the scope of the guidelines adopted by the show-cause order for continued operation, and continued operation *pendente lite* accordingly presents no greater risk, and possibly less risk, than the temporary operation permitted by the show-cause order.

IV. CONCLUSIONS OF LAW

Based upon the Board's evaluation of the NRC Staff's Safety Evaluation Report, the affidavits submitted by the NRC Staff and the Licensee in support of their respective motions for summary disposition and responses thereto, and the testimony of the Staff's and the Licensee's witnesses at the evidentiary hearing held at La Crosse, Wisconsin on December 16 and 17, 1980, we conclude that:

1. Reasonable assurance exists that for an earthquake of up to magnitude 5.5 with peak ground acceleration of 0.12g or less, the soils under pile-supported structures at the LACBWR site are safe against liquefaction.

2. Reasonable assurance exists that in the event that an earthquake of magnitude 5.0 - 5.5 with peak ground acceleration of 0.12g causes damage to the crib house and underground piping, emergency cooling water can be provided for safe reactor shutdown using a dedicated safe shutdown system.

3. Reasonable assurance exists that continued operation of the La Crosse Boiling Water Reactor without a dewatering system for the site will not endanger the health and safety of the public, pending a final determination by the Board on the merits of all remaining matters in controversy in this show-cause proceeding.

V. ORDER

The Board having considered and decided all matters in controversy among the parties concerning the liquefaction potential of the LACBWR site in the event of a magnitude 5.0 - 5.5 earthquake producing a peak ground acceleration at such site of 0.12g, and having satisfied its own concerns with respect to these matters, it is, this 24th day of February, 1981,

ORDERED

in accordance with the Atomic Energy Act of 1954, as amended, and the regulations of the Nuclear Regulatory Commission, that the February 25, 1981 operational date for a dewatering system at the LACBWR set forth in the February 25, 1980 Order to Show Cause is hereby extended pending a final determination by the Board on the merits of all remaining matters in controversy in this show-cause proceeding.

IT IS FURTHER ORDERED in accordance with 10 CFR §§2.760, 2.762, 2.764,[34] 2.785 and 2.786 that this Partial Initial Decision shall become effective immediately and shall constitute, with respect to the matters covered herein, the final action of the Commission thirty (30) days after the date of issuance hereof, subject to any review pursuant to the Commission's Rules of Practice.

Exceptions to this Partial Initial Decision may be filed by any party within ten (10) days after service of this Partial Initial Decision. A brief in support of the exceptions shall be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff). Within thirty (30) days of the filing and service of the brief of the appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

THE ATOMIC SAFETY AND
LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Dr. George C. Anderson
ADMINISTRATIVE JUDGE

Ralph S. Decker
ADMINISTRATIVE JUDGE

[34]The temporary suspension of 10 CFR §2.764(a) and (b) in certain proceedings, as addressed by 10 CFR Part 2, Appendix B, is not applicable to show-cause proceedings such as this.

APPENDIX

List of Exhibits

Exhibit	Description	Identified	Admitted Into Evidence
Staff		Tr.	Tr.
S-1	Prepared Testimony of Leon Reiter In Response to Board Questions, dated 12/5/80	78	85
S-2	Testimony of Howard A. Levin In Response to Board Questions, dated 12/5/80	78	90
S-3	John T. Greeves, Professional Qualifications	79	92
S-4	Memorandum from Robert E. Jackson to D. Crutchfield, dated June 23, 1980, transmitting "Initial Review and Recommendations for Site-Specific Spectra at SEP Sites."	79	88
S-5	Letter from Harold R. Denton, NRR, to Frank Linder, DPC, dated August 29, 1980, transmitting "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Liquefaction Potential at the La Crosse Site."	79-80	96
S-6	Seismic Hazard Analyses, TERA Corp. (NUREG/CR-1582, Vol. 2)	155	155
S-7	Seismic Hazard Analyses, TERA Corp. (NUREG/CR-1582, Vol. 3)	155	155
S-8	Single La Crosse Spectra	201	202
Intervenors			
I-1	Earthquake Research for the Safer Siting of Critical Facilities, National Academy of Sciences (cover page, pp. 14, 15).	135	not offered

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of **Docket Nos. 50-295**
50-304

COMMONWEALTH EDISON COMPANY
(Zion Station, Unit Nos. 1 and 2) **February 18, 1981**

The Director of the Office of Nuclear Reactor Regulation denied a request submitted by Pollution and Environmental Problems, Inc. (PEP) pursuant to 10 CFR 2.206 that the license amendment permitting reracking and compaction of the Zion Unit Nos. 1 & 2 be suspended. The Director found that the information submitted by PEP, issues raised in a spent fuel pool expansion proceeding for another facility and the possibility of future high burnup of nuclear fuel, did not warrant suspension of the amendment.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By letter dated April 17, 1980, Ms. Catherine Quigg, on behalf of Pollution and Environmental Problems, Inc. (PEP), transmitted a request pursuant to 10 CFR 2.206 for the immediate suspension of the license to permit reracking and compaction of the Zion Unit Nos. 1 and 2 spent fuel pool because of issues raised in a spent fuel pool modification hearing for the Salem Nuclear Station which petitioners believed should be considered in the Zion case. Notice of receipt of PEP's request was published in the *Federal Register* on June 23, 1980 (45 F.R. 42095).

I. Background

On February 28, 1980, the Nuclear Regulatory Commission issued Amendment Nos. 52 and 49, to Facility Operating Licenses Nos. DPR-39 and DPR-48, respectively, to revise the Technical Specifications and licenses for Zion Station, Unit Nos. 1 and 2.[1] These amendments would allow modifications of the spent fuel pool to increase the storage capacity from 868 to 2112 fuel assemblies.

The issuance of the amendment was preceded by eight days of evidentiary hearings and supplemental affidavits filed by the parties in response to a request for additional information by the Atomic Safety and Licensing Board (ASLB).[2] The Board's Initial Decision authorizing issuance of the license amendment was issued on February 14, 1980.[3] The petitioner here, Ms. Quigg, did not petition to intervene in that proceeding, but did make a limited appearance statement.

By Memorandum and Order dated February 22, 1980, the Atomic Safety and Licensing Board designated to rule on a requested license amendment to permit modifications of the Salem Nuclear Generating Station, Unit 1 spent fuel pool, directed the parties in that proceeding to answer the following questions:

a. To what extent did the accident at Three Mile Island (TMI) affect the spent fuel pool at that site?
b. In the event of a gross loss of water from the storage pool, what would be the difference in consequences between those occasioned by the pool with the expanded storage and those occasioned by the present pool?

PEP has requested the suspension of the license authorizing modification of the Zion spent fuel pool pending examination of the evidence presented in the Salem case on the Board's questions, particularly as it applies to the Zion spent fuel pool. PEP contends that the Salem Board has given credibility to a loss of cooling accident in a spent fuel pool and, therefore, these issues should be considered in the Zion case.

PEP has requested that the amendment be suspended for a second reason. PEP contends that the NRC may in the future "allow utilities to go to fuel burnups as high as 55-60,000 MWD/MTU." It believes that since "fuel burnup is one of the most important considerations in determining spent fuel thermal heat and radiation output, major factors in a loss of

[1]Amendment Nos. 52 and 49 are attached as Appendix A.
[2]The Board's request for further evidence was triggered by a Board Notification entitled "Pipe Cracks in Stagnant Borated Water Systems at PWRs" dated August 14, 1979.
[3]The Zion ASLB Initial Decision dated February 14, 1980 is attached as Appendix B. The ASLB's decision was affirmed by the Atomic Safety & Licensing Appeal Board (ASLAB). Commonwealth Edison Company (Zion Station, Units 1 & 2), ALAB-616, 12 NRC 419 (October 2, 1980), attached as Appendix C. The Appeal Board's decision became the final agency action on this matter on November 24, 1980, with the Commission declining to review the decision on its own motion under 10 CFR 2.786.

water accident in the pool," the Zion license should be suspended pending completion of an environmental impact statement (EIS) on high burnup nuclear fuel in the spent fuel storage pools.

We have reviewed the information submitted in PEP's request. In addition, we have reviewed the testimony presented in the Zion spent fuel pool modification proceeding, the decisions of the Atomic Safety and Licensing Boards in the Zion and the Salem proceedings,[4] and the decision of the ASLAB in the Zion case. For the reasons set forth below, PEP's request that Amendments Nos. 52 and 49 to Facility Operating Licenses Nos. DPR-39 and DPR-48, respectively, be suspended, is *denied*.

II. Consideration of Issues Raised in Salem Proceeding

The first portion of the PEP request addresses the Salem Board's questions concerning the accident at TMI and how that accident would affect the spent fuel pool. The Salem ASLB specifically examined the status of the TMI spent fuel pool and found (page 27, paragraph 43 of the October 27, 1980 Initial Decision, that even if there had been fuel in the TMI pool at the time of the accident, the accident would not have affected it. The pool remained accessible despite levels of radiation which were higher than normal, and the equipment for cooling the pool and purifying its water was accesssible at TMI after the accident. The Salem ASLB further found (pages 30 and 31, paragraph 46 of the October 27, 1980 Initial Decision), upon close examination of the Salem spent fuel pool and its cooling, purification, alarm, and ventilation systems, that if an accident occurred at Salem similar to the one at TMI, little, if any, impact would occur on the Salem spent fuel pool.

We have compared the findings of the Salem ASLB and the Zion ASLB (the February 14, 1980 Initial Decision), and have further reexamined the hearing record of the Zion ASLB to determine if any issue considered in the Salem proceedings raises questions about the conclusions reached in the Zion proceeding. We have also reviewed the Zion ASLAB Decision (ALAB-616) dated October 2, 1980 for any bearing it may have on conclusions about the possible impact of a TMI type accident on the Zion spent fuel pool.

The Zion ASLB in its Initial Decision addressed contentions dealing primarily with pool boiling (pages 29-45) and pool drainage accidents (pages 84-85). In addressing these contentions, the Board reviewed

[4] The Initial Decision in the Salem spent fuel pool modification proceeding was issued on October 27, 1980. It is attached as Appendix D. An appeal on portions of that decision is currently pending before the ASLAB.

testimony on the spent fuel pool in many of the areas discussed at the Salem hearing on the TMI type accident. This testimony included the following information:

The fuel building is separate from the auxiliary building where most radiation level increases from a TMI type accident would be expected to occur (outside containment). The containment isolation valves are automatic at Zion and will not reopen automatically. The auxiliary building radiation levels would be expected to increase as the recirculation phase began on the long term cooldown following a LOCA and some increase in the radiation levels of the spent fuel pool building would be expected. The spent fuel pool has been analyzed for a number of accidents that could increase the radiation levels in that area. But in any of these events, a TMI type accident or fuel handling accident, the levels of radiation in the spent fuel pool building would not prevent operation and maintenance of vital systems in that area. The water cooling and purification system for the spent fuel pool is located in an adjacent room shielded from the spent fuel and accessible by the railroad doorway if direct access to the pool is prevented. There are ample sources of makeup water to the spent fuel pool including the demineralized flushing water systems, refueling water storage tank, fire protection systems in the spent fuel pool building, and fire hoses which can draw water from the primary water storage tank, secondary water storage tank, and the service water system.

The spent fuel pool area has three area radiation monitors. Two monitors have fixed setpoints and are read directly in the control room. All have alarms. There is an area particulate monitor which also has an Iodine-133 cartridge for detecting iodine. The ventilation system for the spent fuel pool area combines with ventilation streams from other areas and is monitored for particulates and iodine. An indication from this monitor would automatically divert the effluent from the HEPA filters to the charcoal filters. A fuel cooling accident with high humidity might damage the HEPA filters but they could be replaced even with high radioactivity within the fuel building. The charcoal filter booster fan is manually started from the control room.

From our examination of the Initial Decision by the Zion ASLB and the testimony presented in relation to pool boiling and pool draining, we have determined that the Zion spent fuel pool should be no more affected by a TMI type accident than would the Salem spent fuel pool.[5] We believe the

[5]In its review of the Initial Decision, the ASLAB rejected the State of Illinois' appeal and affirmed the Licensing Board's findings that equipment and controls to assure adequate access to makeup water in the event of a severe accident were accessible under any circumstances. ALAB-616, 12 NRC 425-426 (1980).

Zion spent fuel pool would receive only minimal impact from a TMI Type accident.

We have also concluded, as did the Salem ASLB, that there would be little, if any, effect on the pool with expanded storage capacity, because any effect on the pool from a TMI-type accident would not depend on whether the pool contained additional spent fuel assemblies. Rather, the concern is continued accessibility of the pool and its supporting equipment. As stated above, operation and maintenance of vital systems could continue following an accident.

The second issue considered in the Salem proceeding which PEP sought to have examined for the Zion facilities, dealt with the potential for gross loss of water the spent fuel pool and the effect such a loss of water would have on a pool with expanded storage capacity. The Zion ASLB in its Initial Decision noted that it had posed the question to the applicant and staff to describe any design and/or engineering safety features incorporated in the Zion spent fuel storage pool to decrease the likelihood of a severe pool drainage accident (Initial Decision at 84). The testimony presented included discussions of the cause of such an event and the features available to mitigate the consequences of such accidents. The ASLB specifically found that there are adequate design and engineered safety features incorporated into the Zion Station spent fuel pool which would reduce the likelihood of a severe pool drainage accident, and that those features should preclude the possibility of a severe drainage accident in the Zion Station fuel pool (Initial Decision at 86).

The questions posed by the Licensing Board in the Salem proceeding focused on the differences in consequences between a gross loss of water accident at an unexpanded and an expanded pool. The Salem ASLB findings (pages 31-39 of its Initial Decision dated October 27, 1980) were based on testimony dealing with zirconium fire and propagation, dispersion of fission products, increased risk due to the expansion, clad oxidation and fuel melting, and the relative release potential from old versus new fuel in the spent fuel pool. In general, the Salem Board Initial Decision for this particular issue was that: (1) while further analysis might more precisely define oxidation propagation, it was not needed to convince the Board that there would not be significant releases from the older fuel in comparison to the releases from the newer fuel, (2) gross loss of water is in itself an event of very low probability, and (3) there would not be a great difference between the consequences occasioned by the proposed configuration and those occasioned by the present one.

We have reviewed a number of proposed spent fuel pool modifications, as well as those for the Zion/Salem fuel and design of modification. We find little difference between the Zion and Salem modifications that would make any difference in the results of a gross loss of water accident. The impacts on the expanded pool at Zion have been principally limited to those from the older fuel which will remain in the pool and the releases from those elements are not significant when compared to releases from the recently discharged fuel.

Based on a consideration of the above facts and an extensive review of the record of the Zion proceeding, and the decisions of the Zion and Salem Licensing Boards, we find no factual justification to suspend the licenses or amendments at the Zion Station.

III. Request for an EIS

The last PEP request is to suspend the license amendments pending the completion of a full environmental impact statement on high burnup nuclear fuel in spent fuel storage pools. On April 27, 1979, PEP made a similar 10 CFR 2.206 request. Following issuance of a license amendment permitting extended burnup of four fuel assemblies in the Zion Station Unit 2 to a burnup of approximately 55,000 MWD/MTU (the usual Zion fuel is irradiated to 33,000 MWD/MTU), PEP requested preparation of a full EIS on high burnup fuel, both in the reactor and as a spent fuel waste. After careful consideration of the concerns raised by PEP over use of the higher burnup fuel, the Director of the Office of Nuclear Reactor Regulation denied PEP's request on March 13, 1980.[6] The bases of that denial were conclusions that the potential consequences of the accidents given in the Safety Evaluation Reports supporting the Facility Operating Licenses and the amendments for modification of the spent fuel pool, will not change due to four fuel assemblies in the core being irradiated to burnup to 55,000 MWD/MTU.[7]

In its current petition, PEP requests that the license amendments at issue here be suspended pending completion of a full EIS on high burnup fuel in spent fuel storage pools because: (1) fuel burnup is one of the most important considerations in determining spent fuel pool thermal heat and radiation output, major factors in a loss of water accident in the pool and (2) a contention that in the future, the NRC may allow utilities to go to fuel burnup as high as 55,000-60,000 MWD/MTU. We have no request from

[6]Commonwealth Edison Company (Zion Station, Units 1 & 2, DD-80-11, 11 NRC 496 (1980). A copy of the Director's Decision is attached as Appendix E.

[7]By License Amendment Nos. 59 and 39 dated December 31, 1980, the NRC approved performance of the last cycle of irradiation of the four assemblies in Zion Unit No. 1 on the basis that the safety analysis for the last two cycles of irradiation is directly applicable to either unit. The change of unit would also not change the previous environmental impact determination.

the licensee for the Zion Station to extend the fuel burnup of the four assemblies beyond the 55,000 MWD/MTU target. The need for further data to support such a request and the existing confidence to allow the lead test assemblies to operate for two cycles in nonlimiting core positions was pointed out in our March 13, 1980 denial. We also stated our requirement that a full reload of new fuel design would need a detailed safety review and approval by the NRC and that review would depend upon the data from the Zion and other test programs. Until such time as the test program results are available to extend that data base for higher burnup fuels and Zion applies for approval of extended use of such fuels, no further use of high burnup fuels is contemplated or permitted. Thus, no action has been taken or is contemplated for the Zion plants for which an environmental impact statement or appraisal such as is sought by PEP should be prepared. Consequently, the lack of such a document cannot serve as the basis for suspension of the amendments to the Zion licenses.

PEP has also filed with the NRC a petition for rulemaking, pursuant to 10 CFR 2.802, requesting preparation of a generic EIS for a potential future nationwide program of using high burnup fuel in nuclear reactors (Docket No. PRM-51-6). Any generic aspects of PEP's concerns over use of high burnup fuel will be dealt with in the Commission's response to that petition.

IV. Conclusion

For the reasons set forth above, I have determined that the information submitted by PEP does not alter my conclusion nor call into question the conclusions reached by the Zion Licensing Board that Amendment Nos. 52 and 49 will not significantly affect the health and safety of the public or the quality of the human environment. Suspension of the amendments is not warranted. Therefore, the request of PEP is denied.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555 and at the Local Public Document Room for the Zion Station located at the Zion-Benton Public Library, 2600 Emmaus Avenue, Zion, Illinois 60099. A copy of this document will also be filed with the Secretary of the Commission for its review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

In accordance with 10 CFR 2.206(c) of the Commission's Rules of Practice, this decision will constitute the final action of the Commission 25 days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland
this 18th day of February, 1981.

[Appendixes A-E have been deleted from this publication but are available at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **Docket No. 50-289 (Restart)**

METROPOLITAN EDISON COMPANY, *et al.*
(Three Mile Island Nuclear Station, Unit 1) **March 23, 1981**

Responding to various motions by the licensee, the Commission: (1) denies a request that it permit the restart of Unit 1 upon compliance with certain conditions, prior to completion of the ongoing hearings on whether resumption of plant operation should be authorized; (2) defers ruling on a proposed license amendment seeking to transfer the authority to possess, use and operate the unit from Metropolitan Edison to GPU Nuclear Corporation; (3) grants a motion requesting permission to begin hot functional testing using non-nuclear heat; (4) grants a request that it make a decision on the effectiveness of a licensing board decision within 35 days of issuance of such decision; (5) announces that it will handle on a case-by-case basis requests for deferral of implementation dates for various TMI-2 accident related actions required by NUREG-0737; (6) determines that the issue of the financial ability of the licensee to operate the unit should not be litigated in this proceeding; and (7) directs the Executive Director of Operations to ensure that the staff devote sufficient resources to the proceeding.

ORDER

I. Background

On December 1, 1980 the President of General Public Utilities (GPU), Herman Dieckamp, wrote the Commission requesting it to reconsider and modify its Orders of July 2, 1979, and August 9, 1979, CLI-79-8, 10 CFR 141, pertaining to the restart of Three Mile Island Unit One. Specifically, GPU requested the Commission to modify those orders to permit GPU to restart TMI-Unit 1 prior to the completion of the ongoing adjudicatory hearings. GPU proposed that the Director of the Commission's Office of Nuclear Reactor Regulation be permitted to authorize restart upon determining that Metropolitan Edison has taken all of the actions required of other Babcock and Wilcox (B&W) reactor licensees before B&W reactors were permitted to resume operation following the shut-down orders issued by the Commission in the Spring of 1979. In addition, the Director would be required to determine that Metropolitan Edison had performed satisfactorily those tasks listed in Section II of the Commission's Order that are to be completed prior to restart. Finally, the Director would be required to determine that Unit One was in compliance with the "lessons learned" actions applicable to other B&W plants that have been imposed by the Commission following the accident. GPU emphasized that the restart hearings had taken far longer than the Commission originally contemplated and that the delay in authorizing restart was penalizing the residents of its service areas and its investors.

In addition, Metropolitan Edison has filed three motions with the Commission. On January 26, 1981, it advised the Commission that it had filed an application for an amendment to its operating license which would transfer from Metropolitan Edison to GPU Nuclear Corporation (GPUNC) the authority to possess, use and operate the TMI-1 facility. On that date it filed a motion requesting the Commission to modify its July 2, 1979 Order as appropriate to extend to GPUNC the present restriction on Metropolitan Edison that Unit One be maintained in a cold shutdown condition. On January 26 it also filed a motion requesting the Commission to amend its August 9, 1979 and March 6, 1980 Orders, CLI-80-5, 11 NRC 408, to authorize the TMI-1 Restart Licensing Board to consider the qualifications of GPUNC, rather than Metropolitan Edison, to restart and operate TMI-1.

A February 3, 1981 motion requested the Commission to (1) amend the July 2 and August 9, 1979 Orders to permit hot functional testing of the TMI-1 systems and equipment using non-nuclear heat; (2) modify Section VI of the August 9, 1979 Order to provide that the Commission decision on the effectiveness of a licensing board decision authorizing restart be made within 35 days after issuance of the decision rather than 35 days after issuance of the decision and certification by the Director, Office of Nuclear Reactor Regulation, that those short-term actions required for restart have been completed; and (3) that the August 9 Order be modified to make clear that the Commission has the flexibility to defer until after restart licensee's implementation dates for NUREG-0737 action items where such deferral is consistent with implementation schedules for other operating reactors.

The Commission has received views from the parties to the proceeding on each of the GPU/Metropolitan Edison motions. In addition, Commissioner Ahearne on January 22, 1981 requested the Chairman of the Atomic Safety and Licensing Board conducting this proceeding to provide the Commission with the Board's views on actions the Commission might take to expedite the proceeding. On January 28, 1981, Chairman Hendrie and Commissioner Ahearne requested the Licensing Board to provide, after appropriate consultation with the parties, its best estimate of the future schedule for the proceeding. The parties were also asked to provide their best estimates on when Metropolitan Edison could be expected to be in compliance with a number of specified items which could be required for restart. These requests were considered at a special session of the restart hearing held on February 3, and the Licensing Board submitted its response on February 9. The NRC staff and Metropolitan Edison subsequently provided estimates on when the licensee might be expected to be in compliance with the various items.

II. Rulings on the Motions

1. The December 1, 1980 GPU Request That Restart Not Be Tied To Completion Of The Hearing

The intervenors in the proceeding urged the Commission to deny the motion. Procedurally, they argued that the motion to reconsider the July 2 and August 9 orders is untimely and that the licensee has presented no new facts that would cause the Commission to alter its Orders. On the merits, intervenors argued, *inter alia,* that Unit 1 is not ready to be restarted now and will not be prepared for restart for some time. They noted that modifications to the plant must be completed and the entire facility must be inspected by the NRC staff prior to restart. The operators of the reactor must be requalified and relicensed. It is intervenors' view that by the time all of these tasks can be accomplished, the adjudicatory proceedings will be nearly or totally completed, and therefore it is unnecessary to grant the GPU request. The Commonwealth of Pennsylvania took the position that rather than grant the motion, all efforts should be taken to expedite the hearing process consistent with a full and fair adjudication of the issues. The NRC staff took the position that the Commission has the legal authority to take the action that GPU proposes, but took no position on whether the motion should be granted. The staff believed that the policy issue whether the public interest would be served by permitting operation of the facility prior to the completion of the hearing is best left to the Commission for resolution.

The Commission has reviewed the submissions of the parties, the proposed hearing schedule submitted by the Board, and the status reports on Metropolitan Edison's compliance with various items that could be required for restart. The Commission has denied the GPU request of December 1 because it is unable to find that authorizing restart prior to the completion of the hearing would serve the public interest.

2. Metropolitan Edison's Request That The July 2 and August 9 Orders Be Modified To Substitute GPU Nuclear Corporation for Metropolitan Edison

In its response to these motions, the NRC staff requested the Commission to defer its rulings until the staff had the opportunity to complete its review of Metropolitan Edison's proposed amendments to its operating license. The Commission agrees with the staff that it would be premature to act at this time on the motions and will defer action until it has heard further from the staff.

3. Metropolitan Edison's Request That It Be Authorized To Commence Hot Functional Testing

No party to the proceeding objected to Metropolitan Edison's request and the Commission has decided to permit Metropolitan Edison to begin hot functional testing using non-nuclear heat, subject to any appropriate NRC staff review.[1]

In so ruling, the Commission is not now taking a position on a staff proposal, called to our attention by the Board, to allow low power testing prior to the completion of the hearing, if certain findings are made by the Director, Office of Nuclear Reactor Regulation. We defer ruling on that

[1]The NRC staff has indicated that the licensee must review, in accordance with the requirements of Section 6 of the Technical Specifications for TMI-1, the hot functional testing to be performed to determine whether such activities involve an unreviewed safety question or a change in the facility's technical specifications. Staff noted that if the review produces an affirmative answer, the licensee must submit an application for amendment of its operating license. Should the staff determine that an amendment is required, it should evaluate its suitability to the same extent as any other amendment application.

question until we have had a chance to view other developments in this matter, including the progress of the hot functional testing program.

4. Metropolitan Edison's Request That The Commission Decision On The Effectiveness Of A Licensing Board Decision Be Made Within 35 Days After Issuance Of The Decision

No party opposed the request. The Commission believes the request is reasonable and consistent with the Commission's original intent. The request therefore is granted.

5. Metropolitan Edison's Request That The Commission, Where Appropriate, Defer Until After Restart Implementation Dates For NUREG-0737 Action Items Where Such Deferral Is Consistent With Implementation Schedules For Operating Reactors

The NRC staff has filed testimony in the restart hearing proposing that the licensee be required to implement a number of NUREG-0737 actions on the same schedules that are presently set for operating reactors, although it has generally taken the position that the licensee is to be treated as an applicant for an operating license. In its February 3 motion, licensee asserted that it is prepared to meet the same implementation schedules that are required for operating reactors, but expressed the concern that developments subsequent to the close of the hearing record (for example, delays in the procurement of necessary materials and equipment) may make it impossible for it to meet present schedules on all action items. It therefore requested the Commission to modify the August 9, 1979 Order to make clear that the Commission retains the flexibility to defer until restart, upon the recommendation of the Director of the Office of Nuclear Reactor Regulation, licensee's implementation dates for NUREG-0737 action items where such deferral is consistent with implementation schedules for operating reactors.

The staff did not object to Metropolitan Edison's request, noting that such deferral would be granted by the Commission only after it had heard staff recommendations.

The Commission in its August 9, 1979 Order provided the Licensing Board with the discretion to determine, subject to Commission review, what matters must be resolved prior to restart. In the Order the Commission did not indicate whether Metropolitan Edison is to be treated as a licensee of an operating reactor or as an applicant for an operating license. The Commission believes that Unit One should be grouped with reactors which have received operating licenses, rather than with the units with pending operating license applications. It emphasizes though that it expects the

Board to find to the contrary when the record so dictates. Moreover, the Board should not reopen testimony or otherwise delay the proceeding in any way in order to apply this concept.

The Commission notes in this regard that whether Metropolitan Edison is treated as a licensee or an applicant, there may be items where due dates cannot be met for one reason or another, regardless of which category Unit One is placed in. It is this prospect which prompts the licensee's motion. Where developments occur which affect the ability of the licensee to comply with requirements recommended by the Board or proposed to be imposed by the Commission, the Commission will consider those developments on a case-by-case basis in reaching its decisions on immediate effectiveness and ultimate review of the Board's decision. Notwithstanding language in the original order which could be read to the contrary, we intend to retain our flexibility in this regard. To that extent, the licensee's motion is granted.

III. Expediting the Hearing

The Commission has considered various means to expedite the hearing schedule and is taking one action with that objective in mind. The NRC staff has concluded that the relationship between corporate finance and the technical departments of the licensee is such that financial considerations should not have an improper influence on technical decisions. For this and other reasons, the staff has recommended that financial qualification need not be litigated prior to reaching a decision on the restart of TMI-1. Counsel for the Commonwealth of Pennsylvania, representing the Governor of that State, believes that while it is important for the licensee to demonstrate its financial ability to operate TMI-1 simultaneously with the cleanup of Unit 2, the Commonwealth believes that the return of TMI-1 to commercial operation would improve, rather than impair, the licensee's financial health. For example, return of the unit would produce operating revenues and return of the unit to the utility's rate base also might increase the licensee's credit rating and its ability to obtain capital. Therefore, the Commonwealth supports the staff position. The Pennsylvania Public Utility Commission also does not object to the staff proposal. Metropolitan Edison, of course, would not object to removing financial issues from the proceeding. The intervenors take a contrary position, arguing that financial capability is an important safety issue that should be litigated prior to restart.

The Commission has considered the parties' views and determined that, contrary to the position it took in its August 9 Order, the issue of the licensee's financial qualifications should not be litigated in this proceeding.

The Commission does not believe that, in this particular case, litigation of the issue would be productive. In fact the Commission is of the view that the treatment of financial qualifications in the licensing process as a general matter needs reexamination and is undertaking that examination at this time.

Although the Commission is taking the financial qualification issue out of the hearing, the staff is directed to continue to monitor the licensee's financial resources as long as is necessary and to report any health and safety implications to the Commission.[2]

The Licensing Board has also indicated at various times that the proceeding might be expedited if staff gave the proceeding a higher priority and devoted more resources to it. The Commission has always considered the restart hearing to be one of staff's highest priority items and directs the Executive Director of Operations to ensure that sufficient resources are devoted to this matter so that staff documents, including SER supplements and testimony, may be thorough and timely filed.

It is so ORDERED.*

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 23rd day of March, 1981.

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[2]Commissioner Bradford would not have removed financial qualifications as an issue in this proceeding without first giving those parties sponsoring contentions on this subject an opportunity to describe, in response to the staff SER, what they expected to establish in the course of their presentation or on cross-examination.

*Section 201 of the Energy Reorganization Act, 42 U.S.C. 5841, provides that action of the Commission shall be determined by a "majority vote of the members present." Commissioner Bradford was not present at the meeting at which this Order was approved. Had he been present he would have voted to issue this Order. Accordingly, the formal vote of the Commission is 3-0.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of

ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR NUCLEAR POWER OPERATIONS, 40 CFR 190 **March 26, 1981**

The Commission denies a petition filed by the American Mining Congress for a stay of implementation and enforcement of the Environmental Protection Agency's radiation protection standards, 40 CFR Part 190, as applied to NRC-licensed uranium mills. The Commission also denies a petition filed jointly by several uranium mill operators to defer or rescind NRC regulations imposing EPA's Part 190 requirements on uranium mills.

NRC: ENFORCEMENT OF ENVIRONMENTAL RADIATION PROTECTION STANDARDS

Although the authority to establish generally applicable standards for protection of the environment from radioactive materials resides in the Environmental Protection Agency under Reorganization Plan No. 3 of 1970, 84 Stat. 2086, 42 U.S.C.A. 4321 note, the NRC remains responsible for enforcing those standards at NRC-licensed facilities.

REGULATIONS: VALIDITY

An agency's regulations are presumed valid until the promulgating agency or a court modifies or invalidates them.

RULES OF PRACTICE: STAY OF AGENCY ACTION

In order to obtain a stay, petitioners must satisfy a four-fold test: (a) that they are likely to prevail on the merits; (b) that they will suffer irreparable harm without a stay; (c) that other interested parties would not be substantially harmed by a stay; and (d) that the public interest supports a stay. *Virginia Petroleum Jobbers Ass'n* v. *FPC*, 259 F.2d 921, 925 (D.C. Cir. 1958); *Washington Metropolitan Area Transit Commission* v. *Holiday Tours, Inc.*, 559 F.2d 841, 843-44 (D.C. Cir. 1977).

MEMORANDUM AND ORDER

On October 14, 1980, the American Mining Congress (AMC), on behalf of the uranium mining industry, petitioned the Environmental Protection Agency (EPA) to reconsider and revise the radiation protection standards applicable to uranium mills as codified in 40 CFR Part 190. That same day AMC requested the Nuclear Regulatory Commission (NRC) to stay implementation and enforcement of Part 190 as applied to NRC-licensed uranium mills pending EPA's disposition of AMC's petition and any judicial review of EPA's decision. The EPA standards require that, excluding radon and its daughters, the exposure of any member of the public to planned discharges of radioactive material and radiation from uranium fuel cycle operations be limited to 25-millirems to the whole body, 75-millirems to the thyroid, or 25-millirems to any other organ. The standards were issued in 1977 and became effective on December 1, 1980 for radiation doses arising from uranium milling.[1] AMC contends that scientific and technical data which have become available since EPA promulgated Part 190 in 1977, indicate that the standard for uranium mills is impracticable, unachievable at reasonable cost, and is based on a defective record.

AMC renewed its request to the Commission on December 4, 1980, contending that the proposed program for enforcing Part 190 contains unacceptable uncertainties in compliance determination procedures. Among these alleged uncertainties are: the extent of NRC's reliance on computer codes for calculating radiation doses; difficulties inherent in environmental monitoring; and the lack of a definitive determination by NRC whether or not doses from Lead-210 would be included for purposes of determining compliance with Part 190. AMC believes that these alleged

[1]EPA's generally applicable radiation standards for nuclear fuel cycle activities were issued in 1977 pursuant to the authority transferred to the EPA by Reorganization Plan No. 3 of 1970. 42 *Fed. Reg.* 2858 (January 13, 1977).

uncertainties support its request that NRC should delay initiation of additional radiation monitoring requirements.

In a separate action, Kerr-McGee and several other uranium mill operators (Operators) jointly filed late comments on the Commission's proposed amendments to 10 CFR Part 20. 45 Fed. Reg. 26072-73 (April 17, 1980). The proposed amendments would explicitly codify in the NRC's regulations the existing Part 190 requirements and would add certain reporting requirements. Recognizing that these comments were over four months late,[2] the Operators requested that in the alternative, their filing be treated as a petition to rescind any NRC regulation imposing Part 190 requirements on uranium mills. Operators contend that Part 190 is unlawful for a variety of reasons related to its promulgation by EPA. In the alternative, they request that NRC defer implementation of Part 190 until December 1982 to provide the mill operators with time to determine whether they are in compliance with Part 190 requirements and to effect changes, if necessary, or shut down in an orderly manner if they cannot achieve compliance. In addition, Operators suggest that implementation of Part 190 be coordinated with regulations promulgated pursuant to the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA). Finally, they oppose as impracticable and unduly expensive the proposed reporting requirements which would be added to 10 CFR 20.405. The Commission has determined that Operators' filing should be treated as a motion to stay NRC's implementation and enforcement of Part 190. For the reasons discussed below, the Commission has determined that implementation of Part 190 at NRC-licensed uranium mills should not be stayed.

I.

The Commission's disposition of the claims by AMC and Operators requires an understanding of the relationship between the Commission and the EPA in the present setting. Reorganization Plan No. 3 of 1970, 84 Stat. 2086, 42 U.S.C.A. 4321 note, (Plan) transferred to EPA the AEC's authority to establish generally applicable environmental standards for the protection of the general environment from radioactive materials. However, as noted in the President's message transmitting the Plan to Congress, the AEC remained responsible for enforcing the standards at AEC-licensed facilities. This allocation of agency responsibility was memorialized in a Memorandum of Understanding between the AEC and EPA entered into in 1973. 38 *Fed. Reg.* 24936 (September 11, 1973).

[2]Although the comment period closed on June 16, 1980, the Operators did not submit their comments until October 28, 1980.

AMC has petitioned EPA to reopen the rulemaking proceeding on Part 190, reconsider the radiation protection standards, and revise those standards as they apply to uranium milling. In addition, AMC has requested EPA to stay the effective date of the standards as applied to uranium milling pending disposition of the petition. In support of this petition, AMC has submitted to EPA a substantial package of technical information. Because EPA is the agency authorized to issue generally applicable radiation standards, it is the agency responsible for deciding whether that information supports a reconsideration of the standards. Thus, EPA must decide the merits of AMC's petition regarding Part 190 as applied to uranium milling.

It is well established that each agency's regulations are presumed valid until the promulgating agency or a court modifies or invalidates them. This agency does not sit as a reviewing court for a sister agency's regulations. *Cf.* 7 NRC 1, 27 (1978) (NRC recognition of collateral estoppel effect of findings by EPA). Moreover, to the extent that Part 190 is challenged on the basis of new information, it would be unseemly for this agency to second-guess the agency responsible for evaluating and responding to that information. Thus, unless and until EPA or a court modifies or rescinds Part 190, the NRC will proceed on the basis that the regulations are fully in force and will initiate implementation and enforcement of this regulation at NRC-licensed uranium mills. To the extent that the motions now pending before us are based on challenges to Part 190, these motions are denied.

II.

Wholly apart from our reluctance to pass upon the merits of AMC's substantive claims, a review of the current situation shows that there is also no basis for now staying initiation of NRC's compliance program. In order to obtain a stay, the petitioners must satisfy a four-fold test: (a) that they are likely to prevail on the merits; (b) that they will suffer irreparable harm without a stay; (c) that other interested parties would not be substantially harmed by a stay; and (d) that the public interest supports a stay. *Virginia Petroleum Jobbers Ass'n.* v. *FPC,* 259 F.2d 921, 925 (D.C. Cir. 1958); *Washington Metropolitan Area Transit Commission* v. *Holiday Tours, Inc.,* 559 F.2d, 841, 843-44 (D.C. Cir. 1977). Petitioners have utterly failed to sustain their burden on the last three factors.

The possibility of irreparable injury is a critical factor in analyzing any request for a stay and it is absent here. *Ashland Oil* v. *FTC,* 409 F. Supp. 297, 307 (D.C.C. 1976), *aff'd.* 548 F.2d 977 (D.C. Cir. 1976). No such injury will result from initiation of the NRC's program.

Beginning in 1977, the NRC began upgrading its regulation of uranium mills to require more stringent control of radioactive effluents. These actions were taken pursuant to the Commission's authority under NEPA to mitigate environmental impacts and its authority to require licensees to maintain exposures as low as reasonably achievable (ALARA). 10 CFR Part 20. In 1977, the staff issued interim performance objectives for uranium mill tailings management. These objectives required licensees, among other things, to control the blowing of mill tailings, the greatest potential source of off-site exposures.[3] In addition, licensees were required to maintain an environmental and effluent monitoring program.[4] As new licenses were granted and existing licenses renewed or amended, programs meeting the interim performance objectives and monitoring requirements were imposed on most licensed facilities. This evolution in the regulation of uranium mills has led to the establishment of a significant base for attaining compliance with Part 190.

On November 7, 1980 representatives of the American Mining Congress met with members of the NRC Uranium Recovery Licensing Branch. At that meeting, the NRC staff explained that it is instituting a phased program for implementing Part 190 at NRC-licensed mills. The specific elements of the first phase of that program were also described in a November 13, 1980 letter from the NRC Solicitor to Mr. Boggs of the AMC.[5] That letter is hereby incorporated in this order. As explained in that letter, the only new major activity which will be required initially at NRC licensed mills is the additional calculation of doses from monitoring data to determine whether a mill's contribution to the level of radioactivity in its vicinity exceeds the limits in 40 CFR Part 190. First reports by NRC licensees will not be due before spring 1981. Until the NRC has reviewed and analyzed those reports, it does not expect to take additional action against or impose additional requirements on licensees for the purposes of compliance with Part 190 requirements.[6] Thus, initiation of the proposed program to implement Part 190 will not result in irreparable injury to NRC uranium mill licensees. Accordingly, the Commission does not believe that a stay of the implemenation is warranted.

AMC's allegation of uncertainties in NRC compliance determination procedures does not alter this conclusion. Although AMC contends that the

[3]Branch Technical Position - Uranium Mill Tailings Management (May 13, 1977).

[4]Branch Position Papers - "Operational Radiological Environmental Monitoring Programs for Uranium Mills"; and "Preoperational Radiological Environmental Monitoring Programs for Uranium Mills" (January 9, 1978).

[5]Copies of that letter were distributed to NRC licensees who attended a November 14, 1980 briefing on the NRC's implementation of Part 190.

[6]For these reasons, it is also unnecessary to delay initiation of the Part 190 program as requested by Operators.

extent of NRC's reliance on computer models is uncertain, NRC's intention to rely on actual monitoring data (and not computer models) to determine compliance for an operating mill was clearly stated at the November 14, 1980 meeting attended by licensees and representatives of the AMC.[7] AMC's real concern in this regard appears to be that the NRC may decide to use computer models to determine compliance in the future. At this time, the NRC staff does not foresee circumstances that would lead to the use of models instead of actual monitoring data for determining compliance at operating mills. Moreover, the mere possibility of a future change in NRC policy does not constitute imminent irreparable injury supporting AMC's request to stay implementation of the environmental monitoring program. *Pharmaceutical Manufacturers Ass'n* v. *Weinberger,* 401 F. Supp. 444, 449 (D.D.C. 1975); *American Medicorp, Inc.* v. *Continental Illinois National Bank and Trust Company of Chicago,* 475 F. Supp. 5, 7 (D. Ill. 1977).

AMC also contends that uncertainties in environmental monitoring prevent NRC from accurately determining compliance, or noncompliance, with Part 190. Compliance determination based on environmental monitoring is comprised of four elements: data collection; data analysis; dose calculations; and, if necessary, identification of radioactive sources. Data collection, for operating uranium mills, includes the sampling of effluents from the yellow cake dryer and packaging and other stacks; and the collection of samples of air, water, soil, sediment, vegetables, fish, and food which may contain radioactive effluents. Data analysis is the determination of the concentrations of various radioactive elements (radionuclides) in collected samples. The combined activities of data collection and analysis are often referred to as radiological effluent and environmental monitoring. For uranium mills, NRC Regulatory Guide 4.14 describes a program for radiological effluents and environmental monitoring.[8] The guide describes programs acceptable to the NRC staff for measuring releases of radioactive materials to the environment from typical uranium mills. Thus, there is no uncertainty regarding the environmental monitoring program which will satisfy NRC. Moreover, the acceptable levels of accuracy of these

[7]Predictive computer models have a necessary but limited use. In order to meet the requirements of NEPA, predictive computer models are used by the staff in licensing actions to provide estimates of potential impacts of a new or expanded operation where no other information (e.g., actual environmental monitoring data) is available. A determination of the actual impact can only be made through the assessment of actual environmental monitoring data collected during the operation. That compliance with radiation protection standards, such as 40 CFR 190, is to be determined through actual monitoring data has been stated in the environmental impact statements and assessments of uranium mills prepared by the staff over the past several years.

[8]NRC Regulatory Guide 4.14 - Revision 1, Radiological Effluent and Environmental Monitoring at Uranium Mills (Apr. 1, 1980). This revised Regulatory Guide includes responses to public comments.

monitoring programs have been clearly spelled out. NRC Regulatory Guide 4.15[9] describes a quality assurance program which, if implemented, will ensure that the data obtained will satisfy NRC requirements regarding accuracy.

The next step in the compliance procedure is dose calculation, the determination of the dose which would result from exposure to the radiation and radioactive effluents measured by the environmental monitoring program. Staff has provided licensees a clear discussion of the method to calculate doses from environmental monitoring data.[10] Included in this discussion are tables which provide a ready reference set of conversion factors for calculating doses from measured concentrations of radionuclides.[11] Moreover, the staff has also provided a set of parameters for occupancy and local food production and consumption. Licensees can either use these parameters to calculate doses, or can provide their own parameters based on actual measurements. Thus, there is no uncertainty regarding dose calculations which will be acceptable to the NRC for determining compliance with Part 190.

The compliance determination procedure provided by the staff also shows that there is no need to defer implementation of Part 190 until December 1982 as requested by Operators. Their request was based on the mistaken belief that compliance with Part 190 would be determined solely by predictive models. Because doses will be determined in large part by straightforward calculations based on conversion tables provided by the staff, substantial periods of time will not be required for their calculations.

If the dose calculation demonstrates compliance with 40 CFR Part 190 limits, then nothing more needs to be done. On the other hand, if the calculated dose is near or exceeds Part 190 limits, then the compliance determination procedure provides for the identification of radiation sources. This step is designed to separate dose contributions from activities subject to Part 190 from unregulated sources, especially uranium mines. Dose contributions from unregulated sources must be subtracted from the total measured dose before NRC can make a final determination of

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[9]NRC Regulatory Guide 4.15 - Revision 1, Quality Assurance for Radiological Monitoring Programs (Normal Operation) - Effluent Streams and the Environment (February 1979).

[10]Compliance Determination Procedures for Environmental Radiation Protection Standards for Uranium Recovery Facilities - 40 CFR 190 (December 1980).

[11]These tables are based on conversion factors contained in Draft Regulatory Guide RH802-4, Calculational Models for Estimating Radiation Doses to Man from Airborne Radioactive Materials Resulting From Uranium Milling Operation (May 1979).

compliance with Part 190 limits. Preliminary dose data analyzed by the staff suggest that this step will be necessary at only a few mills.[12] An example of the procedure which will be followed was described in detail at the November 14, 1980 meeting between the staff and uranium mill licensees. The NRC believes that such procedures can adequately resolve uncertainties about the radiation contributions from regulated and unregulated sources in the vicinity of uranium mills for which the measured dose is near or above Part 190 limits.

In summary, then, NRC has provided licensees with substantial guidance for determining compliance with Part 190 limits. Staying implementation of the compliance determination procedures would only delay identification of the sources contributing to a measured radiation dose and, thus, prevent the resolution of any uncertainties regarding compliance with Part 190. Moreover, AMC will have ample opportunity to challenge any compliance determination based on environmental monitoring data.[13] Thus, AMC's arguments are premature to the extent that they contend that uncertainties require the Commission to stay initiation of the environmental monitoring program.

Operators contend that it would be wasteful of resources to require compliance with Part 190 now because they will be required subsequently to comply with different more stringent regulations implementing the UMTRCA. The contention is without merit. Both sets of regulations are now in effect and they are consistent with each other. The UMTRCA regulations are based, in part, on the Part 190 standard because they require that an operational monitoring program shall be conducted to measure or evaluate compliance with applicable standards and regulations. 10 CFR Part 40, Appendix A, Criterion 7. Thus, initiation of an environmental monitoring program for the purposes of compliance with Part 190 will also provide an element of compliance with UMTRCA regulations. Moreover, UMTRCA regulations are consistent with performance objectives previously used in upgrading tailings management practices. Control measures instituted to meet performance objectives not only serve to meet UMTRCA

[12] On the basis of a recent analysis, staff has concluded that it is likely that each NRC licensed uranium mill is operating in such a fashion that the Part 190 standard is being met. *See*, 40 CFR 190 Compliance Assessment for NRC Licensed Uranium Recovery Facilities As of December 1, 1980. (February 1981).

[13] AMC also contends that NRC's determination of compliance with Part 190 is uncertain because the staff has not yet determined whether the dose due to Lead-210 will be included in compliance calculations. Part 190 explicitly excludes from compliance calculations the dose due to radon and its daughters (elements which result from the radioactive decay of radon). Lead-210 is a radon daughter. Accordingly, the staff will not include the dose due to Lead-210 in calculations utilizing the environmental monitoring data to determine compliance with Part 190.

regulations but also contribute to achieving compliance with Part 190 limits. Finally, initiation of the Part 190 program will not require Operators to expend resources now on other remedial actions.[14] Thus, the Commission finds no merit in the suggestion that initiation of the Part 190 program should be delayed pending implementation of the UMTRCA regulations.

The public could be harmed by a stay of the NRC program to ensure compliance with Part 190. Members of the public who live in the vicinity of uranium mills are exposed to radiation from those mills. Those persons could be unnecessarily exposed to excess levels of radiation if the NRC delays implementation of its program to enforce compliance with Part 190.

The public interest in protection of the public health and safety also warrants denial of a stay. The importance of protecting the public from excessive levels of radiation should not be minimized. Part 190 limits already apply to most other activities which comprise the uranium fuel cycle. Initiation of a program to enforce Part 190 at uranium mills will now simply ensure that the public's exposure to radiation from certain aspects of uranium milling is no greater than its exposure to other regulated activities in the uranium fuel cycle. Moreover, the Commission's interest in establishing comprehensive and uniform standards for protecting the public from excessive doses of radiation would be compromised by a stay. Thus, the Commission believes that delay in implementing enforcement of Part 190 is not justified.

[14]Operators have opposed the non-compliance reporting requirements proposed in 10 CFR 20.405 as unauthorized, unnecessary, impracticable, and unduly expensive. The Commission need not consider these contentions at this time because initiation of the NRC's implementation program will not include compliance with this requirement. As explained by the licensing staff at its November 14 meeting with uranium mill operators, during this phase of the program licensees will be explicitly exempted from this requirement because uranium millers will be required to provide quarterly reports during Phase 1 of the enforcement procedure. This exemption will be contained in the orders the NRC will issue soon implementing initiation of the program. Moreover, the staff also explained at the November 14 meeting that once the enforcement program is established, the NRC expects to establish simple and standardized reporting requirements which will minimize costs and staff time devoted to compliance assessment. In any event, the cost of complying with reporting requirements does not constitute irreparable injury. *A. O. Smith Corp.* v. *F.T.C.*, 530 F.2d 515, 527-28 (3d Cir. 1976).

For these reasons, the Commission denies the motions by AMC and the Operators to stay implementation and enforcement of 40 CFR Part 190 as applied to NRC-licensed uranium mills.

It is so ORDERED.*

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, DC,
this 26th day of March 1981.

*Section 201 of the Energy Reorganization Act, 42 U.S.C. §5841, provides that action of the Commission shall be determined by a "majority vote of the members present." Commissioner Bradford was not present when this item was affirmed, but had previously indicated his approval of the Order. Had Commissioner Bradford been present, he would have affirmed his prior vote. Accordingly, the formal vote of the Commission was 3-0 in favor of the decision.

Cite as 13 NRC 309 (1981) **ALAB-635**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-466**

HOUSTON LIGHTING & POWER COMPANY
(Allens Creek Nuclear Generating Station, Unit No. 1) **March 10, 1981**

The Appeal Board denies an intervenor's motion seeking directed certification of an interlocutory ruling of the Licensing Board.

RULES OF PRACTICE: DISCRETIONARY INTERLOCUTORY REVIEW

Discretionary interlocutory review will be undertaken only where the ruling below either (1) threatens the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated by a later appeal or (2) affects the basic structure of the proceeding in a pervasive or unusual manner.

APPEARANCES

Mr. James Morgan Scott, Jr., Sugar Land, Texas, for the intervenor Texas Public Interest Research Group.

MEMORANDUM AND ORDER

1. We have before us the January 15, 1981 motion of intervenor Texas Public Interest Research Group (TexPIRG) seeking directed certification of

an interlocutory ruling of the Licensing Board. See 10 CFR 2.718(i), *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-271, 1 NRC 478, 482-83 (1975). The ruling in question — contained in an unpublished order entered on September 15, 1980 — rejected TexPIRG's position that the NRC staff should be required to prepare a supplement to its Final Environmental Statement for the Allens Creek facility. The supplement envisioned by TexPIRG would address the environmental impacts of so-called "Class 9 accidents".

In an unpublished order entered on January 19, 1981, we announced that we would withhold decision on the directed certification motion to await Licensing Board action on the simultaneously filed request that it either (1) refer its September 15 ruling under 10 CFR 2.730(f) or (2) certify the question decided in the ruling under 10 CFR 2.718(i). On March 2, 1981, the Licensing Board acted: in an unpublished memorandum and order, it denied the request. The directed certification motion is thus now ripe for our determination.

2. Almost four years have elapsed since our notation that:

> Almost without exception in recent times, we have undertaken discretionary interlocutory review only where the ruling below either (1) threatened the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated by a later appeal or (2) affected the basic structure of the proceeding in a pervasive or unusual manner.

Public Service Co. of Indiana (Marble Hill Nuclear Generating Station), ALAB-405, 5 NRC 1190, 1192 (1977) (footnote omitted). That standard still prevails. In this instance, it is not met.

The fact that TexPIRG waited four full months before seeking interlocutory review of the September 15 ruling gives a hollow ring to any claim on its part that the ruling threatens it with irreparable impact both *immediate* and serious. Beyond that, it has not been satisfactorily explained why appellate scrutiny of the ruling cannot abide the event of the initial decision and (if dissatisfied with the result reached in that decision) TexPIRG's appeal from it. To be sure, if the ruling were found erroneous on such an appeal, the consequence might well be a vacation of the initial decision and a remand to the Board below. But the same possibility exists with respect to all interlocutory determinations made by licensing boards on matters which have a potential bearing upon the outcome of the proceeding. If, standing alone, that consideration were enough to justify interlocutory review, it would perforce follow that virtually every significant licensing board ruling during the course of a proceeding would be a fit candidate for immediate appellate examination. It is scarcely necessary to expound at any length upon why a drastic alteration of existing practice to accommodate that thesis would be intolerable — as well as in derogation of the Commission's explicit policy disfavoring interlocutory review. 10 CFR 2.730(f).[1]

Directed certification *denied*.
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

[1]No serious claim has been, or could be, made that the ruling in question has "affected the basic structure of the proceeding in a pervasive or unusual manner".

Cite as 13 NRC 312 (1981) **ALAB-636**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Thomas S. Moore, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-155 OLA**
(Spent Fuel Pool Expansion)

CONSUMERS POWER COMPANY
(Big Rock Point Nuclear Plant) **March 31, 1981**

The Appeal Board reverses a referred ruling of the Licensing Board (LBP-80-25, 12 NRC 355) which holds that, in conjunction with agency consideration of an application for a license amendment to expand the spent fuel pool of a facility that has never undergone environmental review, Section 102(2)(C) of the National Environmental Policy Act (NEPA) requires an environmental impact statement (EIS) covering the effects of the spent fuel pool expansion as well as the additional term of operation that such expansion would permit. The Appeal Board rules that unless the proposed spent fuel pool expansion will change reactor operation, the agency's environmental review for this license amendment need not consider the continued plant operation that the expanded pool might permit. The Appeal Board also directs the Licensing Board to reconsider its companion order to the staff to prepare an EIS on the spent fuel pool itself.

RULES OF PRACTICE: IMPROPER SUPPLEMENTAL ARGUMENT

Material tendered by a party without leave of the Appeal Board, after an appeal has been submitted for decision, constitutes improper supplemental argument.

NEPA: RULE OF REASON

Agency interpretation and application of NEPA is subject to a "rule of reason." *Natural Resources Defense Council, Inc.* v. *Morton*, 458 F.2d 827, 834, 837 (D.C. Cir. 1972).

NEPA: SCOPE OF REVIEW

Where the environmental status quo associated with plant operation will remain unchanged by a proposed license amendment to expand a spent fuel pool, the environmental review for such license amendment need not consider the effects of continued plant operation. See *Committee for Auto Responsibility* v. *Solomon*, 603 F.2d 992 (D.C. Cir. 1979), *cert. denied*, 445 U.S. 915 (1980).

NEPA: PURPOSE OF INQUIRY

The purpose of a NEPA inquiry is to identify aspects of a project that can still be changed to mitigate possibly detrimental environmental effects. *Virginians For Dulles* v. *Volpe*, 541 F.2d 442, 446 (4th Cir. 1976).

NEPA: NEED FOR ENVIRONMENTAL REVIEW

Licensing boards should not decide whether a given action is one that significantly affects the environment without the record support provided by the staff's environmental review.

NEPA: CONSIDERATION OF ALTERNATIVES (SECTION 102(2)(E))

Boards must be provided with some factual record before they can determine whether a proposal "involves unresolved conflicts concerning alternative uses of available resources." NEPA, Section 102(2)(E).

NEPA: CONSIDERATION OF ALTERNATIVES (SECTION 102(2)(E))

Section 102(2)(E) of NEPA is not limited to major federal actions having a significant effect on the environment; it may require consideration of alternatives in a case where the preparation of an EIS is not required.

APPEARANCES

Mr. Joseph Gallo, Washington, D. C., and **Mr. Peter Thornton**, Chicago, Illinois, for the applicant, Consumers Power Company.

Ms. Janice E. Moore for the Nuclear Regulatory Commission staff.

Mr. John P. O'Neill II, Maple City, Michigan, intervenor *pro se.*

Mr. Herbert Semmel, Washington, D.C., for intervenors Christa-Maria, JoAnne Bier, and Jim E. Mills

Mr. John A. Leithauser, Levering, Michigan, non-party participant *pro se.*

Messrs. C. Foster Knight and **John F. Shea III** and **Ms. Gail Osherenko**, Washington, D. C., for *amicus curiae* Council on Environmental Quality.

DECISION

In its memorandum and order of September 12, 1980, the Licensing Board held that where a reactor has never undergone an environmental review, Section 102(2)(C) of the National Environmental Policy Act (NEPA), 42 U.S.C. 4332(2)(C), requires the preparation of an environmental impact statement (EIS) covering the effects of a proposed spent fuel pool expansion and the additional term of reactor operation that such expansion would permit. LBP-80-25, 12 NRC 355, 359, 366.[1] We accepted the

[1]Section 102(2)(C) provides, in pertinent part:

> The Congress authorizes and directs that, to the fullest extent possible: * * * all agencies of the Federal Government shall —
>
> > include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on —
> >
> > (i) the environmental impact of the proposed action,
> >
> > (ii) any adverse environmental effects which cannot be avoided should the proposal be implemented,
> >
> > (iii) alternatives to the proposed action,
> >
> > (iv) the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity, and
> >
> > (v) any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented. * * *

Licensing Board's referral of this interlocutory ruling, and the parties have briefed and argued the matter. See 10 C.F.R. 2.730(f).[2]

For the reasons explained below, we conclude that the Board erred. Unless the proposed spent fuel pool expansion will change reactor operation, the agency's environmental review for this license amendment need not consider the continued plant operation that the expanded pool might permit.

I.

A.

Intervenor John O'Neill II, in his Contention VIII, first raised the issue of continued plant operation occasioned by the proposed expansion of Big Rock's spent fuel pool.[3] At a special prehearing conference on December 5, 1979, Mr. O'Neill contended further that a cost-benefit analysis would show that closing the plant would not cause undue hardship because the small amount of power it produces could be easily replaced. Tr. 215-216. Applicant Consumers Power Company (CPC) argued, on the other hand, that continued plant operation is not the object of the proceeding and thus should not be considered. CPC also noted that it expected the staff to issue an environmental impact appraisal (EIA) with a "negative declaration" (*i.e.,* no significant environmental impact from the expanded spent fuel pool itself). Tr. 217. Apparently concerned about the adequacy of an EIA because Big Rock was licensed before the enactment of NEPA and thus had never had an EIS, the Licensing Board deferred ruling on Mr. O'Neill's contention. Instead, it requested the parties to brief the following question (LBP-80-4, 11 NRC 117, 133 (1980)):

> Where the facility has never been subjected to a National Environmental Policy Act of 1969 (NEPA) review because it was licensed before NEPA, does a license amendment which would permit the continued operation of the facility either require or permit considering a cost-

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[2]We invited the Council on Environmental Quality (CEQ) to participate as an *amicus curiae.* It accepted our invitation and appeared in support of the Licensing Board's decision.

[3]Mr. O'Neill's Contention VIII, as submitted, stated:

> Granting of the license is the only way the plant can operate past the year 1981 as things stand now, and thus allow an extension of plant activity that would otherwise be halted. Hence, it is a tacit approval of such extended operation, and should include a review of general plant safety.
>
> - The Kemeny Commission has recommended "periodic relicensing of existing atomic plants on the basis of hearings, inspections and performance criteria."
>
> Big Rock produces very little electricity compared to modern nuclear generators, 73 megawatts at most; the closing of Big Rock would not cause great hardship.

benefit analysis or the need for power in the license amendment proceeding, notwithstanding that the staff may issue a negative declaration?

Although the staff had not yet issued any environmental statement, all of the parties briefed the question, assuming *arguendo* that the staff would eventually issue a "negative declaration" EIA similar to those in other spent fuel pool proceedings.[4] In its decision, the Licensing Board first concluded that the full environmental review of both the expanded spent fuel pool and the continued plant operation it would permit would not result in an illegal retroactive application of NEPA. Although the Board determined that the continued operation of Big Rock was not an "ongoing Federal project" — which in many cases necessitates an EIS — it viewed the proposed license amendment "as requiring a new Federal action for the sole purpose of enabling [applicant] to make a fuller utilization of its operating license than it could otherwise." 12 NRC at 359. It further characterized this as "a new Federal action ... required to enable a private party to complete a project initiated prior to the effective date of NEPA," citing *Minnesota Public Interest Research Group* v. *Butz,* 498 F.2d 1314 (8th Cir. 1974) [hereinafter "MPIRG"]. *Id.* at 360.

The Licensing Board also concluded that ordering an EIS for the continued operation of Big Rock would not conflict with the holdings of *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263 (1979), and *Northern States Power Co.* (Prairie Island Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41 (1978), *remanded on other grounds sub nom. Minnesota* v. *Nuclear Regulatory Commission,* 602 F.2d 412 (D.C. Cir. 1979). It viewed the Appeal Board's finding that no EIS was required in either *Trojan* or *Prairie Island* "as being based merely upon the principle that NEPA does not require the preparation of duplicative environmental reviews for every major Federal action." *Id.* at 361. Unlike Big Rock, both the Trojan and Prairie Island reactors were the subject of an EIS prepared for their original permit and licensing proceedings. But here the Licensing Board reasoned that "because no environmental review was made at the time of the granting license, there would be no duplication, and the Federal action sought, for the sole purpose of permitting a fuller utilization of the license, must be assessed." *Ibid.*

Another important element of the Licensing Board's decision was its finding that approval of this license amendment — which ostensibly would

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[4]The participating parties include a group of intervenors (Ms. Christa-Maria, *et al.*) as well as the applicant, NRC staff, and Mr. O'Neill. The Licensing Board also permitted Mr. John Leithauser to brief the NEPA question even though it had already denied his petition to intervene. 11 NRC at 133, 123.

permit applicant "to utilize a greater term of the license than would otherwise be possible — would be a major federal action with a significant impact on the environment. *Id.* at 363-364. The Board directly linked the spent fuel pool expansion with continued operation. Then, by noting that "[i]t is well-established that operation of a nuclear power plant has a significant effect upon the human environment," *Id.* at 364 n.2, the Board easily made the critical finding that triggers NEPA's EIS requirement. The Licensing Board thus ordered the staff to prepare an EIS but explicitly limited the scope of that analysis to the incremental environmental effects of the pool expansion and the increased term of plant operation. *Id.* at 365, 366.[5]

Intervenors had also argued that Section 102(2)(E) of NEPA, 42 U.S.C. 4332(2)(E), requires the consideration of alternatives to the pool expansion (including plant shutdown), even when no EIS is required.[6] But because the Licensing Board grounded its holding on Section 102(2)(C) of NEPA, which it characterized as more comprehensive, it found it unnecessary to determine the independent applicability of Section 102(2)(E) to this case. *Id.* at 359. After accepting the Board's referral, however, we directed the parties to brief the Section 102(2)(E) point along with their other arguments on appeal.

B.

Before us, applicant CPC first argues that the Licensing Board misconstrued the scope of this proceeding. The proposal, it asserts, is simply to expand Big Rock's spent fuel pool capacity, not to continue plant operation. The notice of hearing (which referred only to the spent fuel pool) and the Appeal Board decisions in *Trojan* and *Prairie Island* assertedly reinforce CPC's "limited scope" argument. The "relevant inquiry," in CPC's view, "is not whether the original license was preceded by NEPA review, but whether the amendment sought significantly changes the environmental impacts of the project as originally approved." Br. 12.

[5]The Board also admitted Mr. O'Neill's Contention VIII and restated it as follows (12 NRC at 366):

> An environmental review of the proposed spent fuel pool expansion is necessary under Section 102(2)(C) of NEPA and would indicate that the environmental costs of this expansion exceed the benefits.

[6]Section 102(2)(E) provides:

> The Congress authorizes and directs that, to the fullest extent possible: * * * all agencies of the Federal Government shall — study, develop, and describe appropriate alternatives to recommended courses of action in any proposal which involves unresolved conflicts concerning alternative uses of available resources.

Second, in an effort to distinguish cases such as *MPIRG, supra,* on which the Licensing Board relied, CPC contends that the post-NEPA modifications of pre-NEPA projects at issue there — unlike the Big Rock spent fuel pool expansion — were themselves further major federal actions necessarily requiring a full environmental review. It therefore asserts that scrutiny of Big Rock's continued operation, which was originally licensed in 1962, would result in an improper retroactive application of NEPA.[7]

Third, assuming *arguendo* that the Board correctly concluded that continued plant operation must be considered in an environmental review, CPC argues that the Board failed to comply with the Commission's regulations, 10 C.F.R. 51, by not awaiting the staff's environmental analysis. Consequently, CPC contends that this "procedural irregularity" (Br. 20) deprives the Board's conclusion of factual and record support.

Finally, CPC urges us to decide whether Section 102(2)(E) is applicable. Relying on the brief it filed before the Licensing Board, CPC contends that this section of NEPA does not apply where, as here, there are no "unresolved conflicts concerning alternative uses of available resources" and the involved proposal has only "negligible" environmental effects.

The NRC staff argues that the Licensing Board's decision does not comport with the Commission's regulations (10 C.F.R. 51), which, in both their present and proposed (amended) form, do not require an EIS for a spent fuel pool expansion. Relying on *Andrus* v. *Sierra Club,* 442 U.S. 347 (1979), the staff contends further that requiring an EIS in this proceeding would improperly "trivialize" NEPA. The staff asserts that considering the environmental impacts of plant operation, whether past or future, would result in an illegal retroactive application of NEPA as well.

The staff views the Licensing Board as lacking authority to order the preparation of an EIS before the staff submits its own independent appraisal. Like CPC, it also argues that the scope of the proceeding is defined by the notice of hearing. The environmental effects of continued plant operation, it argues, are beyond the scope of an application to install additional racks in a spent fuel pool. With respect to Section 102(2)(E), the staff urges us, if necessary, to determine its relevance, rather than to remand the issue to the Licensing Board. The staff believes that although Section 102(2)(E) may require consideration of alternatives regardless of whether an EIS is required, there are no "unresolved conflicts" in this proceeding to invoke that requirement.[8]

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[7]Big Rock received a "provisional" operating license in 1962, followed by a full-term operating license in 1964.

[8]The staff points out (Br. 30), however, that it "has traditionally considered some alternatives to spent fuel pool expansion in the environmental impact appraisals which have been issued" and that it "presently intends to consider alternatives in whatever environmental document it

In support of the Licensing Board, Ms. Christa-Maria, *et al.,* argue that because authorization of this spent fuel pool expansion would permit the plant to continue operating, it is a major federal action with a significant impact on the environment.[9] Because the plant's operation has never been evaluated for environmental impact, NEPA, in their view, requires such an evaluation now (in the form of an EIS) for continued reactor operation. Intervenors thus emphasize that such an environmental review would not be the duplicative one held to be unnecessary in the *Trojan* and *Prairie Island* decisions.

Intervenors also contend that requiring an EIS on continued plant operation is not a retroactive application of NEPA because (1) the EIS ordered by the Licensing Board would concern only *prospective* plant operation, and (2) the government is being asked here, as in *MPIRG, supra,* to approve a further major action " 'required to enable a private party to complete a project initiated prior to the effective date of NEPA.' " Br. 16. Ms. Christa-Maria, *et. al.,* vigorously dispute the arguments of the staff and CPC that the Board exceeded either its jurisdiction or the proper scope of the hearing, arguing not only that NEPA requires an EIS in this case, but also that the Board and Commission have discretion to order its preparation. *Id.* at 21-24. They point out that no agency regulation "deprives the Licensing Board of its authority to require an EIS," and they note NEPA's broad mandate to federal agencies to carry out its provisions "to the fullest extent possible" (42 U.S.C. 4332). *Id.* at 23, 7. With respect to the notice of hearing, intervenors state that its purpose "is simply to advise the public that a proceeding concerning a particular facility has been commenced and [to describe] its general nature, not to set forth a pleading delineating the issues." *Id.* at 26. Thus, the argument goes, the notice of hearing in no way limits the scope of environmental inquiry in this spent fuel pool proceeding.

Intervenors also reject the argument that the Licensing Board improperly interfered in the staff's performance of its duties by requesting an EIS, instead of the allegedly forthcoming EIA. They contend that the Board acted wholly in accord with its authority and obligation to avoid delay by ordering the staff to do now what it believed NEPA would inevitably require as a matter of law. *Id.* at 30-31. Finally, Ms. Christa-Maria, *et al.,* argue that the Licensing Board, if necessary, should have the initial opportunity to decide the applicability of Section 102(2)(E). In the

produces with relation to the Big Rock facility." The staff proposes to rely on the *Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel,* NUREG-0575 (August 1979), for this purpose. *Id.* at 31.

[9]Intervenors emphasize that CPC itself stated in its application that the expansion of the spent fuel pool was "to allow continued plant operation." Br. 1, 20.

alternative, however, they request an extra two weeks in which to brief the issue.

The position of intervenor O'Neill parallels, in large measure, that of Ms. Christa-Maria, *et al.* He also presses several points not rasied by the other intervenors. For instance, Mr. O'Neill suggests (Br. 5-9) that the spent fuel pool expansion itself — apart from the continued plant operation it might permit — is a major federal action with a significant impact on the environment, thus requiring an EIS. In this regard, he discusses certain technical aspects of the pool and contends that this pool expansion proposal is "part of a major new federal policy on spent fuel reprocessing." *Id.* at 8. Mr. O'Neill expresses his confidence that an EIS "will find significant issues of fact that weigh heavily in favor of an alternative to pool expansion, including the alternative of doing nothing." *Id.* at 22. Finally, relying on his arguments to the Licensing Board, Mr. O'Neill asserts that Section 102(2)(E) clearly requires consideration of alternatives in this case. *Id.* at 28.[10]

In its *amicus* brief, CEQ expresses views generally consistent with those of the intervenors. It argues that because a new federal action is involved, the Licensing Board's decision to require an EIS on the spent fuel pool expansion and continued plant operation does not result in a retroactive application of NEPA, citing *MPIRG, supra,* as support. CEQ states that the continued plant operation permitted by the pool expansion makes the proposal a "major Federal action ..." and finds the absence of a prior EIS on the plant a significant factor further militating in favor of a full environmental review of plant operation now.[11] It also addresses the applicability of Section 102(2)(E), contending that certain court decisions,

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[10]Mr. Leithauser, who as a consequence of the denial of his petition to intervene is not a party to this proceeding (see note 4, *supra*), also filed a brief, although he failed to move for leave to do so. See 10 C.F.R. 2.715(d). No party, however, has objected to his continued participation. We therefore accept his brief and accord him the status of non-party participant (essentially "*amicus curiae*") for the purpose of this appeal. (Several procedural orders of this Board referred to Mr. Leithauser as an "intervenor." Those orders were not intended as *sua sponte* reversals of the Licensing Board's denial of his petition to intervene. Rather, our references simply reflected Mr. Leithauser's *own* characterization of his status in this case in the pleadings he filed that were the subject of the procedural orders. See, *e.g.*, Leithauser Motion to Postpone Hearing, filed December 19, 1980; Br. 1, 7.)

Mr. Leithauser argues generally that NEPA is broad enough in its reach to require the preparation of an EIS on this spent fuel pool expansion. In somewhat of a departure from the position of both the Licensing Board and the intervenors, however, he seems to argue that Big Rock is an "ongoing Federal project," governed by the line cf cases holding that NEPA applies to any continued federal involvement in such projects, even if the latter were initiated well before the enactment of NEPA. Lastly, like Ms. Christa-Maria, *et al,* Mr. Leithauser requests additional time to brief the Section 102(2)(E) issue.

[11]In another part of its brief, however, CEQ states that the amendment involved here requires "*at a minimum, an environmental assessment* to determine whether the proposed action significantly affects the environment, and furthermore, ... the likely outcome of an

as well as CEQ's own regulations, require agencies to consider alternatives to a proposed action even when that action does not otherwise warrant an EIS.[12] Applying Section 102(2)(E) to the instant case, CEQ concludes that there are indeed "unresolved conflicts concerning alternative uses of available resources," irrespective of whether the word "resources" is limited to "natural" ones (*e.g.*, land, air, water) or is given a broader construction.

II.

Soon after we held oral argument in this case, *amicus* CEQ submitted a letter ("the January 19 letter") with several attachments, purporting to relate to matters raised at the argument. This material can be categorized as follows:

1. Discussion in the January 19 letter of CEQ's special *amicus* role, its "mandate 'to review and appraise various programs and activities of the Federal Government in the light of the policy set forth in' " NEPA, and the "binding" nature of the Council's regulations on other agencies (p. 1, paragraph 2; p. 2, paragraph 1; and the attached report of the Environmental Law Institute. *NRC's Environmental Analysis of Nuclear Accidents Is It Adequate?* (1980));

2. Discussion in the January 19 letter concerning three letters from CEQ to the NRC and others that express the Council's views on the adequacy of the Commission's NEPA inquiries in other cases (p. 2, paragraph 2), with the attached three letters, dated March 20, August 12, and August 14, 1980;

3. Citations for and brief descriptions of four cases to which CEQ's counsel referred during oral argument; and

4. Corrections to the oral argument transcript.

Applicant CPC has moved to strike essentially the matter described in items 1 and 2 above.[13] It argues that in each instance this matter constitutes "supplemental argument" and is thus "impermissible after an appeal has

environmental assessment for this action would be a decision to prepare an EIS." Br. 4 (emphasis added; footnote omitted).

[12]In this case, consideration of alternatives "would of necessity include the alternative of 'no action.' " *Id.* at 10.

[13]CPC does not mention the Environmental Law Institute report in its motion. But since CEQ's reference to the report falls within that portion of the January 19 letter that CPC moves to strike, we assume CPC objects to the report as well.

been submitted for decision absent an opportunity for all parties to respond thereto."[14] The NRC staff supports the motion.[15]

We agree with CPC that those portions of the January 19 letter and enclosures to which CPC objects are improper supplemental argument. We therefore strike them from the record. See *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-115, 6 AEC 257 (1973).

Insofar as the references to CEQ's "special" relationship to other agencies are concerned, we recognize that the issue of the "binding" nature of the Council's regulations arose briefly during oral argument. Tr. 65, 88, 96. However, we requested no further briefing of the matter, and, as shown below, the resolution of this issue is of no relevance to our ultimate decision. It is clearly supplemental argument and is of no particular value in the disposition of this case.

The material submitted by CEQ dealing with its views on the adequacy of the NRC's environmental reviews in other cases is similarly improper supplemental argument and is, in any event, not relevant to this case. CEQ itself injected this matter into the oral argument. Tr. 66-67. While we do not see its relevance, CEQ had the opportunity to pursue this line of argument or policy position in its brief, but it failed to do so.[16] Fairness to the other parties precludes permitting CEQ to pursue it now.

CPC's motion to strike is *granted.*

III.

A.

The situation presented by this case is unusual, if not unique. Big Rock Point nuclear plant, a private project, has been fully constructed and operative since 1962 and licensed by the federal government for a full term since 1964 — years before the effective date of NEPA.[17] The plant owner has now determined that a modification of the spent fuel pool is necessary and desirable for continued plant operation beyond 1984. Although the

[14]In lieu of our striking the matter, CPC reluctantly suggests that all parties be permitted to reply to the tendered matter.

[15]We recently received a letter from CEQ urging us to deny applicant's motion. Because the letter was an untimely response in opposition to the motion (22 days late) and was not in proper pleading form, we must reject it for filing. See 10 C.F.R. 2.708, 2.709, 2.710, 2.730(c).

[16]For example, the three letters of March 20, August 12, and August 14, 1980, were in existence many months before CEQ filed its brief with us.

[17]See note 7, *supra.*

change purportedly would not affect the basic project (reactor operation) or the term of the license,[18] it nonetheless requires a license amendment and thus federal approval.

The need for government approval invokes the agency's obligations under NEPA. We perceive no real dispute among the parties that the Commission must make a threshold determination whether a "major Federal action significantly affecting the quality of the human environment" is involved here. The disagreement, of course, centers on the outcome of that determination, for that in turn may trigger other obligations and consequences for the Commission and the parties.

Because of the unusual nature of this case, it does not fit neatly into the categories or analytical framework suggested by most of the cases cited to us or discovered in our own research. Those cases generally involved the issue of whether certain post-NEPA federal action on a project initiated before NEPA, but not yet completed, constituted "further major action" so as to require an EIS. See, *e.g., Port of Astoria* v. *Hodel,* 595 F.2d 467 (9th Cir. 1979); *Sierra Club* v. *Hodel,* 544 F.2d 1036 (9th Cir. 1976); *MPIRG, supra; Jicarilla Apache Tribe of Indians* v. *Morton,* 471 F.2d 1275 (9th Cir. 1973); *Jones* v. *Lynn,* 477 F.2d 885 (1st Cir. 1973). The government's suggestion in many of these cases was that projects underway or even simply planned before the enactment of NEPA were totally immune to the future prescriptions of that statute. The courts, however, rejected that notion, at least insofar as any changes or additions to the projects were concerned. If a further major federal action significantly affecting the quality of the human environment is involved — irrespective of the date of commencement of the basic project — then NEPA perforce requires the preparation of an EIS for that *further major* action.

Given the nature of the matter before us,[19] we find these cases inapposite. The federal action sought here is approval of a license amendment to expand the capacity of the Big Rock Point spent fuel pool by the addition of extra racks for the fuel assemblies; it is *not* approval to alter any other aspect of the facility or the term of the license. Moreover, the situation in *Big Rock* is unlike that in many of the cases cited above. For example, Big Rock is not a government-sponsored housing program that evolves over a period of years. See *Jones* v. *Lynn, supra.* Nor is it a federal power project that similarly and typically undergoes many metamorphoses over a decade or longer. See *Port of Astoria, supra.* Renewal of old contracts and negotiation of new ones for activities on federally-administered land

[18]Big Rock's operating license is to expire in the year 2000.

[19]The first step in any NEPA inquiry is to define the "Federal action" requested or involved. *Aberdeen and Rockfish R.R.* v. *SCRAP,* 422 U.S. 289, 322 (1975).

are not involved. See *MPIRG, supra.* Further, we agree with the Licensing Board that this is not "an ongoing Federal project which requires constant reevaluation to determine whether it should continue." 12 NRC at 359.

Thus, the object of this proceeding does not readily lend itself to characterization as a "further major Federal action." No spent fuel pool expansion at any other facility has been found to be a "major Federal action," and no party to this proceeding save Mr. O'Neill, contends that the expansion *per se* is such an action.[20] Rather, the intervenors and CEQ contend that the continued plant operation intended to result from the pool expansion must be taken into account, and that it is this continued operation that makes the pool expansion a major action with significant environmental effects.

The parties offer no real support for their view that continued plant operation must be considered, other than the argument that it is the necessary and intended result of the license amendment and is therefore within the scope of the proposal. As such, the environmental effects or impacts associated with continued plant operation are best described as "secondary" or "indirect" effects of the proposed federal action, in contrast with the "primary" effects directly associated with the spent fuel pool expansion itself (the additional racks, increased concentration of spent fuel, etc.).[21]

The critical question then is whether NEPA requires consideration of the secondary, indirect impacts associated with continued plant operation.[22]

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[20]As discussed *infra,* we are unable to make any finding at this juncture as to whether this particular spent fuel pool expansion itself is or is not a major federal action requiring an EIS. That is a matter initially for the staff's determination and subsequently for exploration during the hearing process.

[21]The Commission's environmental regulations, 10 C.F.R. 51, do not categorize an action's impacts in this manner. As discussed *infra,* however, a number of courts have employed this terminology and analysis. CEQ's regulations provide useful guidance in this area as well. "Direct effects" are defined as those "which are caused by the action and occur at the same time and place." 40 C.F.R. 1508.8(a). "Indirect effects" are those "which are caused by the action and are later in time or farther removed in distance, but are still reasonably foreseeable. * * *" 40 C.F.R. 1508.8(b).

[22]Again, the Commission's environmental regulations do not address this matter. We therefore reject the staff's lead argument that 10 C.F.R. 51, in either its present or its proposed form, somehow provides the solution to this problem.

We also reject applicant's argument that the notice of hearing forecloses consideration of anything other than the spent fuel pool itself. As we recently pointed out in *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-616, 12 NRC 419, 426 (1980), the hearing may encompass "issues *fairly raised* by the application to modify the spent fuel pool ...." (Emphasis added.) Continued plant operation, the intended result of an expanded spent fuel pool, is clearly an issue "fairly raised" by the application to modify the pool. Although it may not necessarily be within the ambit of the environmental analysis required by NEPA in this case (see discussion *infra*), we cannot say continued plant operation is beyond the Licensing Board's "jurisdiction." Compare *Public Service Co. of Indiana, Inc.* (Marble Hill Station, Units 1 and 2), ALAB-316, 3 NRC 167 (1976) (antitrust issues beyond scope of hearing instituted to

Many courts have concluded that "NEPA is concerned with indirect effects as well as direct effects," *MPIRG, supra* at 1322 — providing one does not stray "beyond reasonable forecasting" into "the realm of pure speculation." *North Dakota* v. *Andrus,* 483 F.Supp. 255, 260 (D. N.Dak. 1980).[23] CEQ's regulations also suggest that, once an EIS is to be prepared, it should include discussion of both direct and indirect effects. 40 C.F.R. 1502.16(a) and (b). (The Council's regulations do not explicitly address, however, whether indirect effects should be considered when determining *if* an action is "major.")

In this case, assuming that no alternative storage for spent fuel is found — such as a government-operated away-from-reactor (AFR) facility — and that Big Rock is not shut down for a substantial period of time for other reasons, expansion of the spent fuel pool is necessary to permit the plant to continue operating beyond 1984. Oral argument, Tr. 9. Although we do not believe that a *denial* of this license amendment would necessarily make shutdown a certainty, we also cannot reasonably characterize continued plant operation as a remote or speculative indirect consequence of a *grant* of the amendment. Thus, one might quickly conclude that, in this case, NEPA indeed requires consideration of the secondary impacts associated with continued plant operation.

The "rule of reason," which guides our interpretation and application of NEPA, however, precludes us from reaching so hasty — or simple — a conclusion. See *Natural Resources Defense Council, Inc.* v. *Morton,* 458 F.2d 827, 834, 837 (D.C. Cir. 1972). We would be remiss in our responsibilities were we to fail to scrutinize carefully the realities of this matter. CEQ has urged us not to "isolat[e] the action of the agency from the impacts." Oral argument, Tr. 64. We believe this is sound counsel, and, for that reason, we next consider the real impacts of the Big Rock spent fuel pool expansion.

We *assume* that, as in the case of other spent fuel pool expansions, the applicant will undertake no modifications that will affect reactor operation or any other aspect of the facility.[24] Thus, after the addition of more racks

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consider health, safety, and environmental effects of plant construction); *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-534, 9 NRC 287 (1979) (general safety issues and need for power beyond scope of special proceeding convened to consider interim operation of control building); *Zion, supra* (modification of overall plant emergency plan beyond scope of application to expand spent fuel pool).

[23]But see, *e.g., National Ass'n of Government Employees* v. *Rumsfeld,* 418 F.Supp. 1302, 1305-1306 (E.D. Pa. 1976), and cases cited therein, holding that NEPA is not concerned with "social or economic" impacts in the absence of a primary significant environmental impact.

[24]We make this assumption in the absence of a thorough search of the record (in particular, the application) for support for this "fact." We leave that function to the Licensing Board. We note, however, that applicant asserts this on brief (at 13), and no other party quarrels with the notion that increasing the capacity of a spent fuel pool does not effect any changes in reactor operation.

for the fuel assemblies, Big Rock Point will continue to operate as it has since 1962. To be sure, such operation will have the usual environmental impacts, but they will be the same ones that have been present since the first day of operation. Continued plant operation simply results in maintenance of the environmental status quo. Insofar as this secondary or indirect effect is concerned, there are no environmental *changes* to evaluate.[25]

We believe that in these circumstances, a reasonable application of NEPA does not require consideration of the continued plant operation permitted by the pool expansion. Indeed, the whole purpose in considering primary or secondary impacts of an action is to determine if they have a cause-and-effect relationship with any environmental *changes.* [26] Where, as here, there is no change in the environmental status quo, that purpose need not be served.

Several court of appeals decisions support this analysis. In *Committee for Auto Responsibility* v. *Solomon,* 603 F.2d 992 (D.C. Cir. 1979), *cert. denied,* 445 U.S. 915 (1980), the court concluded that a General Services Administration decision to lease a parking lot to a parking management firm did not require an EIS. The court noted GSA's finding that the level of pollutants would not be altered from its existing level as a result of the new lease. Since GSA's proposal would not alter the environmental status quo, the court held that no EIS was required. As the District of Columbia Circuit stated, "[t]he duty to prepare an EIS normally is triggered when there is a proposal to change the status quo." *Id.* at 1002-1003. Significantly for purposes of the instant case, an EIS apparently had never been prepared for the particular federal facility involved in *Solomon. Id.* at 1002 n.43.

The Ninth Circuit also supports the view that NEPA does not require an EIS when an action does not directly or indirectly bring about any change in the environmental status quo. In *Westside Property Owners* v. *Schlesinger,* 597 F.2d 1214 (9th Cir. 1979), one of the issues was whether the formalization of a German-American pilot training program (which began in 1964) through a 1971 diplomatic agreement constituted a "major Federal action" requiring an EIS. The court concluded that it did not. *Id.* at 1225. Its decision was influenced by, *inter alia,* the following facts: (1) the United States approved the design of the training program long before NEPA; (2) "the 1971 agreement did not affect the pollution produced by

[25]Compare *Virginians for Dulles* v. *Volpe,* 541 F.2d 442, 445 (4th Cir. 1976), where the court found "the FAA's acquiescence in the *vastly expanded use* of the airports require[d] an impact statement" (emphasis added).

[26]CEQ's definition of "indirect effects" reflects a similar concern with measuring changes. 40 C.F.R. 1508.8(b) states (emphasis added): "Indirect effects may include *growth inducing effects* and other effects related to *induced changes* in the pattern of land use, population density or *growth rate,* and related effects on air and water and other natural systems, including ecosystems."

the training"; and (3) a substantial amount of training had already occurred and was "in the same manner as the future training of German pilots." *Id.* at 1224. Strikingly similar factors exist in this case as well: (1) Big Rock received its full-term license from the Atomic Energy Commission in 1964; (2) the proposed spent fuel pool expansion, while permitting the plant to continue operating beyond 1984, presumably will not result in any operational changes and thus will not affect the existing level of the environmental impacts attributable to reactor operation;[27] and (3) Big Rock has been operating for over 18 years and, if the amendment is approved, will continue to do so for the remaining term of its license (subject, of course, to other unrelated circumstances that may develop). See also *Burbank Anti-Noise Group* v. *Goldschmidt*, 623 F.2d 115, 116-117 (9th Cir. 1980), *cert. denied*, 49 U.S.L.W. 3636 (U.S. Mar. 2, 1981); *San Francisco Tomorrow* v. *Romney*, 472 F.2d 1021, 1025 (9th Cir. 1973).

Greene County Planning Board v. *Federal Power Commission*, 455 F.2d 412 (2d Cir.), *cert. denied*, 409 U.S. 849 (1972), is similarly instructive, particularly because it concerns a hydroelectric power plant. The basic project (*i.e.*, the powerhouse and reservoirs) was planned, licensed, and under construction for six months before the effective date of NEPA, but the FPC specifically withheld approval of the transmission lines associated with the plant, pending further consideration of their effect on the environment. Approval of the lines came after NEPA, and the court held that the agency was bound to comply with the statute in that regard. Insofar as the basic project (which was 80 percent complete) was concerned, however, the court found "no basis for applying NEPA retroactively." *Id.* at 424.

Although the *Greene County* court's approach differs somewhat from ours here, the case nonetheless provides a useful precedent and analogy. There, an environmental analysis of the impacts of a power plant, not yet completed but licensed just six months before NEPA, was *not* required in connection with the post-NEPA approval, soon thereafter, of related transmission lines. It follows, therefore, that in connection with a proposal to expand a plant's spent fuel pool, NEPA does not require consideration of the environmental impacts of continued reactor operation where the plant was completed and licensed years before promulgation of that statute and has since been in operation for almost two decades.

Our conclusion is further fortified by the very purpose of a NEPA inquiry — to identify aspects of a project that can still be changed to mitigate possibly detrimental environmental effects. See *Virginians for*

[27]See note 24, *supra*.

Dulles, supra at 446. For example, in *Arlington Coalition on Transportation* v. *Volpe,* 458 F.2d 1323, 1332 (4th Cir. 1972), approval for the federal highway involved had "not been given, construction contracts [had] not been awarded, and actual construction on the highway itself [had] not begun" at the time of the NEPA challenge. Since the project was far from complete, modifications to mitigate environmental effects were easily possible, and the court therefore required an EIS for any further action.[28] In this case, however, the reactor at Big Rock has been fully completed and operative since 1962, and the necessary "Federal action" (*i.e.,* approval of the license amendment to expand the spent fuel pool) purportedly would not provide any opportunity to alter *plant operation.*

NEPA "is not an authorization to undo what has already been done." *Jones* v. *Lynn, supra* at 890. And just as we concluded in *Trojan, supra* at 266 n.6, and *Prairie Island, supra* at 46 n.4, that NEPA does not require duplicative environmental analyses, so too must we conclude that "to formulate an EIS [on continued plant operation] under these circumstances would trivialize NEPA's EIS requirement and diminish its utility in providing useful environmental analysis for major federal actions that truly affect the environment." *Solomon, supra* at 1003.[29]

We believe our judgment here represents "a just and practicable balance"[30] between the spirit of NEPA and the realities of this case.[31]

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[28]The same is true of *Henry* v. *Federal Power Commission,* 513 F.2d 395 (D.C. Cir. 1975), upon which CEQ relies for its view that NEPA requires consideration now of the environmental impacts of the entire Big Rock project. Apart from the fact that the court's actual holding in the case was that petitioners raised the NEPA issue prematurely, it is not without significance that the gasification project involved in *Henry* was neither licensed nor constructed at the time. Thus, again, an EIS would serve a very useful purpose in identifying aspects of the total project still susceptible to modification on environmental grounds.

[29]Nothing in our holding is intended to suggest, however, that the Commission itself could not, as a matter of policy, require evaluation of the environmental impacts of the continued plant operation resulting from a spent fuel pool expansion. Neither NEPA nor the agency's environmental regulations, 10 C.F.R. 51, preclude such an exercise of discretion. Cf. *Offshore Power Systems* (Floating Nuclear Power Plants), CLI-79-9, 10 NRC 257, 261 (1979).

In this connection, Ms. Christa-Maria, *et al.,* contend that the Licensing Board had discretion to order the preparation of an EIS on continued plant operation. Br. 21-25. Because the Board did not purport to exercise discretion but rather held that *NEPA requires* an EIS, we do not reach the issues of whether such discretion was the Board's to exercise and, if so, whether it properly exercised it.

[30]*Jones* v. *Lynn, supra* at 887.

[31]We note that the Supreme Court's most recent NEPA cases evidence a trend toward construing that statute in a manner consistent with our approach here. Although these cases are not on point as to the issue before us, the Court's guidance is useful. For example, *Andrus* v. *Sierra Club, supra* at 349, 364-365, held that Section 102(2)(C) of NEPA does not require

Hence, we conclude that NEPA does not require consideration of the environmental impacts of the continued plant operation likely to result from expansion of the Big Rock spent fuel pool, assuming that expansion will not cause any changes in reactor operation. If this assumption proves to be accurate, the scope of the agency's environmental inquiry may be confined to the effects of the expanded pool itself. We therefore reverse the Licensing Board.[32]

B.

Our conclusion that NEPA does not require the agency to consider the environmental impacts of continued plant operation neither ends our inquiry nor provides a complete disposition of the rulings that the Licensing Board referred to us. The Board ordered the preparation of an EIS "covering the environmental impacts of an expanded spent fuel pool" as well as "the additional term of operation of the facility that such expansion would permit." 12 NRC at 366. Having determined that the Board erred in finding that NEPA requires an EIS on continued plant operation, we now must decide whether the Board also erred in finding that an EIS on the expansion of the pool itself is necessary.

There are two factors of significance in the Licensing Board's ruling on this point. First, its order to prepare an EIS on the pool expansion is closely

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federal agencies to prepare EISs to accompany appropriation requests because the latter are neither "proposals for legislation" nor "proposals for ... major Federal actions." In *Vermont Yankee Nuclear Power Corp.* v. *Natural Resources Defense Council, Inc.*, 435 U.S. 519, 551 (1978), the Court noted that "[t]o make an impact statement something more than an exercise in frivolous boilerplate the concept of alternatives [in Section 102(2)(C)] must be bounded by some notion of feasibility." And in *Kleppe* v. *Sierra Club*, 427 U.S. 390, 399 (1976), the Court held that NEPA does not require "regional" impact statements where the proposed federal action is not regional in scope. Such an EIS would be an exercise in the "impossible," "little more than a study ... containing estimates of potential development and attendant environmental consequences," and a document lacking in the "factual predicate for the production of an environmental impact statement of the type envisioned by NEPA." *Id.* at 402. These decisions share in common with one another and this case a construction of NEPA that does not require an endeavor destined to be of little utility.

[32] Many of the arguments made and cases cited to us in this proceeding concerned the matter of retroactivity — *i.e.*, whether the Licensing Board's decision constitutes an improper retroactive application of NEPA. In view of the approach we take in this opinion, we do not reach, and accordingly do not decide, that issue. As we see it, our inquiry logically led us to determine first whether the impacts associated with continued plant operation were even a required area of consideration under NEPA. We have determined that they are not. Had we decided otherwise, our next step would have been to decide whether such consideration would violate the proscription against applying NEPA retroactively.

tied to its order to prepare an EIS on continued plant operation.[33] Second, the Board had no "record" upon which to support its finding of significant impact because the staff has yet to prepare an EIA or any other environmental document.

In view of these factors, we believe that the Board should reconsider its order to the staff to prepare an EIS on the proposed spent fuel pool expansion. We therefore reverse this ruling as well.[34]

As to the first factor (the linkage between the EIS ordered for the spent fuel pool and that for the additional term of plant operation), our decision in the preceding section effectively eliminates continued plant operation from the scope of the environmental review NEPA requires in this case. The Board should therefore rethink its decision in light of our opinion by focusing on the need *vel non* for an EIS on the pool itself.

We also believe that the Board, in reconsidering its decision, should await the preparation of the staff's environmental analysis, whether that turns out to be an EIA or an EIS.[35] It is unwise, if not improper, to decide without the record support provided by the staff's environmental review, whether a given action significantly affects the environment.[36] See *Jones* v. *Lynn, supra* at 891.

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[33]Indeed, the Board's conclusion that approval of a license amendment to expand a spent fuel pool is a major federal action with a significant effect upon the environment — the finding necessary to trigger NEPA's EIS requirement — is grounded on its belief that (1) the amendment's "sole purpose" is to enable CPC "to utilize a greater term of the license than would otherwise be possible" (12 NRC at 363; see also *Id.* at 359, 360, 361), and (2) "making such operation possible for a period of ten years clearly constitutes a major Federal action" (*Id.* at 364 n.2).

[34]The Board will also necessarily have to reconsider its restatement and admission of Mr. O'Neill's Contention VIII.

[35]While this case was pending, the Commission approved the *Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel*, NUREG-0575 (August 1979), and indicated that this document is now applicable to proposed spent fuel storage licensing actions. 46 Fed. Reg. 14506 (February 27, 1981). The staff has already indicated its intent to rely to some extent on this document. Br. 30-31.

We note in this regard that NUREG-0575 itself states that "[b]ecause there are many variations in storage pool designs and limitations caused by spent fuel already in some pools, the licensing reviews must be done on a case-by-case basis." NUREG-0575, Vol. 1, 8-1. Moreover, in approving the document, the Commission noted that its "action does not affect any other requirements which may exist to address specific environmental and safety issues for individual licensing action." 46 Fed. Reg. 14507. Thus, presumably the staff's environmental review will take account of any factors that distinguish Big Rock from other plants — *e.g.*, its use of mixed oxide fuel. See *Consumers Power Co.* (Big Rock Point Nuclear Plant), CLI-74-33, 8 AEC 221 (1974). The extent to which NUREG-0575, with its generalized approach to spent fuel storage, is relevant to Big Rock, therefore, remains to be determined.

[36]In this vein, it does not seem logical to suggest, as the Licensing Board's opinion does, that an action that otherwise may not have a significant effect on the environment is transformed into one that does have such effect simply by the absence of an environmental review of a different, prior action.

We appreciate the Licensing Board's desire and effort to avoid unnecessary delay in this proceeding.[37] The Board believed that an EIA was forthcoming and that, as a matter of law, it would be inadequate, regardless of its content. It therefore issued its ruling before ever seeing the document and in an obvious attempt to expedite the case. However worthly such an effort may be, this action must be balanced against the integrity of the hearing process. And, in our view, the latter outweighs the former.

First, we find the assumption that precipitated the Board's action — that the staff would issue its "usual" EIA on spent fuel pools — to be an inappropriate prejudgment of the staff's position on an important issue.[38] The Board, staff counsel, applicant, and all other parties should not encourage such prognostication, as it could have a chilling effect on the staff's ultimate recommendation. The staff should be permitted to do its job in an honest and objective fashion, without being inhibited by perhaps the self-serving predictions of one party or another.

Second, if the Licensing Board had permitted the staff to complete and submit its environmental analysis (whether an EIA or EIS), the Board would have had the benefit of a "record" to aid it in reaching its conclusions.[39] Moreover, the parties (and the Board) would have had the chance to defend or challenge the content and conclusions of the document during the course of the hearing, which provides the vehicle for contesting any perceived deficiencies in the staff's analysis. See 10 C.F.R. 2.718(g), 2.721(d), 51.52(d). Prematurely concluding that such a document is inadequate before it is even produced deprives the participants of their opportunity to explore the matter during the hearing, as the Commission's regulations contemplate.

We emphasize that our comments are not intended to reflect a judgment on our part as to whether this proposed spent fuel pool expansion is or is not a major action with a significant effect on the environment. Rather, our purpose is to underscore the importance of not bypassing the staff's function and of permitting the hearing to take its natural course.

C.

The final issue posed in this case is whether Section 102(2)(E) of NEPA requires the consideration of alternatives to this spent fuel pool expan-

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[37]Indeed, we share this concern and trust that when the Licensing Board resumes this proceeding, the staff will endeavor to complete its environmental review quickly.

[38]The assumption apparently originated with applicant's counsel at the December 5, 1979, prehearing conference (Tr. 217), was later promoted by the Board (11 NRC at 133), and was eventually acquiesced in by the staff and other parties.

[39]To illustrate, if the staff prepares an EIA and the Board agrees with its "negative declaration," the required explanatory text in the appraisal can, if adequate, provide the record support for the Board's conclusion. See 10 C.F.R. 51.7(b).

sion.[40] The Licensing Board concluded that it was unnecessary to reach this issue because of its decision that the "more comprehensive" Section 102(2)(C) required an EIS here. 12 NRC at 359. We, on the other hand, conclude that it would be premature to decide this issue now, in the absence of a record upon which to base such a finding.

As is evident from our decision in *Virginia Electric and Power Co.* (North Anna Station, Units 1 and 2), ALAB-584, 11 NRC 451, 456-459 (1980), *petition for review pending sub nom. Potomac Alliance* v. *Nuclear Regulatory Commission* (No. 80-1862, D.C. Cir., filed July 28, 1980), some factual basis (usually in the form of the staff's environmental analysis) is necessary to determine whether a proposal "involves unresolved conflicts concerning alternative uses of available resources" — the statutory standard of Section 102(2)(E). See, *e.g., Id.* at 458 n.14.[41] Since this proceeding is in its incipient stages, there is little in the record that could provide the foundation for the conclusion that this particular spent fuel pool expansion proposal does or does not involve such "unresolved conflicts." Thus, until the record is more fully developed with the inclusion of such documents as the staff's environmental evaluation, a meaningful determination of the relevance of Section 102(2)(E) to this proceeding cannot be made.

Upon return of this case, however, the Licensing Board may once again be able to avoid entirely resolution of this issue. If the staff concludes that this pool expansion license amendment requires an EIS, then the mandated consideration of alternatives therein should suffice insofar as Section 102(2)(E) is concerned. Moreover, the staff has indicated that it "intends to consider alternatives in whatever environmental document it produces with relation to the Big Rock facility." Br. 30. Thus, if the staff prepares an EIA, the parties can apparently expect discussion of the "alternative uses of available resources."

[40] Although the Licensing Board did not reach this issue, we nevertheless asked the parties to address it in their briefs on appeal. While most did, Mr. Leithauser and intervenors Christa-Maria, *et al.*, have requested additional time to brief the matter. In view of the disposition of the Section 102(2)(E) issue we make here, it is unnecessary to rule on their requests.

[41] As we also observed in *North Anna, supra* at 457, Section 102(2)(E) of NEPA is not limited to major federal actions with significant effects on the environment and may require consideration of alternatives even when an EIS is not otherwise required. See *Trinity Episcopal School Corp.* v. *Romney*, 523 F.2d 88, 93 (2d Cir. 1975), *on remand, Trinity Episcopal School Corp.* v. *Harris*, 445 F. Supp. 204 (S.D.N.Y. 1978), *rev'd and remanded sub. nom. Karlen* v. *Harris*, 590 F.2d 39 (2d Cir. 1978), *rev'd sub. nom. Strycker's Bay Neighborhood Council, Inc.* v. *Karlen*, 444 U.S. 223 (1980); *California* v. *Bergland*, 483 F. Supp. 465, 488 (E.D. Cal. 1980).

IV.

For the foregoing reasons, we find that a reasonable application of NEPA does not require the preparation of an EIS on the continued plant operation likely to result from the proposed expansion of the Big Rock spent fuel pool, assuming that the expansion will not effect any change in reactor operation. We therefore reverse the Licensing Board's contrary finding. We also direct the Licensing Board to reconsider its companion order to the staff to prepare an EIS on the spent fuel pool itself. Before doing so, however, the Board should await the submission of the staff's environmental evaluation. Similarly, the Board should await the filing of that document before determining the applicability of Section 102(2)(E) to this case.

Applicant's motion to strike certain material submitted by CEQ following oral argument is *granted.*

The rulings referred to us in the Licensing Board's September 12, 1980, "Memorandum and Order on NEPA Review" are *reversed.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

Cite as 13 NRC 335 (1981) LBP-81-8

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Charles Bechhoefer, Chairman
Dr. Oscar H. Paris
Glenn O. Bright

In the Matter of | **Docket Nos. 50-387 OL**
50-388 OL

PENNSYLVANIA POWER & LIGHT COMPANY and ALLEGHENY ELECTRIC COOPERATIVE, INC.
(Susquehanna Steam Electric Station, Units 1 and 2)

March 16, 1981

The Licensing Board rules on two of the applicants' motions for summary disposition of various contentions of an intervenor, granting one motion and partially granting the other motion.

RULES OF PRACTICE: SUMMARY DISPOSITION

Where a party moves for dismissal of contentions by summary disposition, the Commission's summary disposition rule directs the presiding officer to render the decision sought if the filings in the proceeding, depositions, answers to interrogatories, and admissions on file, together with the statements of the parties and the affidavits, if any, show that there is no genuine issue as to any material fact and that the moving party is entitled to a decision as a matter of law. 10 CFR 2.749(d).

RULES OF PRACTICE: SUMMARY DISPOSITION

All material facts set out in the statement of material facts which accompanies a summary disposition motion are deemed to be admitted unless controverted by the opposing party. 10 CFR 2.749(a).

RULES OF PRACTICE: SUMMARY DISPOSITION

Where motions for summary disposition are supported by affidavit, a party opposing the motion may not rest upon the mere allegations or denials of his answer; his answer by affidavits or as otherwise provided must set forth specific facts showing that there is a genuine issue of fact. 10 CFR 2.749(b).

RULES OF PRACTICE: SUMMARY DISPOSITION

When a response to a summary disposition motion has been provided, the record and affidavits both supporting and opposing the motion must be viewed in the light most favorable to the opposing party. Moreover, the party seeking summary disposition has the burden of showing the absence of a genuine issue of material fact; if it fails to do so, summary disposition will not be granted irrespective of the quality of any response.

RULES OF PRACTICE: SUMMARY DISPOSITION

Prior to granting summary disposition, the presiding officer must be convinced that there are no significant outstanding unresolved questions material to the particular issue under review.

MEMORANDUM AND ORDER

(Ruling on Motions for Summary Disposition of Contentions 2 and 16)

The Applicants in this operating license proceeding have filed motions for summary disposition of all or parts of four contentions: numbers 2, 12, 16, and 17.[1] In this opinion, we are considering the motions relating to Contentions 2 and 16.[2] For reasons hereinafter set forth, we are granting in

[1]The contentions are numbered as set forth in the Licensing Board's Special Prehearing Conference Order, LBP-79-6, 9 NRC 291 (March 6, 1979).

[2]Responses with respect to the Contention 12 motion are not yet due to be filed.

part and denying in part the motion with respect to Contention 2, and granting the motion with respect to Contention 16.

A. General

Summary disposition motions are authorized by 10 CFR § 2.749. Under that authority, we are directed to render the decision sought—here the dismissal in whole or in part of various contentions—"if the filings in the proceeding, depositions, answers to interrogatories, and admissions on file, together with the statements of the parties and the affidavits, if any, show that there is no genuine issue as to any material fact and that the moving party is entitled to a decision as a matter of law." 10 CFR § 2.749(d). This provision is analogous to and has been interpreted in accord with Rule 56 of the Federal Rules of Civil Procedure. *Alabama Power Co.* (Joseph M. Farley Nuclear Plant, Units 1 and 2), ALAB-182, 7 AEC 210, 217 (1974). Both the Commission and the Appeal Board have long encouraged the use of summary disposition procedures to resolve issues where the proponent of the issue has failed to establish that a genuine issue exists. *Northern States Power Co.* (Prairie Island Nuclear Generating Station, Units 1 and 2), CLI-73-12, 6 AEC 241, 242 (1973), *aff'd sub nom BPI* v. *AEC*, 502 F.2d 424 (D.C. Cir. 1974); *Mississippi Power & Light Co.* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 AEC 423, 424-25 (1973); *Dusquesne Light Co.* (Beaver Valley 1), ALAB-109, 6 AEC 243, 246 (1973).

All material facts set out in the statement of material facts which accompanies a summary disposition motion are deemed to be admitted unless controverted by the opposing party. 10 CFR § 2.749(a). Where, as here, motions for summary disposition are supported by affidavit, a party opposing the motion may not rest upon the mere allegations or denials of his answer; his answer by affidavits or as otherwise provided must set forth "specific facts showing that there is a genuine issue of fact." 10 CFR § 2.749(b).

When a response to a summary disposition motion has been provided, we must view the record and affidavits both supporting and opposing the motion in the light most favorable to the opposing party. *See Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), LBP-74-36, 7 AEC 877, 879 (1974). Moreover, the party seeking summary disposition has the burden of showing the absence of a genuine issue of material fact; if it fails to do so, summary disposition will not be granted irrespective of the quality of any response. *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 752-54 (1977). In short, prior to granting summary disposition, we must be convinced that there are no significant outstanding unresolved questions material to the particular issue under review.

We have applied the foregoing standards in ruling upon the motions before us.

B. Contention 2 (Chlorine)

1. The Applicants filed their motion for summary disposition of a portion of Contention 2 on November 6, 1980. On November 24, 1980, Citizens Against Nuclear Dangers (CAND), the sponsor of the portion of Contention 2 to which the motion relates, filed a document entitled "Petition and Motions on Summary Disposition" which, in part, addressed the Applicants' Contention 2 motion. On December 2, 1980, the NRC Staff filed an answer in support of the Applicants' motion. CAND filed a somewhat belated response to the Staff's answer on January 7, 1981.[3] No other party has filed any response to the Contention 2 motion.

Contention 2, as accepted by the Board in LBP-79-6, *supra*, 9 NRC at 301, reads as follows:

> 2. The residual risk of low-level radiation which will result from the release from the facility of radionuclides, and particularly from the release of cesium-137 and cobalt-60, into the Susquehanna River, and the health effects of chlorine discharged into the river, have not been, but must be, adequately assessed and factored into the NEPA cost-benefit balance before the plant is allowed to go into operation.

Applicants' motion requests summary disposition, in their favor, of that portion of Contention 2 which alleges that the health effects of chlorine discharged into the river have not been adequately assessed.

2. Both the Applicants and Staff filed discovery requests against CAND to obtain more specific information about CAND's concerns. In response to the Staff's request for specific information about the amount of chlorine to be released and the health effects which would result,[4] CAND indicated that the adverse health effects from the discharge of chlorine from the plant would be greater than estimated because the Applicants will be compelled to use more chlorine than specified in the application. CAND asserts that more chlorine than anticipated will have to be used because of river pollution resulting from: (1) "continual pumping of billions of

[3]Our Order dated December 9, 1980 invited CAND and other parties to file such a response by January 5, 1981.

[4]NRC Staff's First Round Discovery Requests of the Citizens Against Nuclear Dangers (CAND), dated May 21, 1979, pp. 3-4.

gallons of mine acid drainage into the Susquehanna River from numerous existing abandoned mine workings* * *to make possible the new mining operations" planned in Anthracite coal deposits near the river, and (2) "the *Butler Mine Water Tunnel* waste chemical spills into the Susquehanna River" involving "hundreds of thousands of barrels of highly toxic chemical wastes (possibly including radioactive wastes) [which] were covertly dumped down boreholes into abandoned coal mine voids near Pittstone, Pennsylvania* * *."[5]

Applicants' motion is grounded on the claim that there is no genuine issue of material fact to be heard with respect to the chlorine issue as clarified by CAND in discovery.[6] Through the affidavit of Mr. James Rios, the Supervising Engineering Specialist for the San Francisco Power Division of Bechtel Power Corp. (Rios affidavit), the Applicants assert that the purpose of chlorinating the water systems in the Susquehanna plant is to control the growth of slime-forming organisms on equipment surfaces and to disinfect the potable water supply and sewage effluent. Further, they say that the presence of mine acid drainage and spills of toxic chemical wastes will not result in any significant increase in the rate of growth of biofouling organisms on equipment surfaces, nor will the mine drainage and chemical spills change the amount of chlorine required to disinfect the potable water supply and sewage effluent.[7]

In its December 2, 1980 answer, the NRC Staff independently evaluated the chlorine issue as raised by CAND's and the Applicants' documents and concluded that the chlorine portion of Contention 2, as refined by discovery, lacks a factual basis.[8] This conclusion was supported by the affidavit of John C. Lehr, a Senior Environmental Engineer in the Environmental Engineering Branch, Division of Engineering, of NRC's Office of Nuclear Reactor Regulation (Lehr affidavit). The Staff concurred with the Applicants' statement of material facts and went on to assert that acid mine drainage would tend to decrease the extent of biofouling through direct and indirect toxic effects on the biota. As a consequence, the amount of chlorine needed to defoul the plant's water systems could decrease if the conditions alleged by CAND came into existence.[9] With regard to toxic chemical spills, the Staff said it could not make a judgment as to the need to alter the chlorination level proposed by the Applicants, but it noted that

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[5]Citizens Against Nuclear Dangers Motions and Replies to Interrogatories Concerning Contentions Nos. 2, 16 & 17, dated April 29, 1980, pp. 4-5.

[6]Applicants' Motion, p. 1.

[7]Applicants' Statement of Material Facts as to Which There Is No Genuine Issue To Be Heard (Contention 2 - Chlorine), dated November 6, 1980, pp. 1-2; Rios affidavit, dated November 4, 1980, p. 2.

[8]NRC Staff Answer, at pp. 2, 6.

[9]Lehr affidavit, p. 3.

chlorination is not generally used to treat water polluted by toxic chemicals.[10]

The Staff also addressed the broader question of whether the health effects of the chlorine to be discharged at the levels indicated in the application have been adequately assessed, even though CAND's response to the Staff's interrogatories did not indicate that such was the thrust of Contention 2. (The wording of Contention 2 clearly encompasses such health effects.) The Staff attested that active chlorine chemical species will be reduced to below detectable limits by a dechlorination system. This system will remove chloramines from the effluent, but some chlorides and trihalomethanes will be released by the plant. Chlorides are not likely to be discharged at levels that will threaten the public health.[11] However, the Staff appears to be less certain about trihalomethanes.

Trihalomethanes and halomethanes are suspected to be carcinogenic. An NRC sponsored study which examined the products of low-level chlorination of various natural waters in the U.S. showed that chloroform was the principal trihalomethane product; in freshwater it occurred in concentrations ranging from 2 $\mu g/l$ to 25 $\mu g/l$. Haloforms occurred in concentrations up to 55 $\mu g/l$.[12] The Staff compared these levels with standards set forth in EPA's Interim Primary Drinking Water Regulations, which provide that total trihalomethanes in community drinking water systems serving 75,000 or more persons not exceed 100 $\mu g/l$, and noted that the allowable limit is comparable to or well above the values reported for chlorinated cooling tower waters.[13] The Staff was unable, however, to estimate the likely levels of trihalomethanes to be produced by the Susquehanna plant. The Staff indicated that the Applicants have not made a quantitative estimate of trihalomethane concentrations in the plant discharge and pointed out that active chlorine behavior depends on the specific water chemistry existing under operating conditions, which cannot be predicted accurately.[14] Although the Staff concluded that the use of chlorine for biofouling control will not result in a signficant impact on the public health, it was able to state only that "[t]he trihalomethane content of the discharge *may be* below the maximum contaminant level [established] by EPA under the Safe Drinking Water Act" (emphasis added).[15]

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[10]*Ibid.*
[11]Lehr affidavit, p. 4. Also, *see* Draft Environmental Statement (NUREG-0564), June 1979, pp. 4-4 through 4-7.
[12]Lehr affidavit, p. 6.
[13]*Id.*, p. 7.
[14]*Id.*, p. 5.
[15]*Id.*, p. 9.

CAND has provided two responses with respect to this motion. To deal with the last response first, CAND's January 7, 1981 filing generally denied the Applicants' and Staff's conclusions and went on to allege that many of the Applicants' findings "are based on misleading extrapolation of data" and that Applicants "have cleverly compiled selective statistics to estimate or infer findings beyond the known range on the basis of certain variables within the known range, from which the estimated values are assumed to follow."[16] No details are provided. This response thus fails to present material or substantial facts to support this allegation or otherwise to controvert the facts advanced by the Applicants and/or the Staff.[17]

In its earlier November 24, 1980 filing, CAND mentioned an incipient CEQ study which allegedly will link chlorinated drinking water and cancer, a matter related in general to the health effects encompassed by Contention 2. Although the quantity of permissible chlorine released is controlled by the Environmental Protection Agency (EPA), not by NRC, the NRC is authorized to ascertain the health effects of chlorine releases and to include them in the cost-benefit balance for the facility. *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), ALAB-248, 8 AEC 957, 975-77 (1974). To the extent that the CEQ study mentioned by CAND might include health effects relevant to this facility, it could contradict certain of the NRC Staff's health-effects conclusions. Consequently, our action here will not preclude adjudication of such question.

In addition, CAND refers, in its November 24, 1980 filing, to a plan for the construction of a large ethanol production facility on the Susquehanna River about 15 miles upstream from the Berwick plant. CAND claims that this facility will release "hundreds of millions of gallons of liquid wastes" into the river annually, which will cause an increase in the growth of slime-forming organisms. This increase in fouling organisms would necessitate an increase in the chlorination of the water systems at the nuclear plant, according to CAND.[18]

CAND could be correct. If an ethanol facility is constructed upstream from the Susquehanna plant and does discharge large amounts of organic wastes into the river, it might necessitate an increase in the amount of chlorine used at the power plant, because organic waste in the river could provide nutrients which would favor a greater growth rate by slime-forming

[16]CAND "Motion and Responses Concerning Summary Disposition," p. 2.

[17]To the extent that either of CAND's responses included any facts at all, they were not presented through affidavit. By our Memorandum and Order Inviting Further Responses to Summary Disposition Requests, dated November 4, 1980, at pp. 4-5, we apprised CAND that factual information which may contradict material supplied by affidavit should likewise be presented by affidavit. Despite CAND's failure to supply affidavits, we have nevertheless given due account to such information as has been provided.

[18]CAND filing dated November 24, 1980, p. 3.

organisms. For that reason, we are denying the motion for summary disposition insofar as it bears upon the need for chlorination caused by the discharge into the Susquehanna River of liquid wastes from the proposed ethanol production facility.

3. **Findings of Fact.** Based on our review of the foregoing material, we make the following findings:

1. The purpose of chlorinating the water systems of the Susquehanna plant is to control the growth of slime-forming organisms on surfaces of equipment.

2. If mine acid drainage is released into the Susquehanna River in the future, it will not require an increase in the amount of chlorine used to treat the water. If mine acid drainage has any effect on the amount of chlorine that must be used to defoul the plant's water systems, it will decrease the amount needed, because acid mine drainage tends to decrease the biota of fresh waters.

3. It is very unlikely that toxic chemical pollutants which find their way into the Susquehanna River from the Butler Mine Water Tunnel will necessitate an increase in the amount of chlorine required to defoul the plant's water systems, because chlorination is not normally used to treat waters polluted with toxic chemicals.

4. Organic waste from an ethanol production facility could, if released into the Susquehanna River, provide nutrients which would increase the rate of growth of biofouling organisms and necessitate a greater than expected use of chlorine by the power plant.

5. At anticipated levels of chlorination, the plant's dechlorination system will remove chloramines and will reduce chlorides to levels which will not pose a significant threat to the public health.

6. No assessment of health effects of chlorine use at higher than anticipated levels, such as might be required if organic waste from an ethanol plant were released into the river upstream from the Berwick plant, has been made.

7. Trihalomethanes, which are suspected to be carcinogenic, probably will be released at low concentrations in the effluent at anticipated levels of chlorination. No quantitative estimate of the trihalomethane concentration to be expected in the plant's discharge has been made, however.

8. At anticipated levels of chlorination, the trihalomethane content of the discharge may, or may not, be below the maximum allowed by EPA under the Safe Drinking Water Act.

4. **Conclusions.** The Board concludes that there is no genuine issue of fact with regard to whether acid mine drainage or toxic chemical discharge will necessitate higher levels of chlorination than anticipated. Nor is there a genuine issue of fact with regard to the health effects of chlorides and

chloramines that will be produced at anticipated levels of chlorination at the plant. To the extent that Contention 2 (Chlorine) relates to these matters, therefore, we are granting the Applicants' motion for summary disposition. On the other hand, no assessment has been made of the health effects of a higher level of chlorination, should a higher level become necessary because of the discharge of organic wastes into the river upstream from the plant. Nor have the quantities and health effects of trihalomethanes and halomethanes to be released been adequately assessed, at anticipated or higher-than-anticipated levels of chlorination. To the extent that Contention 2 (Chlorine) relates to these matters, therefore, the Applicants' motion will be denied.

The health effects of various chlorine discharges, whether ascertained through this ruling or through evidentiary hearings, must, of course, be taken into account in any cost-benefit analysis conducted by the NRC with respect to this facility.

C. Contention 16 (Cooling Tower Discharge)

1. The Applicants filed their motion for summary disposition of Contention 16 on October 27, 1980. CAND, the proponent of that contention, responded on December 4, 1980, through a document entitled "Motion and Clarification Concerning Contention 16."[19] On December 5, 1980, the NRC Staff filed an answer in support of the Applicants' motion. In response to our invitation,[20] CAND filed a response to the Staff's answer on January 7, 1981.[21] No other party has taken a position on this motion.

Contention 16, as set forth in LBP-79-6, *supra*, 9 NRC at 320, reads:

> 16. Seventy million gallons of radioactive evaporated water to be vented daily from the Susquehanna facility's cooling towers will pose an economic threat to the dairy industry in the eastern-central area of Pennsylvania. This threat has not been properly evaluated.

The Applicants moved for summary disposition of this contention on the ground that there is no genuine issue of material fact to be heard with respect to the contention.

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[19]By our Order dated November 21, 1980, we granted an extension of time until December 5, 1980, within which parties might respond to the Applicants' motion. As in the case of Contention 2, CAND provided no affidavits in support of its response. See fn. 17, *supra*.
[20]Order dated December 9, 1980.
[21]As in the case of the Contention 2 motion, CAND's response was somewhat belated; it should have been filed by January 5, 1981. See Order dated December 9, 1980. No affidavits accompanied this response. See fn. 17, *supra*.

2. In support of their motion, the Applicants supplied the affidavit of Walter J. Rhoades, the Nuclear Group Supervisor—Mechanical, Nuclear Plant Engineering Department, Pennsylvania Power & Light Co. (hereinafter Rhoades affidavit). Mr. Rhoades attests that the water evaporated from the cooling towers comes from three sources of water supplied to the towers: makeup water, return flow from the Circulating Water System, and return flow from the Service Water System. He asserts that none of these sources is radioactive.[22] He explains that makeup water, which replaces water lost by evaporation, comes from the Susquehanna River and is not allowed to mix with any other plant water; therefore it cannot be the source of radioactive contamination. The other two sources, the Circulating Water System and Service Water System, draw water from the cooling towers and circulate the water through plant equipment for cooling, after which the water is returned to the cooling towers. Both systems are designed to prevent mixing of radioactive fluids with water from the cooling towers. Two independent methods are employed to prevent contamination of cooling tower water. First, physical barriers, *i.e.*, the tube walls in the heat exchangers, separate the radioactive fluids from the cooling tower water. Second, a pressure differential is maintained between the water of the Circulating Water and Service Water Systems and the systems which contain the radioactive fluids; thus if a leak were to occur, the flow would be from the Circulating Water or Service Water Systems into the systems containing radioactive fluids.[23]

Mr. Rhoades further attests that the Circulating Water System is at higher pressure than the steam in the condenser. Thus if a leak develops, water will flow out of the tubes into the condenser. Further, if the pressure of the condensing steam rises above 7.3 inches of mercury absolute, which is a lower pressure than that of the circulating water, the turbine is automatically tripped and the flow of steam to the condenser stopped.[24]

As for the Service Water System, Mr. Rhoades explains that it cools nineteen groups of equipment, only four of which contain potentially radioactive fluids. The four are: radwaste evaporator condensers, reactor building closed cooling water heat exchangers, gaseous radwaste recombiner closed cooling water heat exchanger, and fuel pool heat exchangers. As with the circulating water system, water in the service water system is maintained at a higher pressure than the radioactive or potentially radioactive fluids in the four groups of equipment.[25] The steam going to the radwaste evaporators is either non-radioactive or slightly radioactive and is

[22]Rhoades affidavit, p. 2.
[23]*Id.*, p. 3.
[24]*Id.*, p. 4.
[25]*Id.*, pp. 4-5.

at a pressure of 1 psig, whereas the service water supplied to the evaporators is at a pressure of approximately 128 psig. The water in the reactor building closed cooling water heat exchangers is circulated at a pressure of approximately 81 psig, whereas the service water supplied to the closed cooling water heat exchanger is supplied at a pressure of about 108 psig.[26] The gaseous radwaste recombiner closed cooling water heat exchangers all contain radioactive fluids at pressures less than 5 psig, whereas the service water is circulated through them at a pressure of approximately 76 psig.[27] Finally, the fuel water flows by gravity through the fuel pool heat exchangers, where it develops a head of about 30 psig; the service water circulates through the heat exchangers, however, at a pressure of 84 psig.[28]

The Applicants assert that the foregoing design features will prevent the water evaporated daily from the cooling towers from being radioactive.[29] Assuming that to be so, it follows that the threat to the dairy industry raised by CAND would not exist, and that Mr. Rhoades' review would constitute an adequate evaluation of the situation.

The NRC Staff reviewed the documents submitted by the Applicants in support of their motion and also independently evaluated the issue raised in the documents; the Staff concluded that Contention 16 lacks a factual basis.[30] In support of this conclusion, the Staff supplied the affidavits of Howard B. Holz, a Senior Reviewer in the Auxiliary Systems Branch, Division of Systems Integration, in NRC's Office of Nuclear Reactor Regulation, and Charles Lee Miller, a Nuclear Engineer in the Effluent Treatment Systems Branch, Division of Systems Integration. The Staff attested that there is no radioactivity released from the cooling towers in normal operation.[31] Although no radioactivity is expected in the service water system, a radiation monitor is located on the downstream side of the fuel pool heat exchangers prior to discharge to the cooling tower as a protection device, and the cooling tower blowdown will be sampled periodically for radioactivity. Should radioactivity be detected, measures can be taken to prevent a significant release.[32]

CAND responded to Applicants' motion for summary disposition of Contention 16 with a "clarification" which sets forth certain claims:[33]

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[26]*Id.*, p. 6.
[27]*Id.*, p. 7.
[28]*Id.*, p. 8.
[29]Applicants' Statement to Material Facts as to which There Is No Genuine Issue To Be Heard (Contention 16), dated October 27, 1980, p. 3.
[30]NRC Staff Answer, dated December 5, 1980, pp. 1-2.
[31]Holz affidavit, p. 2; Miller affidavit, p. 2.
[32]Miller affidavit, p. 4.
[33]Citizens Against Nuclear Dangers Motion and Clarification Concerning Contention 16, dated December 4, 1980, pp. 2-3.

(1) That massive cooling tower plumes of steam create severe and almost constant adverse local weather conditions including precipitation that in turn will cause the so-called routine radiation releases vented from the reactor to be carried into these plume storms and then directly back to the land surface in hot spots contaminating nearby vegetation in farm areas, with higher than permissible levels of radiation thereby endangering the food supply—most notably cattle feed and dairy products.

(2) In the event of the type of plumbing accident, such as occurred at Indian Point Unit 2 in October, 1980, radioactive water in one system could become mixed with separate cooling water and escape into the atmosphere devastating Salem Township![34]

As a third claim, CAND went on to state that it intends to submit a new contention alleging that the lack of "fail-safe backup systems" to prevent the type of "plumbing accident" referenced above and "subsequent massive release of radiation, could have disastrous consequences."

The first of these claims might possibly be regarded as a basis for a contention. But it includes no facts which would counteract the affidavits supplied by the Applicants and Staff. Indeed, to the extent that the routine radioactive releases to which reference is now made are not vented from the cooling towers, they have no bearing on Contention 16 as admitted to this proceeding. Absent the showing for a late-filed contention required by 10 CFR §2.714(a), we decline to consider whether the first claim might qualify as a new contention.

The second and third statements (which are related) are clearly irrelevant to the admitted contention: they both relate to accidental releases, whereas the contention concerns the water which is to be "vented daily"—*i.e.*, routine releases.[35] In fact, the third claim expressly mentions a new contention. Again, absent the showing required by 10 CFR §2.714(a), we decline to consider whether these statements might be acceptable as a new contention.

Further, it is clear that CAND's January 7, 1981 response to the motion for summary disposition, from which we quoted in our discussion of

[34]*Id.*, p. 3.

[35]The Indian Point Unit 2 "plumbing accident" was an occurrence which caused the accumulation of several inches of water on the containment floor. See IE Bulletin No. 80-24, November 21, 1980 (of which we take official notice). The design of the water systems, as described in the Applicants' and Staff's filings, indicates that water spilled into the floor of the containment would not be vented through the cooling tower; rather, such water would normally be pumped from the containment sump to holdup tanks. IE Bulletin No. 80-24, *supra*. In any case, an accident such as occurred at Indian Point 2 is not a routine occurrence which could give rise to the daily radioactive releases averred to in Contention 16.

Contention 2, is intended to apply to the Applicants' statements about Contention 16 as well as to the statements about Contention 2.[36] But here again CAND fails to present material or substantial facts to support its allegations. The facts submitted by the Applicants and Staff show that there will be no routine releases of radioactive material from the cooling towers and that the design of the plant will prevent radioactive water from the containment building (given an Indian Point type accident) from mixing with cooling water which is circulating through the cooling towers. CAND has not controverted these facts.

3. **Findings of Fact.** Based on our review of the foregoing material, we make the following findings:

1. Water evaporated from the cooling towers comes from three sources: makeup water from the Susquehanna River, return flow from the circulating water system, and return flow from the service water system.

2. The makeup water, which replaces water lost by evaporation, is not allowed to mix with any other plant water and consequently cannot be the source of radioactive contamination.

3. Water in the circulating water system will not become radioactive in normal operation because the water is separated from the radioactive fluid which it cools by physical barriers (tube walls) and the water circulates at a higher pressure than the radioactive fluid, so that radioactive fluid cannot leak into the circulating water system should a breach occur in the physical barriers.

4. Water in the service water system is also separated from radioactive fluids in the equipment which it serves by physical barriers, and the water circulates at a higher pressure than the radioactive fluid, so that radioactive fluid will not leak into the service water system should a breach occur in the physical barriers.

5. Should radioactive material get into the cooling tower water through some abnormal occurrence, it would be detected by radiation monitoring devices and procedures, so that measures could be taken to prevent a significant release to the environment.

4. **Conclusions.** We conclude that there is no genuine issue of mater.al fact pertaining to the foregoing findings; that, insofar as radioactivity is concerned, there is no threat to the dairy industry in Pennsylvania from the water to be evaporated from the cooling towers; and that Applicants' motion for summary disposition of Contention 16 should therefore be granted.

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[36] CAND "Motion and Responses Concerning Summary Dispostion," dated January 7, 1981, p. 2.

D. Order

Based on the foregoing findings and conclusions, it is, this 16th day of March, 1981

ORDERED

1. That the Applicants' motion for partial summary disposition of Contention 2 (chlorine) is *granted* to the extent that the contention involves chlorination to counteract releases upstream of mine acid drainage and chemical pollutants into the Susquehanna River.

2. That the Applicants' motion for partial summary disposition of Contention 2 (chlorine) is *denied* to the extent that the contention raises (a) the need for chlorination caused by the discharge into the Susquehanna River of liquid wastes from the proposed ethanol production facility; (b) the quantities and health effects of releases of trihalomethanes from the facility; and (c) the health effects of chlorine releases at levels permitted by governing EPA requirements.

3. That the Applicants' motion for summary disposition of Contention 16 is *granted*.

THE ATOMIC SAFETY AND LICENSING BOARD

Charles Bechhoefer, Chairman
ADMINISTRATIVE JUDGE

Dr. Oscar H. Paris
ADMINISTRATIVE JUDGE

Glenn O. Bright
ADMINISTRATIVE JUDGE

Cite as 13 NRC 349 (1981) **DD-81-3**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

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

PACIFIC GAS & ELECTRIC CO.
(Diablo Canyon Nuclear
Power Plant, Units 1 & 2) **March 26, 1981**

The Director of Nuclear Reactor Regulation denies a petition under 10 CFR 2.206 that requested special consideration of Class 9 accidents and analyses of several safety systems prior to the licensing of the Diablo Canyon Nuclear Power Plant. The petition was substantially similar to a petition denied in DD-80-22.

RULES OF PRACTICE: PETITIONS UNDER 10 CFR 2.206

As a general rule, persons must be prevented from using 10 CFR 2.206 as a vehicle for avoiding an existing forum where issues should be more logically presented. This principle is particularly applicable here, where the petitioners are asking the Director to take actions with respect to matters that are pending before a Licensing Board for resolution.

RULES OF PRACTICE: PETITIONS UNDER 10 CFR 2.206

The provisions of 10 CFR 2.206 contemplate requests to institute proceedings in connection with licenses already in force, not requests to institute proceedings to consider whether a license should be granted in the first instance.

NEPA: SEVERE ACCIDENT CONSIDERATIONS

As provided in the Commission's June 1980 policy statement, completed NEPA reviews need not be reopened in the absence of special circumstances.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By petition dated October 17, 1980, David S. Fleischaker on behalf of the Joint Intervenors to the Diablo Canyon Nuclear Power Plant operating license proceeding requested that the Director of Nuclear Reactor Regulation take action pursuant to 10 CFR 2.206 to require preparation of supplemental environmental impact statements on Class 9 accidents at the Diablo Canyon nuclear plant. The petition was supported by the affidavit of Mr. Richard B. Hubbard. Notice of receipt of the petition was published in the *Federal Register,* 45 Fed. Reg. 78317 (November 25, 1980). Counsel for the Pacific Gas and Electric Company (PG&E), the applicant for the Diablo Canyon plant, submitted on January 8, 1981, a response opposing the petition. The petition is similar to a petition filed by the Friends of the Earth in 1979, which was denied in June 1980. DD-80-22, 11 NRC 919 (1980).

The petition requests relief with respect to the Diablo Canyon Nuclear Power Plant, Units 1 and 2, which are being constructed at the PG&E site in California and for which PG&E has applied for operating licenses. The petition, supported by Mr. Hubbard's affidavit, asks that the Commission prepare supplemental environmental impact statements to consider the environmental consequences of Class 9 accidents, evaluate a number of safety features to prevent severe accidents or mitigate their consequences, and prepare a simplified system reliability analysis of several systems in the Diablo Canyon plant. The petition requests that these actions be taken before the Commission issues either the low power or the full term operating licenses for the Diablo Canyon plant. The petition provides the following bases for taking these actions:

1. The environmental impact statements summarily discuss consideration of Class 9 accidents, based on early estimates of reactor accident probabilities and on the Reactor Safety Study, WASH-1400, which has since been repudiated by the Commission;
2. The accident at Three Mile Island, which the NRC concedes constituted a Class 9 accident, emphasized the need to evaluate the possible impact of a serious (Class 9) accident and to prepare to meet the possible consequences;
3. Seismic conditions at the Diablo Canyon site constitute "special circumstances" which would require consideration of environmental consequences of Class 9 accidents under the Commission's policy; and
4. Analyses of Class 9 accidents, additional safety features and system reliability should be performed prior to the granting of any operating license for Diablo Canyon, either low or full power, because

radioactive contamination of the reactor will either foreclose or increase the economic costs and health risks associated with additional engineering safety features that may be required as a result of such analyses.

For the reasons stated in this decision, the petition is denied.

I. PROCEEDINGS RELATED TO THE PETITION

In the June 1980 denial of the Friends of the Earth's petition, the NRC staff has effectively dealt with those portions of the petition which request a supplemental environmental analysis of Class 9 accidents for Diablo Canyon. *See* DD-80-22, 11 NRC 919 (1980). In its petition the Friends of the Earth, like the petitioners here, raised the occurrence of the accident at Three Mile Island and the Commission's "repudiation" of the WASH-1400 study. The Director's decision denying the Friends of the Earth's petition applied the Commission's interim policy to the three sites, including Diablo Canyon, for which a supplemental environmental statement was sought and found that a supplemental environmental statement for any of the three sites was not mandated under the policy.[1] The Commission did not overturn the Director's decision. In the October 1980 petition, the petitioners have not presented any new information which would lead the staff to reconsider the conclusions reached in the decision on the Friends of the Earth's petition. Part II of this decision repeats the analysis provided in DD-80-22 and responds to the petitioners' contention that the site's seismic characteristics comprise "special circumstances" within the meaning of the Commission's new interim policy on severe accident considerations.

Apart from the recent denial of a substantially similar petition under 10 CFR 2.206, it should be noted that the petitioners here are parties to the operating license proceeding for Diablo Canyon and have a motion to reopen the proceeding to consider Class 9 accidents pending before the Licensing Board. The Board has determined that it will defer consideration of this motion until the Appeal Board has ruled on the seismic issues before it relating to Diablo Canyon. *Prehearing Conference Order,* at 3, 26-27

[1]The Commission's statement of interim policy indicates that the Three Mile Island accident was one of the reasons for withdrawing the proposed Annex to Appendix D of 10 CFR Part 50 and substituting a new interim policy on accident considerations in the Annex's place. *See Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969,* 45 Fed. Reg. 40101 *et seq.* (June 13, 1980). The Director rejected the so-called repudiation of WASH-1400 as a basis for preparing supplemental environmental impact statements. *See* DD-80-22, 11 NRC at 931-32, *citing* 45 Fed. Reg. 40102 *and Pennsylvania Power & Light Co.* (Susquehanna Steam Electric Station, Units 1 & 2), LBP-79-29, 10 NRC 586, 589 (1979). *See also Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1-4), CLI-79-10, 10 NRC 675 (1979).

(Feb.13,1981). Thus, the petition here attempts to raise under 10 CFR 2.206 the precise issues that await the Board's consideration in the operating license proceeding. As a general rule of practice, the Commission has stated that "parties must be prevented from using 10 CFR 2.206 procedures as a vehicle for reconsideration of issues previously decided, or for avoiding an existing forum in which they more logically should be presented." *Consolidated Edison Co.* (Indian Point, Units 1-3), CLI-75-8, 2 NRC 173, 177 (1975); *see also Public Service Co. of Indiana* (Marble Hill Nuclear Generating Station, Units 1 & 2), DD-79-21, 10 NRC 717, 724 (1979). That policy is particularly appropriate in this instance, where the petitioners are in effect asking the Director of NRR to establish certain conditions prior to licensing the Diablo Canyon plant — a matter currently before the Board.

The provisions of 10 CFR 2.206 are essentially a mechanism for requesting that the Commission take enforcement action by instituting a proceeding to modify, suspend, revoke, or take other action with respect to a license. The rule contemplates requests to institute proceedings in connection with licenses that are already in force, not requests to institute proceedings to consider whether a license should be granted in the first instance. The initial grant or denial of licenses should be considered in accordance with the procedural requirements of section 189a. of the Atomic Energy Act and Subpart A of 10 CFR Part 2. These considerations have particular applicability here, and in themselves serve as a basis for denying the petition.

II. COMMISSION POLICY ON ACCIDENT CONSIDERATIONS

The June 1980 Director's decision describes the evolution of the Commission's policy on severe accident considerations under NEPA DD-80-22, 11 NRC at 921-24. The term "Class 9 accident" was employed in a Commission rulemaking which had been proposed in December 1971: "Consideration of Accidents in Implementation of the National Environmental Policy Act of 1969," 36 Fed. Reg. 22851 (1971). The proposed rulemaking would have added an Annex to Appendix D of 10 CFR Part 50 to set forth the manner in which various categories of accidents should be taken into account in the environmental review for a nuclear power plant. In September 1979, the Commission announced in *Offshore Power Systems* (Floating Nuclear Power Plants), CLI-79-9, 10 NRC 257 (1979), that it intended to complete the rulemaking begun by the Annex and to re-examine the Commission's policy regarding accident considerations.[2] On May 16, 1980, the Commission issued a statement of interim policy in which it withdrew the proposed Annex and suspended the rulemaking that began in 1971 with the publication of the proposed Annex. *Nuclear Power Plant Accident Considerations under the National Environmental Policy Act of 1969*, 45 Fed. Reg. 40101 (June 13, 1980). The Commission also provided guidance on accident considerations in on-going NEPA reviews in licensing proceedings where a Final Environmental Statement has not yet been issued. Under the Commission's new guidance, environmental impact statements for on-going and future NEPA reviews will give consideration to a broader spectrum of accidents, including severe accidents that may have been designated "Class 9" under the Annex.

With respect to plants for which Final Environmental Statements have been issued, the Commission stated in its new interim policy that:

> "It is expected that these revised treatments will lead to conclusions regarding the environmental risks of accidents similar to those that would be reached by a continuation of current practices, particularly for cases involving special circumstances where Class 9 risks have been considered by the staff.... Thus, this change in policy is not to be construed as any lack of confidence in conclusions regarding the environmental risks of accidents expressed in any previously issued Statements, nor, absent a showing of similar special circumstances, as a basis for opening, reopening or expanding any previous or on-going proceeding.[5]
>
> "[5] Commissioners Gilinsky and Bradford disagree with the inclusion of the preceding two sentences. They feel that they are absolutely inconsistent with an even-handed reappraisal of the former, erroneous position on Class 9 accidents." 45 Fed. Reg. at 40203.

As the Commission noted in its new statement of interim policy, the staff has identified special circumstances in the past which would warrant more extensive consideration of Class 9 accidents. The special circumstances fell within three categories: (1) high population density around the proposed site; (2) a novel reactor design (a type of power reactor other than a light

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[2] In *Offshore Power Systems*, the Commission determined that consideration of a Class 9 accident in the environmental review for floating nuclear power plants was appropriate. 10 NRC at 260-61. The Commission did not use the proceeding to resolve the generic issue of consideration of Class 9 accidents at land-based reactors, but noted that "such a generic action is more properly and effectively done through rulemaking proceedings in which all interested persons may participate." *Id.* at 262. *See also Public Service Co. of Okla.* (Black Fox Station, Units 1 & 2), CLI-80-8, 11 NRC 433 (1980).

water reactor); or (3) a combination of a unique design and a unique siting mode.[3] In *Public Service Company of Oklahoma*, which was decided before the Commission stated its new interim policy, the Commission noted in addition to these three criteria that proximity of a plant to a "man-made or natural hazard" might also represent "the type of *exceptional* case that *might* warrant additional consideration.[4] In DD-80-22, the staff presented the following results of their review for "special circumstances" for Diablo Canyon.

As described in Sec. 4 of the Safety Evaluation Report[5] and Sec. 1.3 of the Final Safety Analysis Report[6] the Nuclear Steam Supply System for each unit of the Diablo Canyon plant is a Westinghouse pressurized water reactor using a four-loop coolant system. The reactor design is basically similar to that of several other Westinghouse reactor designs (Trojan, Zion 1 and 2, and D. C. Cook plants). The Diablo Canyon plant is, therefore, a typical light water reactor facility and the design is not novel.

The Diablo Canyon plant is located in a remote, undeveloped and relatively uninhabited region of San Luis Obispo County. Within 10 miles of the plant, the 1970 resident population density was about 20 persons per square mile. Within radii of 20 and 30 miles, the densities were 55 and 40 residents per square mile, respectively. The population densities were projected to approximately double by the year 2000, thus remaining well

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[3]*See* 45 Fed. Reg. 40102 (June 13, 1980); *Public Service Elec. & Gas Co.* (Salem Nuclear Generating Station, Unit 2), DD-80-17, 11 NRC 596, 615 n. 21 (April 1980). In the first category fell the Perryman site, for which the staff performed an informal assessment in the early site review of the relative differences in Class 9 accident consequences among the alternative sites. The Clinch River Breeder Reactor, a liquid metal cooled fast breeder reactor which is different from the more conventional light water reactor, fell within the second category, novel reactor design, and the staff included a discussion in the final environmental statement (NUREG-0139, Feb. 1977) of its consideration of Class 9 accidents.

The Floating Nuclear Power Plants represented the third category of special circumstances, a combination of unique design and a unique siting mode. Because the plants would be mounted on a floating barge, there would be no soil structure to retard the release and dispersal of activity beneath the plant following a core melt accident as would be the case for land-based plants. The staff concluded that the most likely exposure to the population from the liquid pathway for a floating nuclear plant is significantly greater than for a land-based plant.

[4]CLI-80-8, 11 NRC 433, 434 (Mar. 1980). The four criteria identified in the text have been applied to seven reactor sites in decisions under 10 CFR 2.206. On no occasion did the Commission disturb the Director's findings. *See Duke Power Co.* (Catawba Nuclear Station, Units 1 & 2), DD-81-1 (Docket Nos. 50-413 & 50-414, Jan. 9, 1981 & Addendum, Feb. 6, 1981); *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 2), DD-80-33 (Docket No. 50-389, Nov. 28, 1980); *Pacific Gas & Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2) *et al.*, DD-80-22, 11 NRC 919 (June 1980) (also addressing Palo Verde and Rancho Seco plants); *Public Service Electric & Gas Co.* (Salem Nuclear Generating Station, Unit 2), DD-80-17, 11 NRC 596 (Apr. 1980); *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 & 2), DD-80-6, 11 NRC 371 (Feb. 1980).

[5]Safety Evaluation Report for Diablo Canyon Station, Units 1 & 2 (Oct. 1977).

[6]Final Safety Analysis Report for the Diablo Canyon Station, Units 1 & 2.

within the guidelines of Regulatory Guide 4.7 and 10 CFR Part 100. Therefore, population distribution near the Diablo Canyon plant is not a "special circumstance" within the meaning of the Commission's policy statement.

The Diablo Canyon plant also does not represent a "combination of a unique design and a unique siting mode." The Diablo Canyon site is located adjacent to the Pacific Ocean, which is the only surface water body which could be affected by liquid releases from a Class 9 accident. Ground water near the site is limited to the streambed of Diablo Canyon Creek, an intermittent stream which empties into the ocean. The sandstone bedrock underlying station foundations is, at most, partially saturated (i.e., no water table) for a considerable vertical distance. Its low permeability, combined with the lack of a near surface water table, would preclude lateral movement of contaminated water from the station toward the ocean at more than an extremely slow rate. At a minimum, many years would be available to interdict any such flow. Therefore, there are no unusual hydrogeologic features of the site which would warrant special consideration of the environmental consequences of Class 9 accidents at the Diablo Canyon site.

The staff analyzed the site characteristics and other nearby features to assure the potential for impairment of safety-related portions of station facilities due to natural or man-made hazards occurring nearby. The Safety Evaluation Report states the staff's conclusion that there are no industrial, transportation, or military facilities in the area of the site which have potential to adversely affect plant safety systems. The staff's review specifically ensures that station design is adequate to accommodate other natural characteristics of the site environs. The staff's review has not identified any unusual circumstances with respect to external hazards that would warrant reopening or expanding proceedings on Diablo Canyon.

The petitioners point to the seismic characteristics of the Diablo Canyon site as a "special circumstance" that would warrant consideration of Class 9 accidents. The petitioners base this characterization on a quotation from ALAB-519 in which the Appeal Board described the circumstances surrounding the need to make additional findings regarding the plants' seismic design capability as "exceptional in every sense of that word."[7] The Appeal Board was not considering, of course, in that decision whether the seismic characteristics of the site constituted "special circumstances"

[7] *Pacific Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plants, Units 1 & 2), ALAB-519, 9 NRC 42, 46 (1979), *quoted* in Fleischaker letter at 3 and Hubbard's affidavit at 10.

warranting consideration of Class 9 accidents under the Commission's policy.[8] The Appeal Board only addressed whether a sufficient showing had been made to warrant issuance of a subpeona to two ACRS consultants.[9]

As set forth in its testimony in the operating license proceeding, the staff believes that the Diablo Canyon plant meets the Commission's criteria for seismic design of nuclear facilities. The staff does not believe, therefore, that the seismic characteristics of the Diablo Canyon site constitute "special circumstances" which would warrant special consideration of a Class 9 accident. The staff recognizes, however, that site seismicity and the related design capability of the plants have been subject to additional hearings, most recently in October 1980 before the Appeal Board, in the operating license proceeding. The Appeal Board has not yet rendered a decision on the reopened seismic record. Should the Appeal Board decide that the plants do not meet the Commission's criteria, the staff would reconsider its view that the seismic conditions do not constitute "special circumstances." Any such inquiry will be conducted, however, in the operating license proceeding, not in the context of 10 CFR 2.206. As noted earlier in this decision, the Licensing Board has deferred consideration of the petitioners' motion to consider Class 9 accidents until after the Appeal Board makes its decision.

III. ADDITIONAL SAFETY ANALYSES REQUESTED BY PETITIONERS

In support of the petition, Mr. Fleischaker submitted the affidavit of Richard B. Hubbard. Mr. Hubbard points to a number of safety features and reviews which are discussed in NUREG-0660, *NRC Action Plans Developed as a Result of the TMI-2 Accident,* and then asks that these features be analyzed and the reviews performed prior to the licensing of Diablo Canyon for low power testing or full power operation.[10] In a revised statement of policy on reactor licensing, the Commission has discussed the purpose of the NRC's Action Plan:

> "The Action Plan was developed to provide a comprehensive and integrated plan for the actions judged appropriate by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. In developing the Action Plan, the various recommendations

[8]The policy in effect at the time was the proposed Annex.
[9]This showing is necessary because under 10 CFR 2.720 NRC personnel are not amenable to subpoena except "upon a showing of exceptional circumstances."
[10]*See* Hubbard affidavit at 11-15.

and possible actions of all the principal investigations were assessed and either rejected, adopted or modified. A detailed summary of the development and review process for the Action Plan was initially provided in NUREG-0694, 'TMI-Related Requirements For New Operating Licenses,' and can now be found, as changed, in NUREG-0737, 'Clarification of TMI Action Plan Requirements.' " *Further Commission Guidance for Power Reactor Operating Licenses-Revised Statement of Policy,* 45 Fed. Reg. 85236 (Dec. 8, 1980) (footnotes omitted).

In this revised statement of policy, the Commisssion discussed the requirements for new operating licenses and the schedule for implementing the actions proposed in the Action Plan:

> "In approving the schedules for developing and implementing changes in requirements, the Commission's primary considerations were the safety significance of the issues and the immediacy of the need for corrective actions. As discussed above, many actions were taken to improve safety immediately or soon after the accident. These actions were generally considered to be interim improvements. In scheduling the remaining improvements, the availability of both NRC and industry resources was considered, as well as the safety significance of the actions. Thus, the Action Plan approved by the Commission presents a sequence of actions that will result in a gradually increasing improvement in safety as individual actions are completed and the initial immediate actions are replaced or supplemented by longer term improvements....
>
> "Based upon its extensive review and consideration of the issues arising as a result of the Three Mile Island accident — a review that is still continuing — the Commission has concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0737 can provide a basis for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured by the NRC staff against the regulations, as augmented by these requirements. In general, the remaining items of the Action Plan should be addressed through the normal process for development and adoption of new requirements rather than through immediate imposition on pending applications." 45 Fed. Reg. at 85238.

Litigation of TMI-related requirements should be conducted in accordance with this policy and the Commission's procedural requirements before the Licensing Board, not within the context of 10 CFR 2.206.

Several remarks should be made regarding the specific actions which the petitioners culled from NUREG-0660 and urged should be fed into the Diablo Canyon licensing process at this time. The petitioners point to the safety reanalysis of the Indian Point and Zion facilities and ask that such an analysis be instituted for Diablo Canyon. The reanalysis includes consideration of mitigation features for severe accidents. The Indian Point-Zion study, as well as a study of the Limerick site, was initiated specifically because these plants are located in areas of high population density. The Diablo Canyon site does not share this population characteristic with these other sites such that a special study is warranted. Moreover, the features which Mr. Hubbard identifies as part of the Indian Point-Zion study are also under consideration for all plants as part of a proposed rulemaking on consideration of degraded or melted cores in safety regulation. *See Advanced Notice of Proposed Rulemaking*, 45 Fed. Reg. 65474 (Oct. 2, 1980). The Commission has also proposed interim requirements related to hydrogen control and certain degraded core considerations. *See* 45 Fed. Reg. 65465 (Oct. 2, 1980).

Mr. Hubbard's affidavit also urges consideration of groundwater interdiction methods to control radioactive contamination of water. As discussed above, the ground water characteristics of the site do not compromise "special circumstances" requiring consideration of Class 9 accidents. Special action for Diablo Canyon is not required at this time. The NRC staff proposed a further detailed study of the hydrologic features of all reactor sites in the Action Plan. The liquid pathway interdiction study is designated Task Action III.D.2. Based on currently available data, there is a small likelihood of any hydrologic problems at the Diablo Canyon site. In the event that significant possible impacts are identified in the study, methods of interdiction and mitigation will be specified. A number of mitigation methods are available, including pumping and construction of slurry walls.

Mr. Hubbard also notes the Integrated Reliability Evaluation Program (IREP) and urges that a similar simplified system reliability analysis be performed for a number of systems. *See* Hubbard affidavit at 15. The IREP is described in NUREG-0660 (Task II.C, at pp. II.C-2 to II.C-5). The IREP is a pilot program at present and does not consider seismic or other natural phenomena sequence initiators. NUREG-0660 describes a gradual implementation of IREP and studies for operating reactors. No special reason exists for changing this schedule to apply IREP directly to Diablo Canyon at this time. It should be noted that a study of systems interaction for seismically induced events has been conducted for Diablo Canyon. *See* NUREG-0660, Task II.C.3. This study has been completed and any necessary plant modifications for each unit will be made before any license is issued that authorizes full-power operation.

Finally, Mr. Hubbard points to possible radioactive contamination of the plant as a basis for instituting the analyses listed in his affidavit. Again, the question of whether PG&E is entitled to either a low power or full power license is before the Licensing Board for resolution in light of the Commission's policy on issuance of operating licenses and the schedule for completing required actions before full power licenses may be issued. It should be noted in this regard that the health or economic costs of installing additional engineered safety features are not significantly affected by low power tests, because the fission product inventory for such operation would be less than 0.10 of one percent of the inventory generated by the plant during one year of operation at full power.

IV. CONCLUSION

For the reasons stated in this decision, the staff does not believe that "special circumstances" exist for the Diablo Canyon plant that would warrant consideration of Class 9 accidents. To the extent that the petitioners raise matters that should be considered as conditions for issuance of the Diablo Canyon operating licenses, those matters should be addressed in the operating license proceeding and not resolved under 10 CFR 2.206. The petition is, therefore, denied.

A copy of this decision will also be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c) of the Commission's regulations. As provided in 10 CFR 2.206(c), this decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland
this 26th day of March, 1981

Cite as 13 NRC 361 (1981) **CLI-81-5**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

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

PACIFIC GAS AND ELECTRIC COMPANY
(Diablo Canyon Nuclear Power Plant, Units 1 and 2) **April 1, 1981**

Upon its review of a prehearing conference order by the Licensing Board in this proceeding, the Commission issues additional guidance on the litigation of TMI-related issues in licensing proceedings in the following areas: (1) motions for fuel loading and low power testing; (2) reopening evidentiary records (generally); (3) reopening records to address contentions alleging TMI-related violations of NRC regulations not addressed in previous Commission policy statements on litigation of TMI-related issues (NUREG-0737 and -0694); and (4) challenges under the Commission's December 18, 1980 Revised Statement of Policy (CLI-80-42, 12 NRC 854) that there is insufficient protection to the public despite compliance with either NRC regulations or TMI-related requirements of NUREG-0737 and -0694.

NUCLEAR REGULATION COMMISSION: SUPERVISORY AUTHORITY (ADJUDICATIONS)

The Commission has inherent supervisory authority over pending adjudications. *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-77-8, 5 NRC 503 (1977).

ORDER

The Commission has reviewed the Atomic Safety and Licensing Board's Prehearing Conference Order dated February 13, 1981, as well as the underlying papers and oral argument, and determined that additional Commission guidance, consistent with its Revised Statement of Policy, CLI-80-42, 12 NRC 654 (1980), needs to be provided on litigation of Three Mile Island (TMI) accident related issues in licensing proceedings. The Commission recognizes that this guidance could lead to reconsideration of some of the various rulings contained in the February 13, 1981 Order. In providing this guidance the Commission is exercising its inherent supervisory authority over pending adjudications.[1] *See Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-77-8, 5 NRC 503 (1977).

1. **The Board Should Rule Promptly on Motions for Fuel Loading and Low Power Testing**

   Pursuant to 10 CFR 50.57(c), the filing of a motion for a partial initial decision on fuel loading and low power testing requires an initial determination by the Licensing Board on whether the evidentiary record compiled to that point is adequate for such a partial decision. 10 CFR 50.57(c) does not generally contemplate that a new evidentiary record, based on litigation of new contentions, would be compiled on the motion for fuel loading and low power testing. When the record has been closed but motions to reopen have been filed, the Licensing Board should decide whether the record must be reopened for new evidence directly relevant to the fuel loading and low power licensing request. Decisions on full power issues associated with the motion to reopen could be postponed until later.

2. **The Record Should Not Be Reopened Absent a Showing that Significant New Evidence Which Would Affect the Decision Is Available**

   As we stated in the Revised Policy Statement, where the evidentiary record on safety issues has been closed, the record should not be reopened on TMI-related issues relating to either low or full power absent a showing, by the moving party, of "significant new evidence not included in the record, that materially affects the decision." This is

[1]The Commission is aware of the various participants' requests for certification or directed certification to the Commission regarding the February 13, 1981 Prehearing Conference Order. These motions appealing an interlocutory order are not provided for in the Commission's Rules of Practice and are accordingly denied. 10 CFR 2.730(f). In issuing this Order the Commission is exercising its authority *sua sponte*. The Union of Concerned Scientists' Request to Participate as Amicus Curiae is similarly denied.

in accord with longstanding Commission practice. *E.g. Kansas Gas & Electric Co., et al.* (Wolf Creek Generating Station, Unit 1), ALAB-462, 7 NRC 320, 338 (1978). We emphasize that bare allegations or simple submission of new contentions is not sufficient. Only significant new evidence requires reopening. Of course, in moving to reopen, a party need not supply written testimony of independent experts, but is free to rely on admissions and statements from applicant and NRC staff and official NRC documents or other documentary evidence.

3. **Where A Party Can Adduce Significant New Evidence That an NRC Regulation Would Be Violated by Plant Operation, that Contention Should Be Admitted Notwithstanding the Fact that this Matter Is Not Addressed in NUREG-0737 and -0694**

   Parties are generally free to raise issues of compliance with NRC regulations, subject to 10 CFR 2.714 specificity and lateness requirements, where applicable, and standards for reopening records, where applicable. This holds true for TMI-related issues, and nothing in the Revised Policy Statement affects this. Thus, if a party comes forward on a timely basis with significant new TMI-related evidence indicating that an NRC safety regulation would be violated by plant operation, we believe that the record should be reopened notwithstanding that the noncompliance item is not discussed in NUREG-0737 and -0694. However, the parties are required to make the initial case that significant new evidence is available, not merely make claims to that effect.

4. **Procedures for Arguing that there is Insufficient Protection to the Public Despite Compliance with All NRC Regulations**

   Where the new evidence raises no issue of compliance but rather questions whether there is adequate protection despite compliance with all applicable regulations, a party has two procedural options under the Revised Statement of Policy. First, a party may challenge the sufficiency of an item in the NUREG documents. However, the scope of the inquiry under this option is limited to the particular safety concerns that prompted the specific "requirements" in NUREG-0694 and -0737. What we had in mind was allowing a party to focus on the same safety concern that formed the basis for the NUREG requirement and litigate the issue of whether the NUREG "requirement" is a sufficient response to that concern.[2] Contentions which address a

[2]For example, the Item I.A.1.3 of NUREG-0737, which deals with shift manning and imposes additional requirements above and beyond 10 CFR 50.54(k), deals with the safety concern that

safety concern not considered in NUREG-0694 and -0737 shall not be entertained as challenges to the sufficiency of those requirements. Second, where the contention or new evidence cannot be associated with a safety concern identified by NUREG-0694 or -0737, 10 CFR 2.758 may be used to bring the matter to the Commission's attention without prior litigation on the merits. In this situation, a party must first make a prima facie case to the Board that application of a given rule in this particular proceeding would not serve the purpose for which that rule was adopted. If the party is able to make this case, the Commission will determine whether that rule will be waived or an exception made from its requirements in that case.

We note that quite apart from the procedures of 10 CFR 2.758, parties are always free to bring to the attention of the Commission any matter within its jurisdiction. This course would be available to a party even where a Board had ruled that the party had not made the prima facie case required by 10 CFR 2.758. In such cases, the Commission is under no obligation to respond to the matter.

In addition, of course, the specificity and lateness requirements of 10 CFR 2.714 must be satisfied, where applicable, and the standards for reopening records must be satisfied, where applicable. Thus, to have a late filed contention admitted, the following factors must be considered:

(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 issue or delay the proceeding.

In addition, the proponent of reopening the record must present significant new information, a requirement which could be satisfied by

there must be adequate expertise in the control room at all times to cope with any accident or unexpected event. The concern does not relate to the general design of the control room or to the need for specific control room equipment. Thus, a contention which purports to challenge the sufficiency of the shift manning requirement would have to be based on the argument that this requirement was inadequate to deal with control room staffing, and a challenge to Item I.A.1.3 which focused on control room design and equipment would not be permissible.

reference to new information in NUREG-0737. Finally, it must be shown that the new information would have caused a different result had it been considered originally.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
the 1st day of April, 1981.

Cite as 13 NRC 367 (1981) **ALAB-637**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket Nos. 50-498 OL**
50-499 OL

HOUSTON LIGHTING & POWER COMPANY, *et al.*
(South Texas Project, Units 1 and 2) **April 16, 1981**

The Appeal Board dismisses intervenors' impermissible interlocutory appeals and, treating their filings as requests for directed certification, denies discretionary interlocutory review of the Licensing Board's scheduling order and specification of issues to be considered at an expedited hearing on quality assurance and quality control matters in this operating licensing proceeding.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

The Commission's Rules of Practice prohibit appeals of interlocutory licensing board rulings. 10 CFR 2.730(f).

RULES OF PRACTICE: DISCRETIONARY INTERLOCUTORY REVIEW

Requests for discretionary interlocutory review via directed certification pursuant to 10 CFR 2.718(i) and 2.785(b)(1) are granted infrequently and then only when a licensing board's action either (a) threatens the party adversely affected with immediate and serious irreparable harm which could not be remedied by a later appeal, or (b) affects the basic structure of the proceeding in a pervasive or unusual manner. *Public Service Electric and*

Gas Co. (Salem Station, Unit 1), ALAB-588, 11 NRC 533, 536 (1980), and cases cited there.

RULES OF PRACTICE: SCHEDULING OF HEARINGS

An appeal board generally will examine a licensing board's scheduling decision only if it is claimed that the licensing board abused its discretion by setting a hearing schedule that deprives a party of its right to procedural due process. *Public Service Co. of Indiana, Inc.* (Marble Hill Station, Units 1 and 2), ALAB-459, 7 NRC 179, 188 (1978).

RULES OF PRACTICE: DISCRETIONARY INTERLOCUTORY REVIEW

A licensing board's specification of issues to be heard, like its rejection of some (but not all) of a party's contentions, is a procedural matter not warranting interlocutory review absent exceptional circumstances.

RULES OF PRACTICE: APPELLATE REVIEW

Partial initial decisions that dispose of significant issues but do not yet authorize construction activity are appealable as of right. *Houston Lighting and Power Co.* (Allens Creek Station, Units 1 and 2), ALAB-301, 2 NRC 853, 854 (1975).

APPEARANCES

Mr. Lanny Sinkin, Austin, Texas, for intervenors Citizens Concerned About Nuclear Power, Inc., and Citizens for Equitable Utilities.

Messrs. Jack R. Newman, Maurice Axelrad, and **Alvin H. Gutterman**, Washington, D. C., and **Messrs. Finis E. Cowan** and **Thomas B. Hudson, Jr.**, Houston, Texas, for applicants Houston Lighting & Power Company, *et al.*

Mr. Jay M. Gutierrez for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

I.

Intervenors Citizens Concerned About Nuclear Power, Inc. (CCANP), and Citizens for Equitable Utilities (CEU) have jointly filed two pleadings, each requesting interlocutory review of portions of the Licensing Board's April 1, 1981, Third Prehearing Conference Order. The first, styled a "Notice of Appeal," objects to the Licensing Board's denial of intervenors' motions for a 90-day postponement of the scheduled hearing date in this case.[1] The second pleading is a "Notice of Appeal and Request for Directed Certification" of the Licensing Board's specification of issues set forth in its Second Prehearing Conference Order (December 2, 1980) and reaffirmed in its Third Prehearing Conference Order.[2]

Intervenors argue that they established "good cause" to warrant a substantial delay in the start of the hearing. Specifically, they point to (1) the extended illness of Mrs. Peggy Buchorn, described as "the only representative of [CEU] with the expertise and experience to serve as intervenor in these proceedings;" (2) the unexpected withdrawal of CCANP's legal counsel two weeks before the Third Prehearing Conference; and (3) the unavailability during May of CCANP's representative (Mr. Lanny Sinkin). Intervenors contend that the Licensing Board's refusal to delay the hearing in view of these factors "adversely impacts the goal of a complete record in the initial hearing by restricting the ability of intervenors to prepare for and participate in the hearing." They also assert that the Board gave too much weight to hearing room availability and personal scheduling conflicts in devising the hearing schedule.

In their second pleading, intervenors argue that the Board's delineation of the issues under consideration in the upcoming initial hearing denies them certain relief "specifically mandated" by the Commission in response to a prior request of intervenors. See CLI-80-32, 12 NRC 281 (1980). In that proceeding, intervenors requested a hearing on an order issued by the Commission's Director of the Office of Inspection and Enforcement, directing applicants to show cause why safety-related construction activities at South Texas should not be halted pending modification of certain operations and procedures. The Commission denied intervenors' hearing

[1] Intervenors later suggested a 30-day postponement as an alternative. Tr. 379, 385. The hearing is scheduled to commence May 12, 1981.

[2] Although intervenors address this pleading to the Commission, under the Rules of Practice, requests of this nature fall within our jurisdiction. See 10 CFR 2.718(i), 2.785(b)(1); *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-382, 5 NRC 603, 604 n.1 (1977).

request but granted them the alternative relief of litigating the "quality assurance/quality control" (QA/QC) issues they raised in this operating license adjudication. The Commission further ordered the Board assigned to the licensing proceeding to expedite the hearing on the quality control-related issues and to issue an early, separate decision on this matter. *Id.* at 291-292. Pursuant to that order, the Licensing Board held a prehearing conference and formulated the QA/QC issues that would be the focus of the expedited hearing. Second Prehearing Conference Order, Attachment; see also Third Prehearing Conference Order at 8-11. Intervenors contend that the issues as framed by the Licensing Board improperly deal with applicants' alleged remedial activities. They assert that the relief that the Commission granted them permits consideration in the separate, expedited hearing of *only* the applicants' *past* actions.

Both the NRC staff and the applicants oppose intervenors' requests, primarily on the ground that intervenors have failed to show the "exceptional" circumstances necessary to warrant interlocutory review of either ruling.

II.

The Commission's Rules of Practice prohibit appeals of interlocutory licensing board rulings such as those involved here. 10 CFR 2.730(f). We will therefore treat both of intervenors' filings as requests for discretionary interlocutory review via directed certification. See 10 CFR 2.718(i) and 2.785(b)(1). Such requests, however, are granted infrequently "and then only when a licensing board's action either (a) threatens the party adversely affected with immediate and serious irreparable harm which could not be remedied by a later appeal, or (b) affects the basic structure of the proceeding in a pervasive or unusual manner." *Public Service Electric and Gas Co.* (Salem Station, Unit 1), ALAB-588, 11 NRC 533, 536 (1980), and cases cited. Intervenors have not satisfied these criteria as to either of their requests.

A.

Intervenors ask us to review and overturn the Licensing Board's denial of a postponement of the QA/QC hearing. But as we have stated previously,

> ... we enter the scheduling thicket cautiously. We are inclined to do so only to entertain a claim that a board abused its discretion by setting a

hearing schedule that deprives a party of its right to procedural due process.

Public Service Co. of Indiana, Inc. (Marble Hill Station, Units 1 and 2), ALAB-459, 7 NRC 179, 188 (1978). See also *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-295, 2 NRC 668, 669-670 (1975). Moreover, we are particularly loath to interfere with a licensing board's denial of a request to delay a proceeding where, as here, the Commission has ordered an expedited hearing. In such a circumstance, there must be a compelling demonstration of a denial of due process or the threat of immediate and serious irreparable harm in order to invoke our discretionary review.

Intervenors' arguments do not rise to this level. They contend that, absent at least a 30-day extension of the hearing date,[3] their ability to prepare for and participate in the initial hearing will be "seriously restricted." As noted above, they attribute their need for additional time to the illness of CEU's representative, the withdrawal of CCANP's counsel, and the unavailability of CCANP's representative to prepare for the May hearing. Yet balanced against these considerations are the following facts: (1) intervenors have known since November 19, 1980, that the hearing would commence in early May 1981 and that alterations to the schedule would be disfavored (Tr. 322-323; Second Prehearing Conference Order at 5-7); (2) intervenors have not provided any specific explanation as to why no other members of their organizations are available or able to participate in the upcoming hearing;[4] (3) the parties will have had almost two full months between the Board's oral ruling (at the Third Prehearing Conference) denying the postponement and the first day of hearing; and (4) perhaps most important, the Commission ordered this hearing to be expedited almost seven months ago.

We cannot say that the Licensing Board abused its discretion or denied intervenors due process in weighing these competing interests and devising a schedule that necessarily takes account of other exigencies (such as hearing room and judge availability).[5] Intervenors have failed to show that

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[3]See note 1, *supra*.

[4]In this regard, we note that CCANP recently filed with us a pleading in another pending "appeal" involving *South Texas*, signed by a representative other than Mr. Sinkin and showing an attorney "of counsel" (CCANP's "Opposition to NRC's 'Notice of Appeal and List of Exceptions' and Cross-Appeal-March 24, 1981"). In addition, Mrs. Buchorn of CEU stated at the Third Prehearing Conference that she would do "everything in [her] power to be ready by [the May 12 hearing date]." Tr. 380.

[5]The Board also made numerous efforts to accommodate *intervenors'* needs in setting the hearing schedule. Tr. 389, 391, 393-396. Moreover, while denying a delay in the hearing, it granted intervenors' requests to file its witness lists out of time. Third Prehearing Conference Order at 6.

the Board's hearing schedule will either cause them "immediate and serious irreparable harm" or affect "the basic structure of the proceeding in a pervasive or unusual manner." Under these circumstances, our intervention in this scheduling dispute is not justified.

B.

We must similarly deny intervenors' request for review of the Licensing Board's specification of issues for consideration in the QA/QC hearing.[6] This is yet another procedural matter within the Licensing Board's discretion, not warranting our interference absent a showing of the exceptional circumstances specified in *Salem, supra.* And again, intervenors have failed to demonstrate such circumstances exist here.

As we understand intervenors' argument, their principal objection to the issues formulated by the Board is that they assertedly cover more (and new) quality control-related matters than the Commission intended in its September 1980 order. Intervenors contend that this action thus "denies [them] relief specifically mandated by the Commission in said Memorandum and Order." Beyond this generalized assertion, however, intervenors fail to explain exactly how the Board's statement of issues results in such a denial of relief and consequently "immediate and serious irreparable harm."

Assuming *arguendo* that the Board's issues do broaden the intended scope of the hearing ordered by the Commission,[7] we do not see how this denies intervenors, either in fact or in effect, the separate, expedited hearing on their QA/QC issues. The Licensing Board has not issued any final ruling on applicants' QA/QC program, and intervenors will be free to pursue their related contentions and issues at the hearing. This is neither immediate nor serious irreparable harm. Moreover, we perceive no pervasive or unusual effect on the basic structure of the hearing as a result of an alleged broadening of the QA/QC issues.

Intervenors' request for review of the Board's delineation of issues in this special hearing is analogous to a request for review of a licensing board's rejection of some, but not all, of a party's contentions advanced in connection with its petition to intervene. As we pointed out in *Texas Utilities Generating Co.* (Comanche Peak Steam Electric Station, Units 1 and 2), ALAB-599, 12 NRC 1, 2 (1980), this type of appeal "is unauthorized by the Commission's Rules of Practice" since it does not dispose of the

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[6]The Licensing Board expressly denied intervenors' motion to certify this question to us. Third Prehearing Conference Order at 11.

[7]In view of our denial of the request for directed certification, we express no judgment on whether the Board's ruling is consistent with the Commission's Order.

petition in its entirety. See 10 CFR 2.714a. But, as we also noted, a party aggrieved by such board action can raise the rejection of these contentions on appeal from the board's initial decision. 12 NRC at 2 n.1. The same is true in this case. If indeed the Board's specification of issues is at odds with the Commission's direction and ultimately causes harm to intervenors, they will have every opportunity to challenge the Board's partial initial decision — issued after the hearing — on appeal.[8]

In sum, intervenors have failed to demonstrate — and we are unable to find — the exceptional circumstances that would warrant the exercise of discretionary review, via directed certification, of the two procedural rulings challenged here. Accordingly, intervenors' appeals are *dismissed* and their request for directed certification is *denied*.[9]

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

[8] *Houston Lighting & Power Co.* (Allens Creek Station, Units 1 and 2), ALAB-301, 2 NRC 853, 854 (1975). See also Third Prehearing Conference Order at 11.

[9] Still pending our consideration in this case are the staff's "Notice of Appeal and List of Exceptions" and "Motion for Directed Certification," which relate to a separate Licensing Board ruling.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. Lawrence R. Quarles
Thomas S. Moore

In the Matter of **Docket No. 50-409 SFP**

DAIRYLAND POWER COOPERATIVE (LaCrosse Boiling Water Reactor) **April 27, 1981**

On mootness grounds, the Appeal Board (1) dismisses the referral by the Licensing Board of its ruling in Part III of LBP-80-2 (11 NRC 44, 65-77); (2) dismisses a related NRC staff exception; and (3) vacates Part III under the doctrine of *United States* v. *Munsingwear,* 340 U.S. 36 (1950).

MEMORANDUM AND ORDER

The Dairyland Power Cooperative holds a provisional operating license for its LaCrosse nuclear facility. In an initial decision rendered in January 1980, the Licensing Board acted favorably on Dairyland's application for an amendment to that license which would permit the expansion of the storage capacity of the facility's spent fuel pool. LBP-80-2, 11 NRC 44. That result was later affirmed by us in ALAB-617, 12 NRC 430 (1980).

As ALAB-617 reflects, a sharp controversy had arisen below respecting whether, in the course of passing upon the spent fuel pool expansion proposal, the Licensing Board should inquire into the continued need for the power generated by the LaCrosse facility. Rejecting the insistence of both the applicant and the NRC staff that it lacked the jurisdiction to do so, the Board had conducted an evidentiary hearing on the need-for-power question and then, in the initial decision (11 NRC at 77 *et seq.*), had found that LaCrosse-generated electricity would be needed at least until the end of 1982.

That finding (which was not challenged before us by any party) obviously stripped the Board's jurisdictional ruling of any significance insofar as this proceeding is concerned.[1] Nonetheless, at the end of its initial decision the Board fulfilled a prior oral commitment to refer the ruling to us under 10 CFR 2.730(f). 11 NRC at 104. In addition, the staff filed an exception to the initial decision for the stated purpose of bringing into specific question one of the underpinnings of the ruling.

Given the fact that by then the jurisdictional ruling had become entirely academic (and the additional fact that the ruling had rested upon the particular and seemingly *sui generis* circumstances of this case), in ALAB-631 we might well have simply dismissed both the referral and the exception on that ground. We chose, however, to follow a different course by reason of the pendency of a similar (albeit not identical) question which had been raised in another spent fuel pool expansion proceeding, decided by the Licensing Board and likewise referred to us. *Consumers Power Co.* (Big Rock Point Nuclear Plant), LBP-80-25, 12 NRC 355 (1980). That referral had been promptly accepted because "unlike the situation in *LaCrosse,* the *Big Rock Point* ruling had an immediate and significant practical effect". ALAB-617, 12 NRC at 432. In the totality of circumstances, it seemed "prudent" to hold the *LaCrosse* referral and the related staff exception in abeyance to await our determination of the *Big Rock Point* controversy. *Ibid.*

Big Rock Point has now been decided. ALAB-636, 13 NRC 312, 318 (March 31, 1981) (petition for Commission review pending). In light of that decision, we have reexamined the matter of the warrant for undertaking a close examination of the merits of the *LaCrosse* ruling in its current wholly academic setting. We find none. Stated otherwise, there simply is no need to spend time and resources on a moot inquiry presented in a context which appears most unlikely to recur.

[1] The basis for the ruling was developed in the initial decision, 11 NRC at 65-77.

For this reason, (1) the referral by the Licensing Board of the ruling contained in Part III of LBP-80-2, *supra,* 11 NRC at 65-77; and (2) the related exception of the NRC staff are *dismissed* on the ground of mootness. On the same ground, Part III is *vacated. United States* v. *Munsingwear,* 340 U.S. 36 (1950); *Northern States Power Co.* (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 55 (1978).

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the
Appeal Board

Cite as 13 NRC 377 (1981) LBP-81-9

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Herbert Grossman, Chairman
Dr. Oscar H. Paris
Frederick J. Shon

In the Matter of

Docket No. 50-155 OLA
(Spent Fuel Pool Expansion)

CONSUMERS POWER COMPANY
(Big Rock Point Nuclear Plant)

April 22, 1981

The Licensing Board issues a memorandum in response to the Appeal Board's decision in ALAB-636, 13 NRC 312 (1981), which reversed that Licensing Board's determination in LBP-80-25, 12 NRC 355 (1980), that Section 102(2)(C) of NEPA requires the preparation of an EIS covering the impacts of a proposed spent fuel pool expansion for a plant that had been licensed before NEPA.

MEMORANDUM

(Reassuring Staff Of Lack Of Prejudgment Of Environmental Assessment Suggested By ALAB-636)

In its decision of March 31, 1981, ALAB-636, 13 NRC 312, the Appeal Board reversed this Licensing Board's September 12, 1980 determination, LBP-80-25, 12 NRC 355, that §102(2)(C) of the National Environmental Policy Act (NEPA), 42 USC 4332(2)(C), requires the preparation of an environmental impact statement (EIS) covering the impacts of a proposed spent fuel pool expansion for a plant that had been licensed before NEPA.[1]

[1]The EIS we had required would have been considerably more limited than a construction permit or operating license review since it would not have included as an environmental cost either the cost of constructing the facility or the cost of operating the facility to the extent the

We had based our decision on the grounds that the expansion was for the sole purpose of permitting an additional ten-year term of reactor operation that had never been environmentally reviewed because the facility had been licensed before the passage of NEPA and had never had *any* of its term of operation environmentally reviewed. The Appeal Board held that NEPA does not require considering the environmental impacts resulting from a Federal action that merely permits continued reactor operation without any *change* in reactor operation.[2] We cannot, of course, quarrel with the Appeal Board's reversal of our holding.

The Appeal Board also directed the Licensing Board to "rethink," ALAB-636, 13 NRC 330 (1981), and "reconsider," id. 333, its purported further determinations that (1) the Staff would inevitably decline to prepare an EIS and (2) the failure to issue an EIS would be erroneous also because the impacts from the physical expansion of the spent fuel pool (disregarding the continued plant operation that the expansion might afford) themselves necessitate the preparation of an EIS. The Appeal Board discoursed at length about the "chilling effect" our "unwise, if not improper" premature decision would have in "inhibit[ing]" the Staff from doing its job of determining whether an EIS is necessary in an "honest and objective fashion," and would result in compromising the "integrity of the hearing process." Id. 330, 331.

This Board has no little difficulty in rethinking something that it had not thought in the first place and in reconsidering "an inappropriate prejudgment of the staff's position," (*id.* 331), that is not evidenced in our decision. Simply stated, we never decided that the Staff was unalterably committed to not preparing an EIS or that any effect of the proposed spent fuel pool expansion *other than the continued plant operation it would afford* necessitated the preparation of an EIS. We have carefully reexamined our Memorandum and Order, as well as the Appeal Board's decision, in order to locate the source of the confusion. On reflection, it appears to us to stem from our beginning assumption, apparently not shared by the Appeal Board[3], that a *single* Federal action such as the proposed amendment of the license to permit a spent fuel pool expansion, requires and permits the preparation of but a *single* environmental document encompassing *all* of the impacts of that action: an environmental impact statement, if the action is major and has a significant effect upon the human environment;

operation would not be directly facilitated by the spent fuel pool expansion. LBP-80-25, *supra*, 12 NRC 365.

[2]The Appeal Board disclaimed any reliance upon the prohibition against a retroactive application of NEPA for its decision. ALAB-636, 13 NRC 329 (1981) fn. 32. Such reliance could have served to distinguish this situation from a license renewal application.

[3]See ALAB-636, *supra*, p. 329-330.

an environmental impact appraisal (EIA), accompanied by a negative declaration, if otherwise. §102 of NEPA, 42 USC 4332, *supra,;* 10 CFR §§51.5, 51.7. We had never considered that a single major action having a number of environmental impacts would require or even permit the preparation of separate environmental reviewing documents covering separate impacts, some which may be minor and some major, with an EIS (or EISs) covering the major impacts and an EIA (or EIAs) covering the minor ones. Hence, once we had determined that one of the effects of the licensing action, *viz.* the continued operation of the facility over a lengthy term, necessitated the preparation of an EIS, we ordered that the document also cover all other environmental impacts of the licensing action, even though those impacts standing alone might not have required an EIS. It was this part of our order (12 NRC 366) which was the apparent source of confusion.

We did not believe that the continued plant operation effect and other effects of the expansion could be viewed as separate Federal *actions* requiring the preparation of separate environmental documents (e.g., 2 EISs, 2 EIAs, or an EIS and EIA).[4] As we read 10 CFR §§51.5 and 51.7, the action to be assessed was the amendment of the license to permit the spent fuel pool expansion. Thus, the environmental review would have considered all of the impacts resulting from that action and, in our view, those impacts included continued plant operation.[5]

From the foregoing discussion, it is clear that the parties can be further reassured that our Memorandum and Order did not also postulate, as suggested by the Appeal Board, ALAB-636, 13 NRC 331 (1981), fn. 36, that an action that otherwise does not have a significant effect on the environment may be transformed into one that does by the absence of an environmental review of a different, prior action. The Appeal Board's suggestion was in the context of our having distinguished the instant proposed spent fuel pool expansion from other spent fuel pool expansions in which EISs were not required.[6] We noted that in those cases there had been prior environmental reviews that need not be duplicated for the spent

[4]Ibid.

[5]We concede that if one were to begin with the assumption (as we did not) that the separate impacts of a single licensing action may require the preparation of separate environmental documents, one could easily misinterpret our decision as requiring an EIS to cover continued plant operation and at least one other EIS to cover all other impacts arising from the spent fuel pool expansion. In fact, all that we determined was that the action facilitating continued plant operation required the preparation of an EIS.

[6]*Commonwealth Edison Company* (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980); *Portland General Electric Company* (Trojan Nuclear Plant), LBP-78-32, 8 NRC 413, 449-50 (1978), *aff'd*, ALAB-531, 9 NRC 263 (1979), *Duquesne Light Company* (Beaver Valley Power Station, Unit 1) LBP-78-16, 7 NRC 811, 816 (1978); *Northern States Power Company* (Prairie Island Nuclear Generating Plant, Units 1 and 2), LBP-77-51, 6 NRC 265, 268 (1977), *aff'd,*

fuel pool expansion. However, as we thought was apparent in our opinion, those prior environmental reviews were prepared at the operating license stage and covered the impacts from the operation of the reactor that we recognized should not have to be duplicated in the spent fuel pool expansion proceeding. Since none of the other impacts of the spent fuel pool expansions in those cases could have been covered in the environmental review at the operating license proceeding, the only portion of the review of the proposed spent fuel pool expansion that could have duplicated the prior environmental review was that regarding the continued plant operation. The impacts of the change in fuel pool itself have been deemed negligible in all cases we have discovered. We do not *disagree* with the findings in those cases: we thought we had carefully *distinguished* them from the case at bar.

In sum, we need not await the preparation of the Staff's environmental analysis as suggested by the Appeal Board, *id.* 330, 333, to affirm to the parties that our September 12, 1980 Memorandum and Order did not *in any measure* prejudge the issue of whether the effects of expanding the spent fuel pool, *per se,* necessitate the preparation of an EIS.[7] If one purpose of the Appeal Board's discussion is to facilitate the Staff's making its analysis in "an honest and objective fashion," we have concluded that we would be better advised to clarify our position *before* the Staff makes its assessment. The Staff should be reassured, regardless of the outcome of Commission review, that it can proceed totally uninhibited by our September 12, 1980 Order, which was directed solely towards the question of whether the additional term of operation that the expansion would permit necessitated the preparation of an EIS,[8] a matter on which the Appeal Board has now spoken. Similarly, in view of the Appeal Board opinion, we take no position at this juncture on whether the other effects of the spent fuel pool expansion require the preparation of more than one environmental document.

ALAB-455, 7 NRC 41 (1978), *remanded on other grounds, sub. nom. State of Minnesota v. NRC,* 602 F 2d 412 D.C. Cir. 1979).

[7]ALAB-636 faulted us for purportedly prejudging the question of whether an EIS is necessitated by the impacts that might result from the spent fuel pool expansion, other than from continued plant operation. However, it decided the continued plant operation issue on the merits, apparently accepting our view that this issue could be considered ripe for determination as a matter of law, notwithstanding that the environmental assessment had not yet issued. See LBP-80-25, *supra,* 12 NRC 364, fn. 2. We concede that a strict adherence to NRC procedures might have required our also delaying this question until after the Staff had spoken. But we were aware that operation of a nuclear plant for some years has heretofore always required an EIS, and we were reluctant to delay such a decision lest the delay result in a shutdown for lack of storage space.

[8]That we had requested to assume only "*arguendo*" the Staff's prospective issuance of an EIA was recognized early in the Appeal Board's decision. ALAB-636, 13 NRC 315 (1981). It was apparently forgotten when we were later seen as promoting this assumption into "an inappropriate prejudgment of the staff's position." *Id.* 331.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Herbert Grossman, Chairman
ADMINISTRATIVE JUDGE

Dr. Oscar H. Paris, Member
ADMINISTRATIVE JUDGE

Frederick J. Shon, Member
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 22nd day of April 1981.

Cite as 13 NRC 382 (1981) LBP-81-10

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges

Marshall E. Miller, Chairman
Gustave A. Linenberger
Dr. Linda W. Little

In the Matter of	**Docket Nos. STN 50-510**
	STN 50-511
GULF STATES UTILITIES COMPANY (Blue Hills Station, Units 1 and 2)	**April 28, 1981**

In response to applicant's request for an early site review in connection with this construction permit proceeding, the Licensing Board issues a partial initial decision on site suitability.

TECHNICAL ISSUES DISCUSSED:

Regional demography — exclusion area, low population zone, population center distance;

Meteorology — design basis tornado, atmospheric dispersion of accidental and routine airborne releases of effluents from nuclear plants;

Hydrology — Texas Water Plan, probable maximum flood, probable maximum precipitation;

Seismology and Geology — tectonic province, safe shutdown earthquake, operating basis earthquake, design response spectra, seismic design criteria, soil liquefaction;

ACRS Review;

Common Defense and Security;

NEPA Requirements and the EIS — Closed cycle cooling system, service water system, discharge of chemical wastes, electrical transmission system, construction impacts, socioeconomic impacts, land use impacts, impacts on species populations, water quality, transportation of radioactive materials, alternate sites.

APPEARANCES OF COUNSEL

Mark J. Wetterhahn, Esq., of Conner, Moore and Corber, Washington, D.C.

Stanley Plettman, Esq., of Orgain, Bell & Tucker, Beaumont, Texas, For the Applicant, Gulf States Utilities Company

Richard Lowerre, Esq., Assistant Attorney General, For the State of Texas

Colleen P. Woodhead, Esq. and **Lawrence Brenner, Esq.**, Office of the Executive Legal Director, U. S. Nuclear Regulatory Commission, Washington, D.C. For the NRC Staff

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PARTIAL INITIAL DECISION

(Early Site Review)

I. INTRODUCTION

This is a proceeding on the application of Gulf States Utilities Company ("Applicant") for construction permits for the proposed Blue Hills Station, Units 1 and 2 (the "facility") to be located in Newton County, Texas. This Partial Initial Decision examines the Applicant's request for Early Site Review, in accordance with NRC's "Early Site Reviews and Limited Work Authorizations", 42 *Fed. Reg.* 22882 (May 5, 1977), effective June 6, 1977.

1. The proposed facility would consist of two pressurized water reactors (PWR), each designated for initial operation at approximately 957 MWe (gross).[1] The proposed site is the Applicant's 1221-ha (3016 acre) site on the lower basin of Toledo Bend Reservoir in eastern Texas (Newton County), approximately 14.5 km (9 miles) west of the Texas-Louisiana border and 40.2 km (25 miles) east-northeast of the city of Jasper, Texas.[2] It is 17 miles east of Sam Rayburn Reservoir. Farm-to-Market Road 255 runs east-west about two miles south of the site.

2. On June 27, 1974, Applicant applied to the Atomic Energy Commission, predecessor to the Nuclear Regulatory Commission (NRC), for construction permits for two pressurized light water reactors for its Blue Hills Station, Units 1 and 2, to be located in Newton County, Texas. The Commission issued a "Notice of Hearing on Application for Construction Permits" on October 29, 1975 (40 *Fed. Reg.* 52768, Nov. 12, 1975, as corrected, 40 *Fed. Reg.* 54031, Nov. 20, 1975). At that time, an Atomic Safety and Licensing Board (Board) was established. Since that time, the Board has been reconstituted on two occasions (41 *Fed. Reg.* 37678, 1976; 43 *Fed. Reg.* 8871, 1978). No petitions for leave to intervene were received in response to the original notice. The State of Texas requested participation in the proceeding as an "interested State" pursuant to 10 CFR Section 2.715(c). This proceeding is an uncontested proceeding as defined by 10 CFR Section 2.4(n).

3. Applicant subsequently amended its application for construction permits to include a request for a partial initial decision on early site review leading to construction permits and operating licenses, in accordance with

[1] Staff's Exhibits 7, 7A, 7B, Final Site Environmental Statement (NUREG-0449), Blue Hills Station, Unit Nos. 1 and 2, July 1978, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. (hereafter FSES), received into evidence at Tr. 169; Applicant's Exhibit 2, Appendix A, at 2).

[2] *Ibid.*

"Early Site Reviews and Limited Work Authorizations" (42 *Fed. Reg.* 22882, May 5, 1977; effective June 6, 1977). Reasons cited by Applicant for this amendment were changes in load forecasts, construction schedule, and a resulting slippage in dates for the proposed facility. On May 3, 1978, the Commission issued a "Notice of Hearing on Application for Early Site Review" (43 *Fed. Reg.* 20572, May 12, 1978). Notices (display advertisements) of the public hearing were published in the *Leesville* (Louisiana) *Leader*, the *Jasper* (Texas) *News Boy*, and the *Newton* (Texas) *News*. No timely petitions for leave to intervene were received in response to this Notice. The State of Texas reiterated its desire to participate as an interested state pursuant to 10 CFR Section 2.715(c), which request was granted by the Board. On November 1, 1978, the Commission's Office of the Secretariat received an undated petition for leave to intervene from D. Michael McCaughan, Member, The Environmental Task Force, 3131 Timmons Lane, Apt. 254, Houston, Texas 77027. Both Applicant and Staff opposed this untimely petition. On December 27, 1978, the Board denied the petition, based on a failure to demonstrate interest or standing to intervene, and after consideration of factors required for late intervention [10 CFR Section 2.714(a)(1)(i-v)].

4. The Board published a "Notice of Hearing on Application for Early Site Review" on April 6, 1979 (44 *Fed. Reg.* 22231). The duly noticed prehearing conference and evidentiary hearing were held May 8-9, 1979, in Jasper, Texas. During the course of the hearing, a number of limited appearance statements were received, pursuant to 10 CFR Section 2.715(a) (Tr. 31-41, 126-59, 258-87). On May 8, 1979, the technical members of the Board toured the proposed site and viewed portions of Toledo Bend Reservoir and other points of interest related to the proposed Station, including proposed intake and discharge sites.

5. Following completion of the evidentiary hearing on suitability of the site, the Commission issued certain siting-related documents which the Board considered potentially applicable to the Blue Hills proceeding. These were:

> "Report of the Siting Policy Task Force," NUREG-0625 (August, 1979)
>
> "Advance Notice of Rulemaking: Revision of Reactor Siting Criteria" (July 23, 1980)
>
> Proposed amendments to emergency planning regulations for production and utilization facilities (44 *Fed. Reg.* 54308, September 19, 1979; 44 *Fed. Reg.* 75167, Dec. 19, 1979)

Final rule on emergency planning (45 *Fed. Reg.* 55402, August 19, 1980)

Statement of Interim Policy on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (June 9, 1980)

"Proposed Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License", NUREG-0718 (August, 1980)

Upon further examination of that part of the evidentiary record (ER at 2.2-49, Suppl. 1; Tr. 259-260) relating to possible transfers of water from the Sabine River and its tributaries and reservoirs to other portions of the State of Texas, the Board *sua sponte* obtained and reviewed the "Texas Water Plan: Summary" (Texas Water Development Board, November, 1968).

On September 9, 1980, the Board issued an "Order Requesting Briefs and Additional Information," requesting from Applicant and Staff statements of position and supplemental information on applicability or potential applicability to the instant early site review of the rules, proposed rules, and other documents described *supra.* Responses were provided by the Staff on October 15, 1980 and by the Applicant on January 23, 1981. The "Order" and the "Responses" are hereby incorporated as supplements to the record of this proceeding.

6. The record of this proceeding consists of the transcript of the prehearing conference of May 8, 1979 (Tr. 1-29), the transcript of the evidentiary sessions of May 8-9, 1979 (Tr. 30-289), the Board order and responses (*supra*), and the exhibits which were received in evidence listed in Appendix A, attached hereto.

7. The parties to this proceeding are the Applicant, the NRC Staff (Staff), and the State of Texas (State). The State participated as an "interested state" pursuant to 10 CFR Section 2.715(c). The State of Louisiana did not participate, although copies of the DES were submitted to the Louisiana Board of Nuclear Energy for notification and review (Tr. 251).

8. Pursuant to the requirements of 10 CFR Section 2.101(a-1), the Applicant has submitted proposed findings on the issues on which it has requested review and a statement of the bases or reasons for those findings and has submitted a range of postulated facility design and operation parameters to enable the requested review of site suitability issues to be performed under the applicable provisions of Parts 50, 51 and 100.

9. The Applicant has also submitted information concerning the Applicant's site selection process and long range plans for ultimate

development of the site. With regard to the ultimate development of the site, the Applicant has stated that the Blue Hills site (which has also been referred to by the Staff as Site G) is ultimately capable of supporting four nuclear power facilities of the general size and type being licensed in the United States. However, the Applicant clearly indicated that such information was submitted to fulfill the requirements of Sections 2.101(a-1)(1) and 2.603, and no finding on the suitability of the Blue Hills site for four units was requested or made herein. The Applicant recognized that should it wish to site any more than two reactors at Blue Hills, it must submit a new application, including a new alternative site study, to the NRC (Tr. 14).

10. Following the conclusion of the evidentiary hearing, the Applicant submitted its Proposed Findings of Fact and Conclusions of Law on June 7, 1979. The Staff's proposed findings and conclusions adopted those proposed by the Applicant, with certain specified exceptions and modifications filed July 5, 1979. In response to the Board's Order Requesting Briefs and Additional Information entered September 9, 1980, responses were provided by the Staff on October 15, 1980, and by the Applicant on January 23, 1981. In those instances in which the proposed findings were found by the Board to be complete, accurate and supported by the record, they have been substantially adopted by the Board. In all other cases, the Board made its own findings or substantially modified the proposed findings submitted by either the Applicant or the Staff, based upon the evidentiary record. Any proposed findings of fact and conclusions of law submitted by the parties which are not incorporated, directly or inferentially, into this Partial Initial Decision are herewith rejected as being unnecessary to the rendering of this decision.

II. FINDINGS OF FACT ON SITE SUITABILITY AND SAFETY MATTERS

11. Applicant submitted an application (Applicant's Exhibit 2) and a Preliminary Safety Analysis Report (PSAR) (Applicant's Exhibit 4) containing detailed technical information relative to site suitability and safety matters for which Applicant has requested early site review, as set forth in Attachment A to the License Application (Applicant's Exhibit 2). The PSAR format adheres to Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 1 (USAEC, October, 1972).

12. The Staff reviewed the PSAR and issued its Early Site Review for the Blue Hills Site ("ESR", NUREG-0131) in January, 1977 (Staff Exhibit 8). Supplement 1 to the Early Site Review for the Blue Hills Site was issued in June, 1977 (Staff Exhibit 9). Staff Exhibits 8 and 9 summarize the results

of the Staff's technical evaluation of the suitability of the proposed Blue Hills site for a nuclear power plant, and delineate the scope of the technical matters relative to the radiological health and safety aspects of the proposed facility.

13. Based on its evaluation, the Staff concluded that the Blue Hills site is acceptable under the guidelines of 10 CFR Part 100 for the construction and operation of a nuclear power plant of the general type and size being proposed for other sites in the United States.

14. The Board finds that the Applicant has provided sufficient information relative to the radiological health and safety of the proposed site, and that the Staff's review and evaluation of that information is adequate.

A. Regional Demography and Land and Water Use

15. The exclusion area (radius of 0.86 mile or 1390 meters) is entirely within the site boundary. No public highways, waterways, or railroads traverse the exclusion area. There are no residences within the exclusion area. The site is totally owned by the Applicant, which will have authority to determine all activities within the exclusion area, as required by 10 CFR Part 100 (PSAR Section 2.1.2 and response 31.22; ESR Section 2.1; Tr. 51). The exclusion area can meet the applicable guidelines of 10 CFR Part 100 and is acceptable. The exclusion area also exceeds the value of 0.5 mile, cited in the Report of the Siting Policy Task Force (NUREG-0625, August, 1979) as a distance which "would provide reasonable assurance...that radiation doses beyond this distance would not result in consequences greater than the present guidelines values given in Part 100.11, assuming that the engineered safety features function as designed."

16. The 1970 population within 10 miles of the site was estimated by Applicant to be about 1500 people; within 50 miles, 155,500 people. These populations are projected to approximately double by the year 2020. Transient population resulting from recreational activities near the Toledo Bend Reservoir occurs between four and five miles and reached a total of 23,000 during 1973. The transient population has been estimated to reach 63,000 by the year 2020 (PSAR Section 2.1.3; ESR Section 2.1). On a demographic basis (1970 data), the Blue Hills site ranks very favorably with other sites licensed by the Commission (Applicant's Response to Board Order, 2-3, citing NUREG-0348, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites" (October, 1979)).

17. The Applicant has specified a low population zone of three miles radius. The population within that area is stated to be 10 for the 1970 census year, and the Applicant estimates no more than 22 by the year 2020.

No characteristics of the low population zone have been identified which would preclude formulation of an acceptable emergency plan for the residents within the zone as required by 10 CFR Part 100.3(b). No other finding was requested by Applicant at this stage. Full findings on emergency planning, including consideration of emergency planning zones of 10 and 50 miles radii (45 *Fed. Reg.* 55402), will be made at the construction permit stage.

18. There are no large communities in the vicinity of the site. The largest unincorporated area within 50 miles is the Fort Polk military base with a population of 24,000 and located 33 miles east of the site. There are no communities within 50 miles with a 1970 population of 25,000 or more. The nearest population center is properly identified and this satisfies the 10 CFR Part 100 requirement that a population center distance be at least one and one-third times the distance from the reactor to the outer boundary of the low population zone (PSAR Section 2.1.3.5; ESR Section 2.1). The low population zone is acceptable.

19. A reactor system and engineered safeguards have not yet been defined for the Blue Hills site, and thus offsite doses from postulated design bases accidents cannot be compared to the guideline values of 10 CFR Part 100. However, based on experience at other licensed power plants and those currently under review, the parameters of the exclusion area and low population zone, and meteorological dispersion factors existing at the site, it can be concluded that the Blue Hills site is acceptable under the guidelines of 10 CFR Part 100 for the construction and operation of a two-unit nuclear power plant of the general type and size being proposed for other sites in the United States (ESR Section 2.1).

20. There are no significant industries, waterways, airports, mining activities, railroads, or military facilities within 10 miles of the Blue Hills site. The nearest major roadway is State Highway 87 which passes, at its closest approach, about two miles west of the site. The nearest pipeline is an eight-inch crude oil line passing about five miles southeast of the site, and the nearest railroad is a line of the Sante Fe Railroad 18 miles west of the site. Federal Airway V212 passes about five miles north of the site (PSAR Section 2.2.1 and 2.2.2; ESR Section 2.2).

21. The nature and extent of activities at nearby industrial, military, and transportation facilities have been evaluated. There are no activities in the vicinity currently going on or presently planned which have the potential for precluding use of this site for a nuclear power plant as outlined in the application (PSAR Section 2.2.3; ER Section 2.2).

B. Meteorology

22. As described below, a sufficient description of the regional meteorological conditions of importance to the safe design and siting of a nuclear power plant at the Blue Hills site has been provided (ESR Section 2.3; Tr. 177-96).

23. Snowfall is a rarity in the region, averaging less than one inch per year. However, occasional storms in the general vicinity accumulated up to 10 inches of snow on the ground. One or two ice storms, some occasionally severe, may occur each year in the area. Similarly, the mean annual number of days of hail in the region is one or two. A design load for roofs of safety-related structures of 30 pounds per square foot as proposed by the Applicant, is acceptable for loads due to snow at the Blue Hills site (PSAR Section 2.3.1; ESR Section 2.3.1).

24. Between 1953 and 1974, 116 tornadoes occurred within a 10,000 square mile area containing the site, resulting in a recurrence interval of 670 years for a tornado at the plant site. The design basis tornado proposed is similar to the design basis tornado parameters for Region I as described in Regulatory Guide 1.76 "Design Basis Tornado for Nuclear Power Plants," and is acceptable for the site. These parameters include a maximum wind speed of 360 miles per hour consisting of a maximum rotational speed of 290 miles per hour and a maximum translational speed of 70 miles per hour; a minimum translational speed of five miles per hour; a radius of maximum rotational speed of 150 feet; a pressure drop of three pounds per square inch; and a rate of pressure drop of two pounds per square inch per second. Hurricanes and tropical storms also effect the site area. Because the site is 95 miles inland from the Gulf of Mexico, the velocities of wind from these storms are less at the site than at the Gulf Coast. An operating basis wind speed (defined as the "fastest mile" with speed at a height of 30 feet with a return of 100 years) of 90 miles per hour is acceptable (PSAR Sections 2.3.2.2.1-wind, and 3.3.2.1; ESR Section 2.3.1).

25. The meteorological data from the region has been examined to select appropriate meteorological conditions in considering the design requirements for an ultimate heat sink as recommended in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The meteorological data presented is acceptable for analysis of the ultimate heat sink design concept (i.e., mechanical draft cooling tower and basin) described in the Preliminary Safety Analysis Report (PSAR Section 9.2.5; ESR Section 2.3.1).

26. Sufficient information has been provided to make an evaluation of the local meteorological conditions of importance to the safe design and siting of a nuclear power plant at the Blue Hills site. Two years of data

collected onsite is available to assess the local meteorological characteristics of the Blue Hills site as well as climatological data from three other locations (ESR Section 2.3.2; Tr. 79).

27. The onsite meteorological measurements program conforms to the recommendations and intent of Regulatory Guide 1.23, "Onsite Meteorological Programs." The meteorological measurements program has produced data which, in turn, have been summarized to provide sufficient meteorological description of the site and its vicinity and serves as an acceptable basis for making atmospheric dispersion estimates for use in determining the radiological consequences of accidental and routine airborn releases of effluents from a nuclear power plant (ESR Section 2.3.3).

28. The Blue Hills site is located in a forested terrain. A meteorological model which considers the "sheltering" effect of the trees surrounding the meteorological tower in calculations of atmospheric dispersion factors (X/Q's) for the site was originally proposed (PSAR Section 2.3.4.2 and Appendix 2G). These X/Q's are smaller than those calculated which do not consider the "tree sheltering" effect. As a result of the Staff evaluation of the Applicant's meteorological model, the Staff found that the quantitative reduction of the X/Q's proposed by the Applicant due to the "tree sheltering" effect was not warranted based upon the information available on this phenomenon at this time. The Staff therefore did not utilize the sheltering effect in its development of acceptable X/Q estimates for the Blue Hills site (ESR Section 2.3.4). If further data become available, and if the Applicant at its election proposes it, the Staff will again consider the modifications of its meteorological model to take into account this phenomenon.

29. In calculations of short-term dispersion estimates, a dispersion model modified from that described in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" was used by the Staff. This modified model has incorporated results from recent field experiments in atmospheric dispersion (ESR Section 2.3.4; Tr. 229-35).

30. Using the modified dispersion model, which considers directionally variable exclusion boundary distances and site specific directional frequencies of atmospheric dispersion conditions, conservative assessments of post-accident atmospheric dispersion conditions have been made for the Blue Hills site by the Staff. In the model, meteorological data for two years of onsite data collection with wind direction and speed measured at the 33-foot level were used (ESR Section 2.3.4; Tr. 230-34).

31. The relative concentration for the 0-2 hour time period which is exceeded no more than five percent of the time is 1.1×10^{-3} seconds per

cubic meter at an exclusion distance of 1,369 meters measured from the outside edge of the containment buildings (ESR Section 2.3.4).

32. The relative concentration values for various time periods at the outer boundary of a Low Population Zone of 4,800 meters, calculated on a conservative basis, are a X/Q of 1.7×10^{-4} sec/m^3 for 0-8 hours, a X/Q of 1.2×10^{-4} sec/m^3 8-24 hours, a X/Q of 4.8×10^{-5} sec/m^3 for 1-4 days, and a X/Q of 1.4×10^{-5} sec/m^3 4-30 days (ESR Section 2.3.4).

33. Average atmospheric dispersion conditions for the Blue Hills site were estimated using an atmospheric dispersion model for long-term releases based on the "Straight-Line Trajectory Model" described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." The model assumed a ground-level release only and considered the effects of airflow recirculation and stagnation. Neglecting plume depletion and radioactive decay, the highest offsite annual average relative concentration of 4.1×10^{-5} seconds per cubic meter would occur at the east boundary 1,369 meters from the reactor complex (ESR Section 2.3.5).

34. Sufficient information concerning those meteorological conditions which are of importance to the safe design and siting of a nuclear power plant at the Blue Hills site has been provided. The design basis tornado parameters proposed for the site conform to the provisions of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The Applicant's onsite meteorological program conforms to the provisions of Regulatory Guide 1.23, "Onsite Meteorological Program," and has produced two years (October 15, 1973 - October 14, 1975) of onsite meteorological data which provide an acceptable basis to determine site atmospheric dispersion conditions and which were used to make both conservative and realistic estimates of atmospheric dispersion characteristics for accidental and routine gaseous releases, respectively, for the Blue Hills site (ESR Section 2.3.6).

35. In response to a question raised at the hearing, the Board explored the meteorological dispersion characteristics at distances well beyond the low population zone. At a distance of 50 miles, the annual average X/Q is approximately two orders of magnitude lower than at the low population zone, i.e., the dispersion is two orders of magnitude higher (Tr. 86). The Board considers that the question was adequately addressed (Tr. 34-5, 75-87, 186, 228-40). In any event, 10 CFR Part 100 guidelines and other NRC requirements are met.

C. Hydrology

36. The Blue Hills site is located in the Mill Creek basin eight miles west southwest of the Toledo Bend Dam. The lower portion of the Toledo Bend Reservoir is between the site and the dam. When the water level is at the top of the spillway gates, the closest point of the reservoir is just over one mile from the site. The site lies on a ridge between two small creeks, Copperas and Mitchell Creeks, which are approximately a mile apart at the site. The proposed plant grade is 270 feet above mean sea level; 97 feet above the top of the dam spillway gates, more than 50 feet above the higher creek bed (Mitchell Creek) near the site (PSAR Section 2.4.1.2; ESR Section 2.4.1).

37. Toledo Bend Reservoir is located on the Sabine River at river mile 156.5, where the drainage area is 7,178 square miles. The top of the dam is 185 feet above mean sea level, the top of the power pool (that portion of the reservoir used for hydroelectric power generation) is 172 feet above mean sea level, and the tops of the gates are 173 feet above mean sea level. At elevation 172 feet above mean sea level, the reservoir covers 182,000 acres and contains almost 4.7 million acre-feet of water. Water from the reservoir is used for irrigation, municipal and industrial water supplies, hydroelectric power generation and recreation. The water supply for normal plant operation would be obtained from the Toledo Bend Reservoir (PSAR Section 2.4.1.2; ESR Section 2.4.1). In view of the uncertainty of the probability, timing, and manner of implementation of the Texas Water Plan (5, *supra*) (Board Order at 5-6 and Staff's Response at 10-11), the Board finds that prior to issuance of any limited work authorization or construction permit, the current status of this Plan in regard to the Toledo Bend Water Reservoir must be examined.

38. The probable maximum flood elevation calculated by the Applicant for the Mill Creek basin using a conservative methodology is estimated to be 243 feet above mean sea level near the site; this is well below plant grade of 270 feet above mean sea level. Because of this large freeboard, the probable maximum flood does not constitute a threat to the Blue Hills site. Since no dams exist in the Mill Creek basin, the Blue Hills site is not susceptible to a dam failure flood. Surges and seiches on Toledo Bend Reservoir will not affect the site because it is more than a mile away and almost 100 feet above the normal reservoir water level. There is no other large water body near the site. Due to its inland location, the Blue Hills site is not susceptible to tsunami flooding. Relatively mild winters in the site area preclude the possibility of ice flooding and associated damage to safety-related facilities (PSAR Sections 2.4.3, 2.4.4, 2.4.5, 2.4.6 and 2.4.7; ESR Section 2.4.2 and 2.4.3).

39. An ultimate heat sink of the general type proposed by the Applicant (mechanical draft cooling towers and basins) could be designed to safely shut down and to maintain a nuclear power plant in safe shutdown for at least 30 days in the event of the loss of water to the plant from the Toledo Bend Reservoir (PSAR Sections 2.4.11.6, 9.2.5.3; ESR Section 2.4.4; Tr. 119). A specific ultimate heat sink design will be reviewed at the construction permit stage.

40. The Applicant has stated that the roofs of all safety-related buildings and the site grading and drainage will be designed to prevent a threat to safety-related facilities by the localized probable maximum precipitation (PSAR Section 2.4.10; ESR Section 2.4.5).

41. An analysis of an accidental spill of liquid radioactive wastes was provided. A postulated failure of a boron management system holdup tank releasing approximately 124,000 gallons to the groundwater was evaluated. The analysis showed that all radionuclides will be below the maximum permissible concentration listed in 10 CFR Part 20 Appendix B at the point where Mitchell Creek leaves the site exclusion area. In addition, there is no present or projected future use of any of the surface waters in the Mill Creek basin. There is little likelihood of contamination of potable water supplies outside the site exclusion area from an accidental release of liquid effluents (PSAR Section 2.4.12; ESR Section 2.4.7).

42. The site is located in sediments of the Gulf Coastal Plain, which contain large quantities of water commonly occurring under confined conditions. The permeable sands containing the groundwater are interbedded with less permeable clays, silts and silty clays which act to confine the water in the sands. Groundwater beneath the site occurs in two zones. A perched water table, within 20 feet of the surface, is present above localized lenticular clay interbeds. The main water zone is at a depth of 70 to 80 feet below the site. Recharge is by percolation of water flowing around the overlying lenticular clay bodies and by infiltration from Copperas Creek. Groundwater movement is to the northeast apparently toward Toledo Bend Reservoir (PSAR Sections 2.4.13.1.2, 2.4.12.2; ESR Section 2.4.8).

43. Nearly all the wells within 10 miles of the site extract less than 10 gallons per minute. There are no wells downgradient of the plant between the site and Toledo Bend Reservoir. The Applicant states that there are no present plans to use groundwater for plant operation; all the water used will come from Toledo Bend Reservoir. Groundwater levels at the site are at elevations ranging from 190 to 210 feet above mean sea level, excluding the perched water tables. There is little likelihood of contamination of potable water supplies outside of the site exclusion area from an accidental release of radioactive liquid effluents (PSAR Section 2.4.12.5; ESR Sections 2.4.8, 2.4.9).

44. Based on evaluation of the present groundwater levels, topography at the site and the removal of the higher perched water table during construction, the proposed design basis groundwater level of 215 feet above mean sea level is conservative and acceptable for use in the design of a nuclear power plant at the Blue Hills site (ESR Section 2.4.8).

45. The flood analysis for the Blue Hills site meets the criteria in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and flooding does not constitute a threat to the site (ESR Section 2.4.9).

D. Seismology and Geology

46. The seismology and geology review of this site addressed the geologic history of the region including physiographic, lithologic, stratigraphic and tectonic settings as well as the subregional and site-specific geology and seismology. Investigations have been sufficient to adequately assess site geologic conditions in accordance with "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Appendix A, to 10 CFR Part 100.

47. The tectonic province approach, as described in 10 CFR Part 100 Appendix A, was followed to determine the vibratory ground motion corresponding to the safe shutdown earthquake. The selected safe shutdown earthquake acceleration of 0.13g represents an appropriate and conservative reference acceleration for seismic design of structures at the Blue Hills site (PSAR Section 2.5.2.10; ESR Section 2.5).

48. The site is located within the Gulf Coastal Plain physiographic province which is the onshore portion of the Gulf Coast Geosyncline which extends under the Gulf of Mexico to the edge of the continental shelf. The sedimentary deposits in the region range in age from Jurassic to Recent and consist mainly of unconsolidated sands, silts, clays, limestone and chalk with minor amounts of salt. The sediments form a wedge that diverges seaward, exceeding 50,000 feet in total thickness. At least 20,000 feet of sediments underlie the Blue Hills site. Due to consolidation of the thick sedimentary section, the general dip of the strata increases gulfward at slightly greater angles than the present land surface. Differences in resistance to erosion of the sediments resulted in a series of linear topographic belts which are parallel to the Gulf Coastline. The more resistant formations form landward facing cuestas with relief up to 400 feet or more. Salt domes which are common to the east Texas region are not known to occur closer than approximately 55 miles from the site (PSAR Sections 2.5.1.1.4.3.5 and 2.5.1.1.6.6; ESR 2.5.1).

49. In the site vicinity, there may be faults (none is known to exist within a five mile radius of the site) of nontectonic origin characterized by steep, near surface dips which become less steep with depth and eventually

pass into bedding planes. Another characteristic of these faults is the thicker strata on the downthrown side, where accumulation occurred simultaneously with fault movement. They are referred to as growth faults and are predominantly of low stress, since they are shallow rooted. They typically do not develop large strain and sudden stress releases which are characteristic of damaging earthquakes, and therefore, are not considered to present a hazard to the proposed site (PSAR Section 2.5, 2.5.1.1.4.3.3; ESR Section 2.5.2).

50. There are no geologic faults or other tectonic structures that present a potential hazard to the proposed site (PSAR Sections 2.5.2.2, 2.5.2.8; ESR Section 2.5.2).

51. The Blue Hills site is located in the eastern part of the West Gulf Coastal Plain. The Mississippi Alluvial Plain divides the Gulf Coastal Plain province into east and west segments. As a result of a comprehensive investigatory program, it was concluded that no deformational zones, such as folds, fissures, slips, faults and shears, have been found at the site and the nearest known salt dome is approximately 55 miles south of the site. In addition, no oil, gas, or other mineral extraction has been or is presently being conducted with a five-mile radius of the site, and groundwater extraction in the vicinity of the site is not sufficient to cause subsurface subsidence. Also, there is no record of subsurface mining or other similar underground workings in the area which might create a subsidence problem at the site. All lineaments recognized in a ten-mile radius of the site on small-scale infrared and large-scale panchromatic photography were investigated in the field and no indication of fault offset was observed (PSAR Sections 2.5.1.1, 2.5.3).

52. There are no geologic structures or conditions resulting from man's activities, such as mining or oil extraction, that present a hazard to the site. In addition, the problem of subsidence is not a factor at the Blue Hills site (ESR Section 2.5.3).

53. A conservative value of 0.13g is proposed for the safe shutdown earthquake acceleration level. The intensity corresponding to a mean acceleration of 0.13g is VII (MM). Based on a detailed review of the tectonic province, earthquake aclinity and geologic structures surrounding the site, earthquakes as large as this have not been observed in the historical record of seismicity for the Gulf Coastal Plain, except in the area of the Southern Cordilleran Front, the complex region at the intersection of the Ouachita Tectonic Belt, the Wichita Structural System, and the northern Mississippi Embayment. Neither the high seismicity nor the structural complexity found in these areas where large earthquakes have occurred is present in the vicinity of the Blue Hills site. For the safe shutdown earthquake, 0.13g represents an appropriate and conservative reference

acceleration for seismic design of structures at the Blue Hills site. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," response spectra scaled to this maximum acceleration for the design of a nuclear power plant will be used at the Blue Hills site and this is acceptable (PSAR Sections 2.5.2.10, 3.7.1.1; ESR Sections 2.5.4, 2.5.5, 2.5.6).

54. It is proposed to use 0.07g for the acceleration level corresponding to the operating basis earthquake, which is representative of intensity VI (MM). Considering the low seismicity of the Gulf Coast Seismic Zone, the proposed operating basis earthquake is conservative. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," response spectra scaled to this maximum acceleration of 0.07g for the operating basis earthquake will be used for the design of a nuclear power plant at the Blue Hills site and this is acceptable (PSAR Sections 2.5.2.11, 3.7.1.1; ESR Section 2.5.7).

55. The geologic investigations and the laboratory analyses performed on the soil specimens, including determinations of shear strength, consolidation, dynamic properties, and seismic resistance to earthquake effects are adequate to serve as the basis for the design of safety-related plant structures (ESR Section 2.5.8).

56. The plan for the support of safety-related structures is uncomplicated and acceptable. Upper clay and upper sand strata will be excavated. Deep plant foundations will rest directly on or in the middle sand stratum, i.e., the third sequence. Shallower plant foundations will rest on compacted granular backfill supported by the middle sand stratum. The proposed foundation design is based on an envelope of dimensions, structure depths, loadings, and stated assumptions. Therefore, at the construction permit application stage, the Applicant will validate the applicability of the foundation design to the specific nuclear power plant design proposed as follows: The Applicant will submit for NRC review and approval of its criteria for construction control during (a) excavation and backfilling of the foundations, (b) remedial foundation treatment, (c) proofrolling of the foundation, and (d) removal of unsuitable materials from the middle sand stratum. Standard Penetration Test data in the middle sand strata will be provided for review as comparative plots of blowcount and effective pressure (ESR Section 2.5.9).

57. The large mat foundations supporting plant structures impose relatively low net bearing pressures on the structural fill and soils of the middle sand stratum. Table 2C-3 of the Preliminary Safety Analysis Report indicates that net dynamic bearing pressures due to the safe shutdown earthquake are also relatively low, and that the site soils have adequate bearing capacity (ESR Section 2.5.10).

58. Criteria for the lateral earth pressure acting on subsurface foundations have been established. The proposed design criteria for lateral earth pressures described in the PSAR are acceptable (PSAR Section 2.5.4.10.2; ESR Section 2.5.11).

59. The liquefaction potential of the middle sand stratum was analytically evaluated by comparing the computed dynamic stresses induced in the site soils by the safe shutdown earthquake to the resistance of these same soils to cyclic stresses during tests in the laboratory. The assumptions used in the analysis are conservative, the margins of safety for the various conditions are adequate, and risk of liquefaction due to seismic effects is remote at the Blue Hills site (PSAR Section 2.5.4.8.4; ESR Section 2.5.12). Based on the field and laboratory tests conducted, the dynamic properties of the soils used in the analysis are reasonable for this site and are acceptable (ESR Section 2.5.13).

60. Stability analyses for permanent slopes surrounding the proposed plant area have been performed. None of the slopes is, itself, seismic Category I. All slopes will be constructed at two horizontal to one vertical. The location of these slopes with respect to the proposed location of the safety-related structures is such that slope failures would not endanger these structures. Slope stability considerations at the site are acceptable (PSAR Section 2.5.5; ESR Section 2.5.14).

E. Review by the Advisory Committee on Reactor Safeguards

61. The Advisory Committee on Reactor Safeguards (ACRS) completed its review of the request of the Gulf States Utilities Company to determine the suitability of the Blue Hills site for a nuclear power plant at its 203rd meeting on March 10-12, 1977, in Washington, D. C. Members of the ACRS Subcommittee visited the site on January 28, 1977, and a Subcommittee meeting was held the same day in Jasper, Texas. The ACRS report for the Blue Hills Early Site Review is dated March 16, 1977. The report concludes that subject to the comments and recommendations referenced in the report, the ACRS believes that adequate information is available to conclude that the Blue Hills site is suitable for a light water reactor nuclear power plant of the general type and size currently being proposed for other sites in the United States (Early Site Review, NUREG-0131, Supplement 1, Section 18.0, June, 1977).

F. Common Defense and Security

62. The activities to be conducted under the construction permit will be within the jurisdiction of the United States. All of the Applicant's directors

and principal officers are citizens of the United States, and the Applicant is not owned, dominated or controlled by any alien, foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the Applicant has agreed to safeguard any such data which might become involved in accordance with the requirement of 10 CFR Part 50. The Applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved (Applicant's Exhibit 2 at 4-7). Issuance of construction permits for the Blue Hills Units Nos. 1 and 2 will not be inimical to the common defense and security

G. National Environmental Policy Act Requirements and the Environmental Impact Statement

63. As required by 10 CFR Part 51, the Applicant submitted an Environmental Report. The Environmental Report, as amended, was received into evidence as Applicant's Exhibit 3. Pursuant to the requirements of the National Environmental Policy Act of 1969 and based on the environmental information submitted by the Applicant in the Environmental Report, as supplemented, (Applicant's Exhibit 3) and on its independent analysis and review, the Staff prepared a Draft Environmental Statement (DES) which was issued in June, 1977. By a Notice of Availability published June 9, 1977, the public was invited to comment on the DES (42 *Fed. Reg.* 29571). Copies of the DES were also provided to appropriate Federal, Texas and Louisiana and local agencies for their comments (FSES at iv; Tr. 249-51). In July, 1978, the Staff published its Final Site Environmental Statement (FSES) (43 *Fed. Reg.* 31997 (July 24, 1978)) which includes, among other things, the full text of all comments received with respect to the DES (Appendix A) as well as the Staff's responses to those comments (Chapter 11). The FSES was received into evidence as Staff Exhibit 7, 7A and 7B. In the preparation of its environmental impact statement, the Staff had discussions with a number of Louisiana and Texas state, local and regional officials (FSES Section 1.2).

64. The FSES, as amended by the record of this proceeding, fully describes, as necessary to the Applicant's requested findings, the plant site, certain major systems of the proposed facility, the environmental effects of site preparation, plant and transmission line construction, certain of the environmental effects of plant operation, the Applicant's preconstruction environmental monitoring program, alternative site and subsystem considerations.

65. The Staff concluded on the basis of its analysis and evaluation, set forth in the FSES, including the consideration of alternatives that, subject

to certain conditions for the protection of the environment, site G (Blue Hills) is a suitable location for a nuclear station of the general size and type described in the Applicant's environmental report and the environmental statement (FSES at v). The Applicant has agreed to supply the additional information and abide by the environmental conditions contained in Paragraph 7 of the Summary and Conclusions and Sections 4.5.1, 4.5.2 and 10.5 of the FSES. The Board, on the basis of its consideration of the entire record, concurs that these are appropriate conditions to be imposed in the Partial Initial Decision. Further, the Board finds that the FSES as supplemented and corrected by the testimony and evidence presented in this proceeding, is a comprehensive and adequate review and evaluation of the environmental impacts associated with the Applicant's proposed findings relating to plant construction and operation.

66. The site has been adequately investigated and described, including current geology, hydrology, meteorology, terrestrial ecology, aquatic ecology, water use, regional demography, community characteristics, its economy and historical and archaeological sites, and national landmarks and land use of the site of the Blue Hills Station and the surrounding area, including road, rail, transmission and water supply corridors (ER).

67. The plant cooling systems will operate on a closed cycle basis, utilizing round mechanical-draft cooling towers. Through buried pipelines, the Toledo Bend Reservoir will provide the source of makeup water and other water usage for plant operation. Similarly, buried pipelines will be used to discharge all plant effluents to the Toledo Bend Reservoir (ER Section 3.4.2.1).

68. The nuclear service water system will consist of a water storage reservoir, cooling towers, and other equipment necessary to dissipate all residual and excess heat from the reactor and associated equipment. A circulating water blowdown will be maintained to prevent excessive salt buildup and scaling in the circulated water systems (ER Section 3.4.1.1; FSES Sections 3.4.2, 3.4.3).

69. The proposed intake site is on a point of sparsely vegetated land extending into the Texas side of the reservoir approximately 2.4 km (1.5 miles) upstream of the Toledo Bend Reservoir Dam (Tr. 94-7, 222). Makeup water will be pumped from this location to the plant site (approximately 11.9 km or 7.4 miles) through underground pipes. Cooling tower blowdown from the circulating water system and other plant systems will be discharged into a discharge system collection sump and then pumped to the Toledo Bend Reservoir through underground pipelines. To the extent possible, the discharge pipelines will share the same right-of-way as the intake water pipes (ER Sections 3.4.3.2, 3.4.3.5, 3.4.4; FSES Sections 3.4.4, 3.4.5).

70. State-of-the-art technology exists and equipment is available such that light-water-cooled nuclear power reactors of the general types being proposed and licensed, can be designed to provide effluents which meet the dose design objectives set forth in 10 CFR 50, Appendix I. Compliance with Appendix I will be considered at the construction permit stage (FSES Section 3.5).

71. The construction and operation of the Blue Hills Station will result in the discharge of chemical wastes to the Toledo Bend Reservoir. The chemical wastes result from (1) the concentrating effect on the dissolved solids in the intake water because of cooling tower evaporation and subsequent blowdown, (2) the addition of chemicals to the various systems during operation, which are eventually released at a controlled rate into the effluent stream, and (3) construction wastes. During operation, all waste water from the station, including cooling tower blowdown, will be directed to the discharge system collection sump. After being monitored for pH, conductivity, temperature, and chlorine level, the waste water will be discharged to Toledo Bend Reservoir (ER, Fig. 3.6-1). The discharge from the facility can be carried out by the proposed system in compliance with all applicable state and Federal regulations on the discharge of chemicals, oil and other wastes (ER Section 3.6.2.1; FSES Section 3.6.1; Tr. 100-3).

72. Makeup water for the cooling towers will be supplied from the Toledo Bend Reservoir and the blowdown will be discharged to the Reservoir. Sulfuric acid will be added to the circulating water to control bicarbonate alkalinity and prevent scale formation. To control biological growth in the circulating water system, chlorine will be added periodically. Total residual chlorine will be monitored and the system designed so that discharge to Toledo Basin Reservoir can be limited to 0.2 mg/liter total residual chlorine; actual limits will be set by the cognizant regulatory authority having jurisdiction over such releases (ER Sections 3.6.2.2, 3.6.2.3; FSES Section 3.6.1.2; Tr. 100-3).

73. A sewage treatment plant will be installed in the early construction stage. The basic treatment plant will be supplemented with temporary facilities to handle any excess flow. The treated effluent from the plant will be discharged into a leach field during construction and startup of the Blue Hills Station Unit 1. During operation, the treated effluent will be discharged into the plant outfall. The treated effluent from this plant will comply with applicable discharge standards (ER Section 3.7.1; FSES Section 3.6.2.1).

74. The two diesel generators will provide a standby power source for each unit and will be tested at least monthly. The pollutant levels resulting from this source will meet the applicable standards. Solid waste, other than radioactive, will be disposed of offsite by a commercial contractor or onsite

by methods that meet all local and state standards (ER Sections 3.7.5, 3.7.6; FSES Sections 3.6.2.2, 3.6.2.3).

75. The electrical transmission system proposed for the Blue Hills Station includes approximately 200 miles of 500-kv lines (ER Section 3.9; Tr. 91). To provide power for construction, about 11 km (6.7 miles) of the 500-kv line will have underbuilt provisions for two 138/230-kv lines. Three individual routes are proposed by the Applicant to incorporate the Blue Hills Station power into the existing electrical network. The routes are fully described in the ER. Two of the routes will terminate at substations and the third will tie in with an existing 500-kv transmission system. Most of the land (i.e., approximately 90%) traversed by the transmission routes is currently commercial forest, and approximately 52% of the proposed lines parallel existing rights-of-way. All lines will originate at the station switchyard within the property boundaries (ER Section 3.9.1; FSES Section 3.7).

76. An approximately 20-mile railroad spur to connect the Blue Hills Station with the nearby Atchison, Topeka and Sante Fe railroad (ER, Fig. 2.1-3) is proposed. This spur extends north then west from the site, across generally undulating terrain that is primarily forest with only a small amount of pasture (ER Section 10.10.1.1; FSES Section 3.8.1).

77. The proposed makeup and discharge pipeline runs easterly from the site for approximately 8.5 miles to the intake and discharge locations on the Toledo Bend Reservoir (ER, Appendix F, Fig. II.4:1). The corridor requires approximately 170 acres of land, including approximately 15 acres within the property boundary. Forests are primarily upland types with a variable mixture of pines and hardwoods. Most of the area has been or is scheduled for logging (ER Section 10.2.6.26; FSES Section 3.8.2).

78. The proposed two-lane asphalt concrete access road extends north from FM 255 for approximately three miles to the site (ER, Appendix F, Fig. II.3:2). About one-third of the road is within Gulf States' property. Construction of the right-of-way will require approximately 40 acres of land, but only about 37 acres of construction clearing because of the overlap with the existing road (ER Section 10.10.2.1; FSES Section 3.8.3).

79. Site preparation will involve clearing of the land. Marketable timber will be removed and the remaining trees and brush will be used for erosion control or will be burned in accordance with state and local regulations. That which cannot be burned will be buried in designated areas. During construction, soil will be excavated and used for site fill. Dust resulting from construction activities will be controlled by water trucks, sprinkler systems or chemicals and these measures will adequately minimize this impact. Herbicides will be used to restrict the regrowth of vegetation on shelled and paved roads. Pesticides, if used, will meet appropriate state

requirements. Noise resulting from site preparation and construction will be within acceptable ranges and noise impacts will not be significant. Because of the densely forested characteristics of the site area and the remoteness of the site, visual impact will be negligible. Construction of the railroad spur, access roads and water intake and discharge structures and pipelines and transmission lines will likewise require permanent commitments of land and require clearing of the rights-of-way (ER Section 4.1; FSES Section 4.1).

80. No natural landmarks listed in the Federal Register are within five miles of the proposed site. The proposed plant site has no known major archaeological significance; however, four archaeological localities were identified by the Applicant. The Applicant has stated that an archaeologist will be available for consultation through the construction period should any additional archaeological discoveries be made. Conditions for preservation of the four localities and any future archaeological sites are presented in FSES Section 4.5.2 (ER Section 2.3, FSES Sections 2.9.1, 2.9.2, 2.9.3, 4.5.2).

81. The transmission system proposed for the Blue Hills Station includes approximately 200 miles of 500-kv transmission lines connecting the power plant with the Nona and Rivtrin substations and with the Gulf States Line 559. Rights-of-way for these transmission lines will require about 4,300 acres of land. About 90% of the total length is through forested land, 7% is through pasture land, and the remainder includes transportation and water crossings and residential and recreational land. Land currently used for grazing, farming and recreation will only be temporarily affected by construction activities and will remain available for such use after construction. No herbicides or pesticides will be used in clearing vegetation. Cleared forest will represent a loss in annual timber production of approximately 400,000 ft^3/year of pine wood. Approximately 52% of the total length of proposed routes parallel existing railroad, pipeline or transmission line routes. Because of existing rural roads, no new access roads will be required. The Blue Hills-Nona transmission route crosses about 0.8 km (0.5 mile) of the Big Thicket National Preserve near the Jack Gore Baygall Unit. However, by paralleling an existing pipeline right-of-way, the impact will be minimal. No historical or archaeological sites will be significantly affected by the proposed rights-of-way (ER Sections 3.9.1, 4.2, 4.2.1.2, 4.2.3.3; FSES Section 4.1.3). At the time of the construction permit application, results obtained from the surveys of the proposed transmission line routes to determine the presence of any proposed or nominated endangered species or threatened plant species or habitat critical to their existence will be submitted (FSES Sections 4.3.1.2, 10.5).

82. The principal construction impacts on surface water and groundwater will be those associated with construction of the intake and discharge

structures and with relocation of an unnamed tributary of Mitchell Creek. Construction of the proposed makeup and discharge structures will necessitate the disturbance of approximately 1,000 ft. of shoreline and the removal of an estimated 50,000 yd^3 of material, of which approximately 82% will be dredgings. The Applicant has stated that water used for construction of the main power plant will be provided by a well field consisting of three wells, each with a 200 gpm capacity. The Applicant has stated that only one well will be used to meet normal construction requirements, and the three will be available for the emergency fire protection supply. Dewatering of groundwater seepage during excavation will be minimal because the deepest point of the proposed excavation will be approximately 15 feet above the water table. Construction of the plant and associated onsite facilities (excluding transmission corridors) will involve clearing about 366 acres of forested land and some erosion will be unavoidable. Because of past land-use practices, the nature of the soils, rough topography and the drainage pattern, strict control procedures will be necessary to minimize erosion. The Applicant has stated that a detailed erosion control program will be submitted for Staff approval prior to or at the time that application for construction permits is made (ER Sections 4.1.1.2, 4.1.2.2; FSES Sections 4.2, 4.3.1.1, 4.5.2).

83. The transmission lines will have the greatest visual intrusion where they cross residential or recreational areas; however, these effects and others, e.g., those due to noise and avian mortality, are expected to be small and acceptable (ER Section 3.9.8; FSES Section 5.5.1.2).

84. The range of socioeconomic impacts of construction has been identified (Tr. 60-2, 98-100, 196-214, 218-9) and is adequate to permit anticipatory planning by the affected areas. The Applicant has agreed to begin early planning discussions with local officials and regional planners to discuss methods of limiting and adverse impacts that may occur as a result of plant construction. The Applicant shall submit for NRC review a report of the results of these discussions at least six months prior to the time that application for construction permits is made and at that time transmit copies of such report to the affected governmental agencies and regional planning agencies. This report shall contain a statement of the Applicant's position with respect to the following: planning and mitigation funds, provisions for planning expertise, mobile home zoning ordinances, prepayment of taxes and as to making portions of the site available for public use (ER Sections 8.1, 8.2, Appendix E; FSES Sections 4.4, 4.5.2).

85. Present land use on the site is primarily forest production (FSES 4-1). About 148 ha (366 acres), or 12% of the 1220 ha (3,016 acres) site will be altered from their present use (i.e., timber management) by site preparation and onsite corridor construction (excluding transmission corridors). Of this,

approximately 50 ha (123 acres) will be permanently committed during the lifetime of the plant (FSES 4-1). Since this acreage represents only a minute fraction of the available forest land in this region, removal of the designated land will not have a significant impact on local or regional land-use patterns (ER Table 4.1-1; FSES Section 5.1.1).

86. Drift resulting from operation of the mechanical-draft cooling towers contains dissolved and suspended materials that will be deposited on the landscape in a pattern dependent upon the prevailing meteorological conditions. Land-use impacts from this drift deposition on vegetation are expected to be minimal. No additional ground-level fogging or icing will result from the cooling tower operation (FSES Sections 5.1.1, 5.3.1.2 and 5.5.1.1). During certain weather conditions, the cooling tower plume will be visible for several kilometers. The nearest airports, located 17 miles south and 10 miles west southwest of the site, are not expected to be adversely affected by the plumes (ER Section 5.1.7; FSES Section 5.1.1).

87. Operation of the proposed electrical transmission system will require the periodic maintenance of approximately 200 miles of 500-kv transmission line rights-of-way. Existing rights-of-way will be paralleled for 52% of the total length. The approximately 4,300 acres of new land required is presently about 91% forested and will be replaced and maintained in a grass, herbaceous and woody shrub stage by a three- to five-year mowing cycle. The amount and use of land is not expected to significantly affect overall land-use in the area. Grazing, farming and recreational land crossed by the transmission lines will remain available for their respective uses (ER Section 3.9.8, 5.6; FSES Section 5.1.2).

88. All rare and endangered species are available externally to the site and their populations are not expected to be significantly affected by construction and operation (Tr. 64-6). A comprehensive forest management program including consideration of the red-cockaded woodpecker habitat will be furnished for the site at or prior to the time that application for construction permits is made (Tr. 93). Construction activity on the transmission lines, access road, railroad spur and water pipelines will be monitored to ensure that the effects of construction on the red-cockaded woodpecker are considered. The route of the railroad spur will be adjusted to minimize, to the extent practicable, impact to bog areas. Overall, the impacts on species populations from the reduction in forest habitat caused by construction are expected to be minimal (FSES Sections 4.3.1.1, 4.3.1.2).

89. Loss of reservoir water resulting from evaporation and drift losses from the proposed mechanical-draft towers is not expected to affect any other reservoir-water usage. Although there will be chemical discharges, the discharges from the station to the reservoir will not significantly affect any present or known future recreational or consumptive uses of the Toledo

Bend Reservoir or lower Sabine River Basin (ER Responses 5.8 (p. R-108) and 8.2 (p. R-112); FSES Section 5.2.1).

90. Since the proposed Blue Hills Station is located in a remote area and there are no major groundwater users near the site, changes in groundwater quality and availability due to plant operation are not anticipated. The Applicant has stated that wells used for construction water supply will be capped. However, these wells would not be used for potable water consumption, demineralized water makeup and fire emergency. If so, the use of these wells not be expected to significantly affect other groundwater usage in the area (FSES Section 5.2.2).

91. The heat dissipation system presently proposed by the Applicant for the Blue Hills Station will consist of a closed-loop cooling system with mechanical-draft cooling towers. At full rated load, a small amount of heat will be released to the Toledo Bend Reservoir as cooling tower blowdown, and substantially all of the waste heat will be dissipated to the atmosphere. The environmental effects of operation of this system will be those associated with cooling tower blowdown (thermal and chemical effluents discharged to the reservoir) and cooling tower effects (such as drift deposition and ground-level fogging and icing) (FES Section 5.3). A potential exists for background total dissolved solids buildup above required levels during periods when the reservoir is stratified because of insufficient reservoir circulation and mixing between the hypolimnion and epilimnion. The Applicant should analyze breaching the Cofferdam No. 3 to reduce the stratification potential (FSES Sections 5.3.3, 5.3.4).

92. The proposed discharge system consists of a multiport submerged diffuser. The plant effluents will have to be discharged in such a manner as to comply with all applicable Federal and state requirements. While the discharge from the facility will take place in the State of Texas, inasmuch as the discharge structure will be located near the old Sabine River channel, the boundary between Texas and Louisiana, it is possible that waters within Louisiana may be affected (Tr. 252-4). Because the present water quality requirements of the two states in the Toledo Bend Reservoir are nearly identical, there would appear to be no conflict. Secondly, both the Staff and Applicant have shown that effluents discharged into the Toledo Bend Reservoir will be diluted to required levels within a mixing zone which extends relatively short distances, i.e., less than one hundred feet, from the discharge port. Thus, a relatively small volume and/or area of the reservoir would be affected. Lastly, at the appropriate time, both Texas and Louisiana would have an opportunity to participate in the standards setting for discharges from the Station. *See* Federal Water Pollution Control Act, Sections 401(a)(1) and (2), and 402. The discharge of chlorine in compliance with governing regulations will not result in any adverse impacts on the

aquatic organisms in the Toledo Bend Reservoir and downstream of the Toledo Bend Dam. Sanitary discharges will also be in compliance with appropriate requirements. The impact of discharges is expected to be minimal (FSES Sections 5.3.3, 5.5.2.2).

93. The transportation of cold fuel to a reactor, or irradiated fuel from the reactor to a fuel reprocessing plant, and of solid radioactive wastes from the reactor to burial grounds is within the scope of the NRC report entitled, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants." The environmental effects of such transportation as contained in Table S-4 to 10 CFR Part 51 have been taken into account in the environmental impact analysis of the Blue Hills Station (FSES Sections 5.4.4 and 7.2).

94. The environmental impact of the uranium fuel cycle has been taken into account in the environmental impact analysis of the Blue Hills Station (FSES Section 5.6).

95. The population in the Blue Hills region will increase by approximately 700 persons when operations begin. Of these, approximately 200 will be employed in plant operations, while the remainder will compose the secondary labor force and family members of the work force. Regional income will be increased by the presence of the primary and secondary labor force employed in the Blue Hills region. Retail sales are also expected to increase as a result of the new population doing business in the region (ER Appendix E; FSES Section 5.7.2).

96. An adequate monitoring program to determine the circulation process in the lower basin of the Toledo Bend Reservoir was conducted to serve as a baseline to assess the physical effects of the proposed cooling system. The program provided: (1) detailed data on the bathymetry and physiographic features of the lower basin; (2) detailed current profiles at selected stations; (3) temperature structure during the late summer and early stages of fall mixing; and (4) seasonal variation of temperature structures. In addition, a special field study was implemented to determine the dispersion characteristics of the lower reservoir basin by a long-term fluorescent dye release at the site of the proposed blowdown discharge (ER Appendix D; FSES Section 6.1.1).

97. Subject to the conduct of a Preconstruction Supplemental Monitoring Program recommended by the Staff in the FSES, the baseline aquatic monitoring program is adequate (ER Appendix F, Section IV; FSES Section 6.1.5.2). An offsite preoperational radiological monitoring program to provide for measurement of background radiation levels and radioactivity in the plant environs will be reviewed at the construction permit stage. The preoperational program, which provides a necessary basis for the operational radiological monitoring program, will also permit the Applicant

to train personnel, and to evaluate procedures, equipment and techniques. The program will be initiated two years prior to operation of the facility (FSES Section 6.1.2).

98. Adequate baseline studies of surface waters and groundwater have been performed and an adequate onsite preoperational meteorological program has been conducted. This baseline terrestrial monitoring program is deemed to be satisfactory. The preoperational terrestrial monitoring program will be evaluated at the construction permit stage (ER Section 2.5, 2.6, Appendix F, Sections II and III; FSES Sections 6.1.3, 6.1.5).

99. The Applicant plans essentially to continue the preoperational offsite radiological monitoring program during the operating period. However, refinements may be made in the program to reflect changes in land-use or preoperational monitoring experience. Detailed information on the thermal, meteorological, hydrological, ecological and chemical operational monitoring programs will be provided in the operating license application (ER Section 6.2.1.2; FSES Section 6.2).

100. The environmental impacts of postulated accidents involving radioactive material during operation and during transportation have been adequately considered in the environmental impact analysis (FSES Sections 7.1 and 7.2).

101. The need for power from any units proposed for construction at the Blue Hills site will be evaluated at the construction permit phase (Foreword to ER Section 1.0).

102. The review of alternative energy sources will be made at the construction permit phase (FSES Section 9.1; Tr. 67-68).

103. The Applicant's service area extends 400 miles across Louisiana and into East Texas and is subdivided into three divisions of major power demand: the Baton Rouge area, the Lake Charles area and the East Texas area (ER Fig. 9.2-4). The Applicant stated that a comparison of the three areas showed that each area has conditions suitable for nuclear plants (ER 9.2.1.5.7). The Applicant noted that, since the Louisiana power demands are expected to be met by the River Bend Nuclear Power Station near Baton Rouge and two additional coal units near Lake Charles, and since further load demand is anticipated in the East Texas-West Louisiana area, siting a plant elsewhere to serve the East Texas-West Louisiana area could lead to economic and reliability problems generated by longer transmission lines (ER 42, p. R-19; Tr. 173). The Applicant conducted a comprehensive well-documented site selection process within the East Texas-West Louisiana part of its service area. This process considered, among other factors, site area characteristics, geology, tectonics, seismology, population, power transmission, land use, water availability, transportation and air quality. It identified the Blue Hills site (Site G) as the optimal

location for a nuclear power station, with proper mitigation measures, among 49 sites considered in the East Texas and Western Louisiana (Western region), that area being selected on the basis of load demand. (Sites outside the Applicant's service territory were among those examined (Tr. 220).) At least two specific sites, among these 49 sites, in the Central and Lake Charles region of Applicant's service area, i.e., outside the Western region, were considered (ER at Fig. 9.2-9 and 9.2-4). Moreover, the Atchafalya River on the eastern portion of the Gulf States service area region was considered and rejected as a result of safety and environmental considerations (ER at R-57-R-58). Transmitting power from the eastern or central portion to the projected power demand in East Texas would necessitate the construction of many additional miles of transmission lines, increasing environmental impacts and capital expenditures (*Id.* and ER 9.2.1.5.2). As a result of the initial review by the Applicant, 35 of the 49 sites were eliminated from further consideration. The second phase of the site analysis eliminated eight additional sites based upon the distance from existing transmission lines and in-depth geological and environmental analysis. The third phase of the site analysis consisted of a detailed investigation of six remaining sites, one of which was examined at the request of the Staff. The sites were evaluated with respect to their ability to utilize nuclear, oil and coal generating facilities. Review of the site selection process employed by the Applicant within this area did not reveal any sites there that are obviously superior to the one selected by the Applicant (ER Section 9.3.4; FSES Section 9.2.5; Tr. 173-7). The Staff reviewed the information provided by the Applicant as provided in 10 CFR Section 2.101(a-1)(1) and the Staff personally visited the six candidate sites selected for in-depth analysis (FSES Section 9.2.5). No major flaw was found in the site, and the site appears to be a good site for a nuclear facility if appropriate mitigation action, particularly in regard to socio-economic impacts, is taken before a construction permit is issued (Tr. 173). Thus, we find that the site selection process for the East Texas-West Louisiana area included methods, criteria and considerations given to alternative sites that are acceptable and in full compliance with NEPA and NRC requirements. Alternatives to the heat dissipation system selected were also considered and it was concluded that the circular mechanical-draft towers were optimal. Among the alternative heat dissipation systems considered by the Staff, no system is superior to the mechanical-draft circular cooling towers selected for use by the Applicant (ER Section 10.1; FSES Section 9.3.1.9).

104. The Applicant carefully considered alternatives with regard to railroad right-of-way, access road, makeup and discharge water lines and transmission line corridors prior to selecting the proposed routes and on an

overall basis, no superior routes to those selected by the Applicant have been identified (ER Sections 10.3, 10.9, 10.10.1, 10.10.2; FSES Section 9.3).

105. Among the alternatives considered, the proposed intake (site E), from the standpoint of overall suitability and the physical location and design of the discharge system in relationship to the Toledo Bend Reservoir is such as to minimize environmental impacts associated with construction and operation of the facility and are acceptable (ER Section 10.2.7; FSES Sections 5.5.2, 9.3.2; Tr. 94-7, 222).

106. Inasmuch as the final design of the intake structure has not been completed, review of the actual design and its impact on the Toledo Bend Reservoir will be deferred to the construction permit review phase. The Applicant will submit a report assessing entrainment and impingement associated with the intake structure as well as the feasibility of an intake structure located offshore in a deeper region of the reservoir at or prior to the time that application for construction permits is made (FSES Section 5.5.2.1).

107. Irreversible and irretrievable commitments of resources have been adequately discussed and analyzed in the environmental impact analysis. The ultimate cost benefit balancing process will be deferred until the construction permit phase. However, the comprehensive analyses conducted by the Staff and Applicant have revealed nothing that would preclude use of the Blue Hills site for a nuclear power station. Neither have the Staff's analyses identified on an overall basis alternatives to the site or proposed plant features, including transmission lines, railroad and road access, intake and discharge pipelines, discharge system and proposed intake site E that are superior to those selected by the Applicant.

108. When the actual design of the Blue Hills Station Units 1 and 2 is developed and the Appliant desires to proceed with his application for construction permits, the Applicant will provide, among other items, the following to the Staff:

(1) An evaluation, with necessary supporting information, of the similarities and differences between the actual station design and the station design evaluated in the Final Site Environmental Statement. This evaluation will permit a determination of whether the impact of the actual station design will or will not be significantly greater than or different from the impacts described in the Final Site Environmental Statement.
(2) If the actual plant design will produce an impact or an activity not previously or adequately evaluated in the Final Site Environmental Statement, the Applicant will prepare an environmental evaluation of the design change or new activity. When the

evaluation indicates that such design change or activity may result in a significant adverse environmental impact that was not previously or adequately evaluated or that is significantly greater than that evaluated in the Final Site Environmental Statement, the Applicant shall provide a written evaluation of such design change or activity to the Director, Division of Site Safety and Environmental Analysis for review.

(3) Sufficient information to permit evaluation of the need-for-station and consideration of alternative energy sources, based on a specific date for commencement of commercial operation and revised time sensitive information (e.g., population growth load forecasts, cost estimates, etc.). Unless significant new information is obtained that substantially affects the conclusions reached on alternate sites, no new evaluation of this subject will be required.

(4) A comprehensive evaluation of the multilevel siphon intake system (*See* Final Site Environmental Statement Section 9.3.2) with fish-return facility, unless the state-of-the-art is such that it is appropriate to review this alternative.

(5) An evaluation of the possibility of making a breach in Coffer Dam No. 3 to reduce the potential for total dissolved solids (TDS) buildup in Toledo Bend Reservoir.

(6) Data on the distribution and seasonal abundance of ichthyoplankton, adult fish, and the Asiatic clam (*Corbicula* sp.) in the open-water regions of Toledo Bend Reservoir and a proposed method for control of the latter.

(7) Data on the occurrence of striped bass spawning in Toledo Bend Reservoir.

(8) Quantitative data on the suspended solids, bed load sediments, and periphyton communities in Copperas, Mitchell, and Mill Creeks.

(9) A detailed erosion control plan as discussed in Final Site Environmental Statement Sections 4.3.1.1 and 4.3.2.

(10) A complete description of the pesticide and herbicide treatment program should the Applicant decide that these chemicals are to be used for rights-of-way maintenance as discussed in FSES Section 5.5.1.2.

(11) A detailed description of all preoperational monitoring programs (those which will be implemented after the Construction Permit is issued, but before an Operating License is granted) and the preconstruction supplemental aquatic monitoring program. These programs should incorporate those suggestions offered by the Staff in Final Site Environmental Statement Section 6.1.5.1.

(12) Detailed information and appropriate maps of any significant new changes in the environmental status (e.g., land use, habitats of rare, threatened or endangered species) of the proposed transmission line, pipeline, and railroad access routes.
(13) If the construction schedule described in Final Site Environmental Statement Section 4.4 that provided the basis for the Staff's assessment of community impacts is not achieved, then updated information should be provided on the socioeconomic parameters discussed in this section.
(14) Results of planning negotiations among the Applicant, local officials, and regional planners (Final Site Environmental Statement Section 4.4: 4.4.12). The Applicant should begin early planning negotiations with local officials and regional planners to discuss methods of limiting the adverse impacts that are likely to occur as a result of plant construction. Local items for discussion could include, for example, planning expertise, development of mobile home zoning ordinances, prepayment of taxes, and incentives for workers to commute greater distances. In addition, these negotiations should consider public use, where possible, of the open space used for this project. The Applicant will submit a discussion of its activities carried out under this item and the mitigative activities it will undertake for Staff review at the time a Construction Permit application is filed (*supra*, par. 84).(15).
(15) Results obtained from surveys of the proposed transmission corridor routes to determine the presence of any proposed or nominated endangered species or existence (Final Site Environmental Statement Section 4.3.1.2).
(16) A forest management plan for the site that includes consideration of the red-cockaded woodpecker.
(17) Final plans for minimizing construction impacts or for avoiding the bog communities along the proposed corridor for the railroad spur and transmission line C.
(18) Final designs for both the temporary and permanent sewage treatment facilities (Final Site Environmental Statement Section 11.1.4.6) and revised estimates of water requirements (Final Site Environmental Statement Section 11.1.3.1).
(19) Information on the specific methods to be employed to control particulate emissions from the onsite concrete batch plant (Final Site Environmental Statement Section 11.1.3.9).

109. The Applicant will be required to honor the following commitments to limit adverse effects during construction:

(1) Marketable timber will be removed from the site, and remaining trees and brush will be cleared and either used for erosion control or burned. All burning will be in accordance with State and Federal regulations. Tree stumps and other organics not burned will be buried under adjacent waste areas.

(2) Soil excavation from borrow areas that is unsuitable for fill will be deposited in designated waste areas, and some topsoil will be set aside for restoration of the borrow areas after construction is completed. Tops of borrow areas will be covered with stored topsoil and then planted with slash and loblolly pines (Final Site Environmental Statement Section 4.1.1).

(3) Fordable streams will have shell or gravel placed in the stream bed; other streams will have temporary bridges or culverts installed during construction.

(4) The amount of spoil drifting from the dredging for the makeup intake and discharge structures will be limited to approximately 1% of the total spoil dug from the bottom. Shoreline vegetation will not be disturbed except where it is necessary to gain access to the reservoir.

(5) To minimize disturbance to the reservoir, excavation and construction of the makeup intake structure will take place behind a sheet piling wall. Excavated and dredged material from construction of the makeup intake and makeup channel will be removed to a spoil area on the peninsula; material dredged for the discharge pipe will be deposited adjacent to the discharge pipe.

(6) No explosives will be used in site excavations.

(7) Temporary construction facilities will be removed when construction is completed and these areas will be paved, seeded, sodded, and/or planted according to a prescribed plan.

When no longer in use, temporary construction roads will be disked, scarified, and seeded, and the slope intersections will be rounded to minimize erosion and provide a natural appearance (side slopes in borrow and waste areas will receive similar treatment). All restored areas will be graded to prevent accumulation of standing water.

(8) Permanent lawn areas will be planted as soon as feasible.

(9) A natural border along the periphery of the cleared plant site will be encouraged by allowing natural reseeding and by planting indigenous vegetation.

(10) Dust must be controlled during site preparation and construction through the use of water trucks, sprinkler systems, and chemicals such as Soil Penetrant 400, EARTH-PAK, and COHEREX.

(11) Erosion control will include grading, placement of slash in draws and water courses adjacent to cleared areas, and protection of slopes using peripheral interception ditches, catch basins, and drop pipes equipped with energy dissipators. Additionally, slopes will be treated using chemical soil binders (e.g., Aerospray S2 Binder or Curasol AE) and then mulched and seeded.

(12) During construction, wastes from portable chemical toilets will be transported offsite for proper disposal. Wastes from permanent toilet and wash facilities will be processed in a sewage treatment plant; all treatment plant discharges will meet applicable State and Federal standards.

(13) Floor drain effluent from shop facilities will be discharged into the storm drain system.

(14) Petroleum product wastes will be collected and removed from the site. Waste interceptors will be provided to remove construction wastes (e.g., oils, greases, paints, or solvents) and minimize the impact on neighboring surface waters.

(15) Wash water from the batch plant and from concrete trucks will be discharged into a specially constructed ditch, where cement particles can settle out before the water spills into a berm-enclosed waste area that serves as an evaporation-absorption field. After completion of the power plant, the earth berm will be graded to the elevation of the waste area. Waste loads of concrete will be dumped at a designated waste area.

(16) Controlled spray of herbicides (e.g., Bromacil or Monuron) will be used to inhibit regrowth of vegetation on shelled and paved areas onsite. Application rates of herbicides and pesticides will be such that concentrations in the stream systems will not exceed Texas Water Quality Board requirements; aquatic concentrations will be monitored at the U. S. Geological Survey Gauging Station on Mill Creek.

Pest control, when necessary, will include localized controlled application of a short-lived malathion class of compound (malathion, parathion, EPN) for insects and may include poison baits (e.g., Pyralin or Fumasol) for rats and mice. However, the use of traps for problem rodents is preferred.

(17) Combustible construction wastes will be burned, and noncombustible wastes will be disposed of within the borrow area by landfill methods; both operations will meet applicable State and Federal regulations. Outdoor burning, construction activity, and application for permits shall be accomplished in accordance with

the Texas Clean Air Act and the Rules and Regulations of the Texas Air Control Board.

(18) Noise-reducing apparatus for construction equipment will comply with Federal and industrial standards.

(19) During construction, effluent from the sewage treatment plant will be discharged into a leaching field to prevent as many of the nutrients as possible from reaching the streams.

(20) Effects of siltation upon the creek systems will be minimized through extensive erosion control efforts.

(21) No historical landmarks or archaeological sites within an 8-km (5-mile) radius of the plant site will be disturbed by construction of the station. Any archaeological site that is endangered by transmission line construction will be reexamined and tested.

The Applicant shall not disturb any archaeological site or locality or any historical site without prior approval from the Staff. Should any additional archaeological discoveries be made either on the plant site or within the rights-of-way, the Applicant shall notify the Staff immediately. The four localities identified in FSES Section 2.9.2 shall be posted and an onsite archaeologist shall be available when these sites are in danger of being disturbed unless the State Historic Preservation Officer determines that these localities do not meet the criteria in the National Register of Historic Places (Addendum 2) for inclusion in the Register.

(22) Where a residential or recreational area is serviced by a single road and this road is obstructed by construction activities, an alternate access route will be provided.

(23) Existing roads will be used for access to the transmission corridors.

(24) A forest management plan that will include consideration of the red-cockaded woodpecker will be submitted with the Construction Permit application. Construction activity on the right-of-way for the proposed transmission line A should be carefully monitored by a biologist to ensure that areas with red-cockaded woodpecker nesting or roosting trees are not destroyed. Likewise, on proposed routes B and C, careful investigation should be made for nest and roost trees and areas with active red-cockaded woodpecker colony use, and these areas should be avoided.

(25) An effort will be made to minimize or avoid disturbance of bog communities within the proposed corridor for the railroad spur and transmission line C.

(26) To ensure continued and adequate protection of endangered species during additional development phases of the proposed

facility, the Applicant should maintain consultations with the U.S. Fish and Wildlife Service.

III. CONCLUSIONS OF LAW

110. Following completion of Commission and Atomic Safety and Licensing Appeal Board review, this Partial Initial Decision shall remain in effect for a period of five years or, where the Applicant for the construction permit has made timely submittal of the information required to support the application, as provided in Section 2.101(a-1), until the proceeding for a permit to construct a facility on the site identified in this Partial Initial Decision has been concluded, unless the Commission, Atomic Safety and Licensing Appeal Board, or Atomic Safety and Licensing Board, *sua sponte* or upon motion by a party to the proceeding, finds that there exists significant new information that substantially affects the earlier conclusions and reopens the hearing record on site suitability issues.

111. Upon good cause shown, the Commission may extend the five-year period during which this Partial Initial Decision shall remain in effect for a reasonable period of time not to exceed one year.

112. Based upon our review of the entire record in this proceeding, which are reflected in the foregoing findings, the Board has concluded, to the extent of its review, that the Blue Hills site (Site G) is a suitable location for nuclear power reactors of the general size and type proposed under the requirements of the Atomic Energy Act of 1954, as amended, and Commission regulations promulgated thereunder.

113. Based upon our review of the entire record in this proceeding and the foregoing findings and in accordance with 10 CFR Part 51 of the Commission's regulations, the Board has concluded that the application and the record of the proceeding contain sufficient information and that the review of the application by the Staff has been adequate to support the foregoing findings and the following conclusions and order.

114. We conclude that:

A. The environmental review conducted by the Staff pursuant to the National Environmental Policy Act of 1969 has been adequate to support issuance of this Partial Initial Decision;
B. The requirements of Sections 102(2)(A)(C) and (E) of the National Environmental Policy Act of 1969 and 10 CFR Part 51, to the extent applicable, have been complied with in this proceeding;
C. The Board has independently considered the final balance among conflicting factors contained in the record of this proceeding. After weighing the environmental, economic, technical and other bene-

fits against environmental and other costs, and considering available alternatives, the Board has determined that the Blue Hills site (Site G) is suitable with respect to the factors reviewed, and the Partial Initial Decision should be issued subject to the conditions for the protection of the environment discussed in paragraph 65 and set forth in paragraphs 108 and 109, *supra*, as well as the following:

(1) When the actual design of Blue Hills Station Units 1 and 2 is developed and the Applicant desires to proceed with its application for Construction Permits, the Applicant shall provide to the Staff the information specified in Findings 37, 91 and 108.
(2) The Applicant shall take the necessary actions set forth in Finding #109 to avoid unnecessary adverse environmental impacts from construction activities.
(3) The Applicant shall establish a control program that shall include written procedures and instructions to control all construction activities as prescribed in Finding #109 and shall provide for periodic management audits to determine the adequacy of implementation of environmental conditions. The Applicant shall maintain sufficient records to furnish evidence of compliance with all the environmental conditions herein.
(4) Before engaging in additional construction activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated by the Staff, the Applicant shall provide written notification to the Director, Division of Site Safety and Environmental Analysis, and obtain approval to proceed.
(5) If unexpected harmful effects or evidence of irreversible damage are detected during facility construction, the Applicant shall provide to the Staff an acceptable analysis of the problem and a plan of action to eliminate or significantly reduce the harmful effects or damage.
(6) The Applicant shall monitor the total residual chlorine concentration in the discharges to Toledo Bend Reservoir and shall design its system so that the concentrations can be limited to the value established by the Environmental Protection Agency in the NPDES permit for the Blue Hills Station.

(7) The Applicant shall submit a plan to the Department of the Interior acceptable to the National Park Service that describes the methods for mitigating the environmental impact in crossing the Big Thicket National Preserve along proposed transmission line B.

D. The issuance of permits for construction of the facilities, if built, insofar as they are based upon the findings and conditions herein, will not be inimical to the common defense and security.

IV. ORDER

IT IS ORDERED, in accordance with 10 CFR Sections 2.760, 2.762, 2.785, and 2.786, that this Partial Initial Decision shall constitute, with respect to the matters covered therein, the final action of the Commission thirty (30) days after the date of issuance hereof, subject to any review pursuant to the Commission's Rules of Practice. Exceptions to this Partial Initial Decision may be filed by any party within ten (10) days after service of this Partial Initial Decision.

Within thirty (30) days thereafter (forty (40) days in the case of the Staff), any party filing such exceptions shall file a brief in support thereof. Within thirty (30) days of the filing and service of the brief of the Appellant (forty (40) days in the case of the Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

THE ATOMIC SAFETY AND LICENSING BOARD

Gustave A. Linenberger
ADMINISTRATIVE JUDGE

Dr. Linda W. Little
ADMINISTRATIVE JUDGE

Marshall E. Miller
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 28th day of April, 1981.

[Appendix A has been omitted from this publication but is available at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.]

Cite as 13 NRC 420 (1981) **LBP-81-11**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Herbert Grossman, Chairman
Gustave A. Lineberger, Jr.
Dr. Frank F. Hooper

In the Matter of **Docket No. 50-395**

SOUTH CAROLINA ELECTRIC & GAS COMPANY, *et al.*
(Virgil C. Summer Nuclear Station, Unit 1) **April 30, 1981**

Upon balancing the five factors set forth in 10 CFR §2.714(a)(1), the Licensing Board grants an untimely petition to intervene (subject to petitioner's taking the proceeding as it currently stands), admits petitioner's contentions dealing with emergency planning and corporate management, and denies its other contentions.

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

Failure to read the Federal Register does not justify the untimely filing of an intervention petition. *New England Power and Light Co.* (NEP Units 1 and 2), LBP-78-18, 7 NRC 932, 933-34 (1978).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

Newly acquired standing or organizational status does not constitute justification for an untimely filing of an intervention petition. *Carolina Power and Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 & 4), ALAB-526, 9 NRC 122, 124 (1979).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

A petitioner cannot sit back and observe the proceeding, and then intervene upon deciding that its interests are not being adequately protected by existing parties. *Pacific Gas and Electric Co.* (Diablo Canyon Power Plant, Units 1 and 2), ALAB-583, 11 NRC 447, 448 (1980); *Duke Power Co.* (Cherokee Nuclear Station, Units 1, 2 & 3), ALAB-440, 6 NRC 643, 644 (1977).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

In determining the extent to which the grant of an untimely intervention petition will delay a proceeding, the appropriate test is the measure of delay directly attributable to the tardiness of the petition. *Long Island Lighting Co.* (Jamesport Nuclear Power Station, Units 1 and 2), ALAB-292, 2 NRC 631, 650, fn. 25 (1975).

PARTIAL ORDER FOLLOWING PREHEARING CONFERENCE

(Admitting FUA on Contentions 1, 2, 7-13 and 27, and Denying FUA's Other Contentions)

On March 22, 1981, nearly four years after the notice of opportunity for hearing in this operating license proceeding was published (42 Fed. Reg. 20203, April 18, 1977), and three months before the hearing had already been scheduled to begin (on June 22, 1981), Fairfield United Action (FUA) filed a petition to intervene, together with a supplement thereto setting forth 27 contentions. For each contention, petitioner stated a basis which, in many cases, included names or descriptions of potential witnesses and references to supporting documentation. By the time FUA's petition was filed, the Licensing Board had issued an order (on March 10, 1981) setting a final prehearing conference in the proceeding for April 7, 1981, and had requested the existing parties to file their suggestions by March 31, 1981 with regard to all actions to be taken by the Board at the conference. Applicant filed a response on March 30, 1981 and suggested, among other things, that FUA's petition be considered at the conference. Applicant served that response on FUA and the Board also arranged for its March 10, 1981 order setting the prehearing conference to be served on the petitioner.

Petitioner appeared at the prehearing conference by a non-attorney member, Dr. John Ruoff, to argue in support of the contentions it had raised and by an attorney, Robert Guild, Esq., a member of the bar of the State of South Carolina making a special appearance to argue the merits of the late intervention. The Staff joined applicant in arguing against allowing intervention at this late date, and later reaffirmed this position in a written opposition to the petition, which it filed on April 13, 1981. The main thrust of the applicant's and Staff's opposition to the petition is the purported lack of cognizable "good cause" for the late filing and the alleged delay that might be caused by allowing an intervention so shortly before the scheduled hearing date. The major reasons given by petitioner for the late filing were that the petitioner was only recently incorporated, on September 5, 1980; that its members have only recently educated themselves with regard to the Summer Nuclear Station through participation in petitioner's program; that some of petitioner's members have only recently moved to Fairfield County; that the members who have lived for many years in the County have until recently relied upon information from applicant concerning the operations of the plant, which they now believe to be false and misleading; that petitioner's members who resided in proximity to the facility at the time of the filing of the application for the operating license in 1977 lacked knowledge that they had interests that might be adversely affected by the granting of the license, of their rights and remedies available to them, and of the notice of opportunity for hearing published in the *Federal Register;* that until mid-February 1981 petitioner believed that it had no right to participate as a party in this proceeding since the deadline for intervention had passed in May 1977; that it believed until mid-February 1981 that its interests were represented by the existing intervenor Brett Bursey, when it was informed that Mr. Bursey's ability to put on an affirmative case was restricted by the Licensing Board; and that the ability of petitioner to inform itself of developments in the proceeding had been severely hampered by the absence for several years of a properly managed local public document room in Fairfield County.

In addition to alleging a lack of good cause and inevitable delay that would result from admitting petitioner, applicant and Staff applied the other three factors contained in the five-factor test of 10 CFR §2.714(a)(1) against the petition to conclude that it should not be granted. They did not, however, challenge FUA's standing to intervene or the legal sufficiency of its contentions, and it is clear that they could not: the members reside well within the geographical limits required for intervention and many of the contentions were either encompassed in contentions admitted by the Board on behalf of intervenor Brett Bursey or would otherwise be ruled admissible in an operating license proceeding.

The Board rules on the intervention by dividing the contentions into two parts in applying the five-factor test of 10 CFR §2.714(a): (1) the corporate management contentions (1, 2, 27) and emergency planning contentions (7-13); (2) all other contentions. As specifically discussed below, by applying the five-factor test to these two categories of contentions in the current posture of the proceeding we admit FUA to the proceeding only on the corporate management and emergency planning contentions. In doing so, we require that the newly admitted intervenor take the proceeding as it currently stands with formal discovery concluded and only the specifics of FUA's affirmative case on those issues accepted as they were detailed in the supplemental petition and the prehearing conference.

Good Cause for the Late Intervention

The Board agrees with applicant and Staff that, with respect to the good cause requirement, petitioner had not substantiated its charges of misrepresentation by the applicant in its dissemination of information to the public; petitioner has not demonstrated that it exercised due diligence with regard to its rights, remedies and its potential interest in the proceedings; failure to read the *Federal Register* does not justify non-timely filing of a petition (*New England Power and Light Co.* (NEP Units 1 and 2), LBP-78-18, 7 NRC 932, 933-934, (1978)); newly acquired standing or organizational status is not an excuse for delay (*Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1-4), ALAB-526, 9 NRC 122, 124 (1979)); a petitioner cannot sit back and observe the proceeding, and then intervene upon deciding that its interest is not being adequately protected by existing parties (*Puget Sound Power & Light Co.* (Skagit Nuclear Power Project, Units 1 and 2), ALAB-559, 10 NRC 162, 172-173 (1979), *vacated as moot* CLI-80-34, 12 NRC 407 (October 9, 1980); *Duke Power Co.* (Cherokee Nuclear Station, Units 1, 2 and 3), ALAB-440, 6 NRC 643, 644 (1977); *Pacific Gas and Electric Co.* (Diablo Canyon Power Plant, Units 1 and 2), ALAB-583, 11 NRC 447, 448 (1980)); and, the poor maintenance of a local public document room (which the Board judges to be the fact upon reading the submittals and considering the discussion at the prehearing conference) does not justify the four years of delay and failure to raise the matter with NRC or the applicant.

With regard to petitioner's reliance upon post-TMI requirements as providing good cause for late intervention, however, the Board does not agree with applicant and Staff that they do not provide good cause for the late intervention with regard to corporate management and emergency planning contentions. Especially with regard to emergency planning, we agree with another licensing board, *Cincinnati Gas and Electric Co.* (William H. Zimmer Nuclear Station), LBP-80-14, 11 NRC 570, 574 (1980), that the criteria for emergency planning have undergone vast changes that have considerably expanded the scope of relief in operating license proceedings since the TMI-accident and especially during 1980. Without repeating in detail the changes summarized in *Zimmer*, we do note the example cited there (*Id.* at 573) of the extension of emergency planning from the low population zone (LPZ) to the Emergency Planning Zones (EPZs). This concept was formally adopted in the final rules published on August 19, 1980 (45 Fed. Reg. 55402) which established an EPZ for airborne exposure with a radius of about 10 miles from the facility and an EPZ for contaminated food and water with a radius of about 50 miles. The affidavits submitted with the petition to intervene identify members of FUA who live within those zones and, consequently, who formally became principals in the Commission's concern over emergency planning. We note further that it was during this period in mid-1980 in which the Commission's policy on EPZs was evolving that the members of FUA began their involvement in NRC emergency planning meetings and organizational activities, culminating in FUA's incorporation in September of 1980. Tr. 586.

Had FUA filed this petition in the middle or latter part of 1980, we would have no hesitation in determining that there was good cause for the delay in filing the petition to the extent of the emergency planning issues. Similarly, although to a lesser extent, because of the Commission's focus on management capability in the post-TMI era we would have found good cause for the delay in filing the management capability contentions.

As it is, petitioner delayed some months longer in apparent reliance upon Mr. Bursey's intervention before filing its petition in March of 1981. As we have stated before, such reliance is legally insufficient to constitute good cause for the additional delay, although we can understand a reluctance to file a petition three years after the issuance of a notice of opportunity for hearing in the face of a strong possibility of rejection when there is an intervenor already participating in the proceeding. Had that added delay in filing disadvantaged any parties other than petitioner itself (by circumscribing its prehearing activities), or delayed the proceedings, we might find a lack of good cause. However, since it does not delay the proceeding and there was good cause for the bulk of the delay in filing these contentions, we find that factor to be of almost no weight (or of slight weight against petitioner) in deciding upon the intervention with regard to the corporate management and emergency planning issues. With regard to the other contentions, we find an absence of good cause for the delay.

The Delay Factor

The Board agrees with applicant (Applicant's Answer to Untimely Petition, p. 10) that in cases of very late intervention the fifth factor specified in 10 CFR §2.714(a)(1), the extent to which participation by the late petitioner will broaden the issues or delay the proceedings, becomes very important. We further agree with both applicant and Staff as to the contentions other than those concerning corporate management or emergency planning that the admission of petitioner would broaden the proceeding and cause unwarranted delay at this late stage. We would weigh this factor as heavily against admitting petitioner on these contentions as we would weigh the lack of good cause. With regard to emergency planning and corporate management, however, we see no delay resulting from petitioner's admission if, as the Board orders, petitioner's admission on these contentions be subject to the same conditions prevailing with regard to the other parties. When a petitioner files a late petition he must generally take the proceedings as they are, and we see no reason to make any special accommodations for this petitioner that would result in delaying the proceeding. At the time the petition was filed, the hearing had been scheduled to begin on June 22, 1981, and we intend to maintain that schedule. Furthermore, the parties' affirmative cases should have been disclosed and discovery concluded except on those issues on which the Staff's and applicant's positions were still evolving. We hold petitioner to the specifics disclosed in its supplemental petition or at the prehearing conference on the corporate management and emergency planning issues, except to the extent that the latter area is still evolving or has not been publicly disclosed.

In view of the fact that the corporate management and emergency planning issues had already been admitted to the proceeding (by Board question or intervenor contention), we see no broadening of issues and only a desirable particularization of its position in FUA's detailed presentation of these contentions.

The Board expects that no delay will ensue from admitting petitioner on these contentions if the appropriate test of delay is employed, i.e., measuring the delay that could be attributed directly to the tardiness of the petition. *Long Island Lighting Co.* (Jamesport Nuclear Power Station, Units 1 and 2), ALAB-292, 2 NRC 631, 650, fn. 25 (1975). Had petitioner filed a timely petition, it would have served itself by having before it a full discovery period. While the other parties could have also discovered petitioner's case, discovery would not have benefitted them on the issues we are admitting. Petitioner has made full disclosure in its supplemental petition of the bases for its contentions, including the names or offices of its potential witnesses to the extent we are admitting its contentions, for the Board will not allow additional witnesses. Consequently, FUA's late entrance into the case has not occasioned a delay in discovery that could prolong the proceeding. With regard to applicant's and Staff's evolving positions on emergency planning, discovery is presently where it would have been had petitioner been admitted when the notice of opportunity was issued. We direct, in this regard, for the benefit of all of the parties, that the parties cooperate in informal discovery with regard to the evolving plans.

While the Board intends to adhere firmly to the hearing starting date of June 22, 1981, notwithstanding any failure in cooperation with regard to informal discovery, the Board intends to exercise its prerogatives in controlling the proceeding to penalize an offending party either by restricting its case or by providing a further hearing at a later time for the benefit of an aggrieved party.

Nor, do we see any way in which petitioner's sooner entrance into this proceeding could have resolved the issues being admitted. Emergency planning is not yet ripe for resolution, and neither the corporate management nor emergency planning issues are susceptible to summary disposition regardless of their state of preparedness.

Even if we consider delay in terms of the time of concluding the proceeding measured with or without petitioner's participation, we cannot foresee unwarranted delay. To be sure, the hearing may last longer because of petitioner's participation but, in view of petitioner's apparent intensive preparation of its pleadings and its demonstrated knowledge of the areas on which it is being admitted, together with the Board's resolve to prohibit repetitious examination, the Board anticipates very little *unproductive* delay.

Ability to Contribute to a Sound Record

It is this factor that the Board weighs most heavily in favor of admitting petitioner to this proceeding on the corporate management and emergency planning contentions and which it weighs most heavily against petitioner with regard to the other contentions. As is apparent from FUA's pleadings and from the general discussion at the prehearing conference, petitioner's members have become well versed in the former areas, independently of any intention of intervening in this proceeding, through their participation in rate-making proceedings and in the ongoing emergency planning. We can only contrast petitioner's familiarity with the substance of these issues with its lack of prior involvement or expertise in the other issues it raised. On those other issues, it named few or no witnesses committed to testifying on its behalf but sought mainly the opportunity to search for such witnesses. In view of the late date, we see no reason to afford that opportunity.

Moreover, while perhaps not grounds for admitting this petitioner, we cannot help but consider what the state of the record might be on the issues we admit without its participation. The existing intervenor, Mr. Bursey, throughout this proceeding has exhibited an inability to effectively manage his case, which includes the area of emergency planning. Moreover, considering the difficulties Mr. Bursey has encountered in preparing his own case, we expect little help from him in assisting the Board with regard to the issue raised by the Board regarding corporate management.

(However, in this regard, we would expect the Staff to render valuable assistance since it, too, has raised serious questions with regard to applicant's engineering organization and hands-on operating experience. See reference to SER in ACRS letter of March 18, 1981, pp. 2-3.) However, with petitioner admitted on the corporate management issues it raised itself, we anticipate a much fuller development of the record, in a more adverserial manner.

Other Means to Protect Petitioner's Interest and Extent to Which Petitioner's Interests will be Represented by Existing Parties

As is ordinarily the case, this proceeding represents the best forum to consider the admissible contentions and petitioner is best qualified to represent its own interests. For that reason, these factors almost always weigh in a petitioner's favor but are given relatively lesser weight than the other factors. The Board has, however, taken these factors into account with regard to the specifics of this petition. We note that, with regard to emergency planning, petitioner has had dealings with NRC and other public officials without benefit of this formal proceeding but, on the other hand, has encountered considerable difficulty in gaining full access to the counties' evolving emergency plans. Tr. 597-603.

Petitioner's admission into this proceeding on the emergency planning contentions should not only facilitate its being heard on those issues in this forum, but should also serve to open some of the emergency planning to public input and scrutiny as should have been the case from the first.

With regard to petitioner's being adequately represented by the existing parties, we have already expressed our opinion on the manner in which the existing intervention has been handled. We see no reason why petitioner should have any confidence that Mr. Bursey will represent its interests any better than he has, so far, represented his own.

In summary, we have applied the five-factor test to FUA's proposed intervention on the corporate management and emergency planning issues and have concluded that, while the good cause factor weighs slightly against admission, petitioner's ability to contribute to a sound record and the lack of delay or broadening of the proceedings weigh heavily in its favor, and the other two factors weigh slightly in its favor. We, therefore, conclude that the five factors weigh in favor of admitting FUA on the corporate management and emergency planning issues.

On applying the five-factor test to the remainder of the issues raised by FUA, we conclude that the good cause, delay, and ability-to-contribute to-a-sound-record factors weigh heavily against admission, and that only the lesser factors of availability of other means to protect petitioner's interest and the extent to which petitioner's interest will be represented by existing parties weigh slightly in its favor. We must, therefore, reject petitioner's intervention on those other issues. Had we not been able to separate its petition into two discrete parts for applying the five-factor test, we would have denied the petition as a whole, because the factors of lack of good cause for failure to file on time and the extent to which admitting petitioner on those issues would broaden the issues and delay the proceeding would outweigh any benefits from admitting petitioner.

ORDER

For all of the foregoing reasons and based upon a consideration of the entire record in this matter, it is, this 30th day of April 1981

ORDERED

That Fairfield United Action *is admitted* as an intervenor in this proceeding on contentions 1, 2, 7-13, and 27, subject to all of the rights, obligations, and restrictions of the other parties as discussed above and determined in other Board orders; and,

That the remainder of the contentions raised in FUA's supplemental petition are *not admitted*.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Herbert Grossman, Chairman
ADMINISTRATIVE JUDGE

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **Docket No. PRM-2-10**

CITIZENS ADVISORY BOARD OF THE METROPOLITAN AREA PLANNING COUNCIL FOR OMAHA, NEBRASKA, AND COUNCIL BLUFFS, IOWA **April 20, 1981**

The Commission denies a petition for rulemaking submitted by the Citizens Advisory Board. The petitioner requested a variety of amendments to the Commission's Rules of Practice, 10 CFR Part 2, including provisions for informal hearings where formal hearings would not be held and requests for hearings to be filed by persons not attempting to intervene in the proceeding. Petitioner also sought expanded service of all docket-related papers and the holding of all hearings and meetings at reactor sites at times maximizing public attendance.

NRC: INFORMAL HEARINGS AND MEETINGS

NRC currently holds informal public hearings or meetings near the site, in the area of NRC regional offices, or in the Washington, D.C. area on matters of special public interest relating to both specific nuclear plants and to more generic issues.

NRC: INFORMAL HEARINGS AND MEETINGS

NRC informal hearings and meetings are designed and conducted to achieve several objectives—to inform the public of proposed NRC or licensee actions, to enable the public to observe firsthand the NRC regulatory process at work, to air differing views on the matters in issue, and to provide an opportunity for the public to question NRC and licensee personnel directly.

NRC: INFORMAL HEARINGS AND MEETINGS

Members of the public are generally notified in advance of the informal hearings or meetings through notices published in local newspapers, notices published in the Federal Register, radio and television announcements, or through a combination of these methods.

NRC: INFORMAL HEARINGS AND MEETINGS

Technical meetings between the NRC staff and the licensee or applicant are generally open to the public pursuant to the NRC "Open Meetings" policy, which is fully described in a policy statement issued on June 28, 1978 (43 FR 28058) and in another published on October 20, 1978 (43 FR 49082).

NRC: INFORMAL HEARINGS AND MEETINGS

The Commission is keenly aware of the public interest in informal hearings and meetings and will continue to investigate and encourage approaches which will meaningfully enhance, in a sound and practical manner, the positive effects of public participation in the nuclear licensing process.

NRC: INFORMAL HEARINGS AND MEETINGS

Public participation in the NRC regulatory process is not, however, a goal which can be pursued without regard for budgetary and personnel limitations, and the Commission must take into account reductions in its financial resources and limitations on its personnel strength in pursuing the fulfillment of all of the NRC's responsibilities and objectives.

NRC: INFORMAL HEARINGS AND MEETINGS

Mandatory diversion on additional NRC resources to informal public meetings or hearings in every type of license proceeding throughout the entire country, as petitioner seeks here, would adversely affect the ability of the NRC to fulfill its fundamental environmental, health, and safety statutory responsibilities.

NRC: INFORMAL HEARINGS AND MEETINGS

Making informal hearings mandatory in all operating license proceedings, as the petitioners have requested here, is not appropriate at this time and the Commission declines to formalize the use of informal hearings as a requirement in all sucn cases.

OPERATING LICENSE HEARINGS: HEALTH AND SAFETY ISSUES

Informal hearings will not be made mandatory for all operating license proceedings.

RULES OF PRACTICE: STANDING TO INTERVENE

Interested members of the public may request intervention in adjudicatory hearings pursuant to Section 189 of the Atomic Energy Act providing they can demonstrate the requisite interest at stake, the standing requirement, 10 CFR 2.714; *see Portland General Electric Company* (Pebble Springs Nuclear Plant, Units 1 and 2), CLI-76-27, 4 NRC 610 (1976).

RULES OF PRACTICE: DISCRETIONARY INTERVENTION

The Commission has discretion to order hearings upon request even where standing has not been shown. *Public Service Company of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), CLI-80-10, 11 NRC 438, 439 (1980).

RULES OF PRACTICE: RIGHT TO PARTICIPATE

Limited participation in NRC proceedings by nonparties is also permitted under the current rules, 10 CFR 2.715(a).

RULES OF PRACTICE: STANDING TO INTERVENE

The purpose of the requirement of standing is to establish that participants in the hearing process will contribute in a meaningful way to the development of a complete record on health, safety, and environmental issues, and this ensures that the public funds used to provide the adjudicatory resources for such proceedings will not be expended on matters which are of no relevance to the issues then being adjudicated.

RULES OF PRACTICE: RIGHT TO PARTICIPATE

The requirement of standing will not be essentially abrogated to allow, as petitioner suggests, any person to request an adjudicatory hearing, irrespective of whether that person can show any personal stake in the outcome of the hearing and also irrespective of whether that person intends to participate in the hearing itself.

RULES OF PRACTICE: NOTICE OF HEARINGS

10 CFR 2.715(b) requires that the Secretary of the Commission serve notices of hearing upon all persons requesting such notices, and a docket list relating to each nuclear licensing matter is compiled by the Office of the Secretary, and persons on this list receive direct notification of all hearings, pre-hearing conferences, oral arguments, and other formal proceedings associated with the docket.

RULES OF PRACTICE: NOTICE OF HEARINGS

All formal proceedings are noticed in the Federal Register and in local publications selected as being those reasonably calculated to provide the widest notice to the largest number of potentially affected people.

RULES OF PRACTICE: SERVICE OF PAPERS

NRC Local Public Document Rooms (LPDRs) are situated near the site of each licensed or proposed nuclear power plant and each LPDR contains the entire file of docket-related papers for that site, along with other NRC documents of general public interest. The main NRC Public Document Room in Washington has all dockets and corresponding docket-related papers on file as well as most other publicly-available NRC documents.

RULES OF PRACTICE: SERVICE OF PAPERS

The LPDRs, PDR, and requests under the Freedom of Information Act adequately provide interested members of the public access to NRC documents, particularly those related to a specific proceeding.

RULES OF PRACTICE: SERVICE OF PAPERS AND FINANCIAL ASSISTANCE TO PARTICIPANTS

Nationwide service upon request, as petitioner suggests, would be an unjustifiable expense, would not measurably add to public knowledge regarding NRC proceedings, and would seem contrary to the recent ruling of the Comptroller General that Section 502 of the 1981 fiscal year Energy and Water Development Appropriations Act (Pub. L. 98-367) prohibits the NRC from providing certain documents and transcripts free of charge to non-applicant parties in adjudicatory proceedings. *See* CG Opinion No. B-200585 (Dec. 3, 1980).

NRC: LOCATION OF INFORMAL HEARINGS AND MEETINGS

Public meetings of an informal nature are ordinarily held near the site, particularly when they involve issues relating to the nuclear plant, and meetings with licensees or applicants may be held either near the plant or reactor site or in Washington, depending on the nature of the meeting, the convenience to the parties involved, and urgency of the meeting.

NRC: LOCATION OF INFORMAL HEARINGS AND MEETINGS

Requiring that all formal and informal hearings and meetings normally be held proximate to the actual or proposed nuclear plant site would not, in most cases, increase public convenience since reactor sites are not located in high population density areas, and most interested members of the public would reside in a nearby population center, where hearings are held under the current system; furthermore, reactor sites are not properly equipped to accommodate large public meetings.

NRC: SCHEDULING OF INFORMAL HEARINGS AND MEETINGS

The efficient conduct of hearings and meetings requires that they generally take place during normal business hours on weekdays, although special arrangements are often made to accommodate members of the public wishing to appear at the hearing but unable to do so during business hours.

DENIAL OF PETITION FOR RULEMAKING

This petition for rulemaking was filed by the Citizens Advisory Board of the Metropolitan Area Planning Agency for Omaha, Nebraska, and Council Bluffs, Iowa on March 13, 1980. Petitioner sought a number of amendments to the Commission's Rules of Practice, 10 CFR Part 2. The petitioner's proposals were set out in the Federal Register notice requesting comment on the petition. 45 Fed. Reg. 26071. In brief, the petitioner sought the following:

1. An amendment to 10 CFR 2.105 which would require that an "informal hearing" be held by the NRC staff in all licensing cases where a "formal hearing" is either unavailable, not requested, or requested and denied.
2. An amendment to 10 CFR 2.714 giving persons not attempting to intervene in a licensing proceeding the right to request a formal hearing.
3. An amendment to 10 CFR 2.715 providing that any person so requesting would be furnished by the Secretary of the NRC all docket-related papers and be sent notice of all hearings, conferences, and informal proceedings.
4. An amendment to 10 CFR 2.751 requiring that all hearings and NRC-licensee/applicant meetings be held at a site and at times maximizing attendance by a majority of persons potentially affected.

Thirteen public comments were received on the petition, all of which opposed the petition. Commenters stressed that petitioner's suggestions would add cost and delay to the licensing process, were unnecessary in view of current NRC rules providing for public participation in licensing and were subject to abuse by persons seeking only to delay licensing rather than contribute to the process by good faith participation.

We have considered the Citizens Advisory Board petition and the comments submitted in response and have concluded that the petition should be denied. The reasons for our denial of the Citizens Advisory Board petition may best be understood in light of the NRC's current practice with regard to informal public meetings or hearings, particularly since the substance of most of the concerns expressed by the petitioner are already met under our present practice. We will discuss these matters in response to the four basic areas of concern raised in the Citizens Advisory Board petition.

1) NRC Informal Hearings and Meetings.

The NRC currently holds informal public hearings or meetings near the site, in the area of the NRC regional offices, or in the Washington, D.C. area on matters of special public interest relating both to specific nuclear plants and to more generic issues. Recent informal public hearings or meetings have covered a wide range of subjects, including (a) environmental, health, and safety matters related to applications for construction permits or operating licenses for nuclear power plants, (b) upgrading emergency preparedness plans at operating nuclear power plant sites, (c) the NRC's proposed policy and procedures for enforcement actions, and (d) NRC enforcement actions against specific licensees. Such meetings and hearings are designed and conducted to achieve several objectives: to inform the public of proposed NRC or licensee actions, to enable the public to observe firsthand the NRC regulatory process at work, to air differing views on the matters in issue, and to provide an opportunity for the public to question NRC and licensee personnel directly. To maximize participation, members of the public are generally notified in advance of the informal hearings or meetings through notices published in local newspapers, notices published in the *Federal Register,* radio and television announcements, or through a combination of these methods.

The public meetings on environmental, health, and safety matters related to applications for construction permits or operating licenses for nuclear power plants are noteworthy. These meetings have generally been in two areas: (1) special meetings on environmental, health, or safety matters among the NRC staff, licensee/applicant personnel, and the public, and (2) other technical meetings between the NRC staff and the licensee/applicant. Two examples illustrate these types of informal public meetings. In the early stages of NRC consideration of the construction permit application for Palo Verde Units 4 and 5, open public meetings were held in Phoenix, Arizona, on environmental matters (October 12 and 13, 1978) and on safety matters (October 17 and 19, 1978). At the Palo Verde meetings, information was presented to the public, and question-and-answer sessions followed.

In connection with a proposed increase of the maximum power rating in the operating license for the Fort Calhoun Nuclear Station, an informal meeting was held on January 16, 1980 in Omaha, Nebraska. NRC staff members participated along with the licensee Omaha Public Power District (OPPD), parties (both individuals and groups) that had previously requested a formal NRC hearing on the matter, and other members of the public. During the meeting, OPPD presented its plans for the power increase, the NRC staff discussed its review of OPPD's proposed power increase, and other participants made their views known and questioned

OPPD and NRC participants. Shortly after the meeting, the request for a formal NRC hearing was withdrawn, and the NRC received favorable comments on the exchange of information and views which had taken place.

Technical meetings between the NRC staff and the licensee or applicant are generally open to the public pursuant to the NRC "Open Meetings" policy, which is fully described in a policy statement issued on June 28, 1978 (43 FR 28058) and in another published on Ocotober 20, 1978 (43 FR 49082). Other special meetings are held where circumstances and public interest commend such action. For example, approximately 70 meetings were held in 1980 with the public, local officials, and other interested organizations in the area near the Three Mile Island (TMI) plant on various subjects related to the status of TMI.

As to the three other general subjects mentioned above, approximately 130 informal hearings or meetings took place during 1980 in areas immediately surrounding the operating or proposed nuclear plant itself or in the general areas where such plants are or will be situated. Over 30 of the 130 local meetings focused on the NRC emergency preparedness program. These 30 meetings and workshops involved providing the public with information on proposed NRC emergency preparedness regulations, presenting an evaluation of the status of the emergency preparedness plans for the nuclear power plant in that area, and giving the public an opportunity to question NRC and licensee personnel directly on these topics. In addition, proposed policy and procedures for NRC enforcement actions were discussed at several regional public meetings which took place in 1980. The enforcement policy and procedures and the schedule of meetings were announced in NRC press releases and published in the *Federal Register.* 45 FR 66754 and 45 FR 69077.

Open, informal meetings have also included matters subject to NRC enforcement actions, usually where licensees had received an NRC notice of violation of the terms or conditions in their construction permits or operating licenses. In 1980, such open enforcement meetings were held, among others, in Athens, Alabama, on Browns Ferry Unit 3 (containment penetration closures and TVA operational procedures); in Madison, Indiana, on the Marble Hill Nuclear Power Station (upgrade of quality assurance program and construction management activities); in Sacramento, California, on the Rancho Seco Nuclear Generating Station (valve misalignment and administrative procedures); in Bay City, Texas, on South Texas Project Units 1 & 2 (construction activities and quality assurance program); in New York City on Indian Point Unit 2 (river water leakage into containment); and in South Haven, Michigan, on the Palisades Nuclear Power Station (mispositioned safety system valves and routine surveillance test procedures). At these open meetings, NRC personnel questioned the licensee on various aspects of the violation and proposed or completed remedial actions, with the public observing the entire process. Following this segment of the meeting, the public had an opportunity to question the NRC personnel present and, at times, those of the licensee. Attendance at such open enforcement meetings ranged from one person to large crowds of several hundred persons.

As a result of activities such as the examples noted above, the NRC has found that open, informal meetings and hearings have positive, useful effects in permitting the public to judge for itself the effectiveness of nuclear regulation by the NRC. The NRC staff continues to explore ways to improve its anticipation of matters which have considerable public interest, so that informal hearings and meetings may be scheduled. Open and informal meetings also provide a valuable forum for members of the public to receive information on NRC practices and policies directly from NRC personnel, and to make known their own views on such matters. Positive effects flow from a face-to-face exchange of ideas and from the ability of the public to have question-and-answer sessions. Usually, the questions range from general subjects and NRC regulations and policies to the licensee's actual compliance with such NRC mandates.

The Commission is keenly aware of the public interest in this area and will continue to investigate and encourage approaches which will meaningfully enhance, in a sound and practical manner, the positive effects of public participation in the nuclear licensing process. To that end, increased efforts have been and are being made to afford interested members of the public an opportunity to participate in these informal meetings and hearings at convenient locations, usually in the vicinity of the nuclear power plant location. In addition, providing effective advance notice in widely-disseminated local and national media is obviously necessary and will continue to be part of NRC practice. Finally, the NRC will attempt to schedule such hearings and meetings with due regard for the most appropriate and convenient time of the day for all concerned. In some of the instances noted above, meetings and hearings have continued through the evening well into the early morning hours.

Public participation in the NRC regulatory process is not, however, a goal which can be pursued without regard for budgetary and personnel limitations. The Commission must take into account reductions in its financial resources and limitations on its personnel strength in pursuing the fulfillment of all of the NRC's responsibilities and objectives. In addition the recent decision of the U.S. Court of Appeals for the District of Columbia Circuit in *Sholly v. NRC*, No. 80-1691 (Nov. 19, 1980), creates further uncertainties for NRC budgetary and personnel resources, pending possible Supreme Court review of that decision. The impact of *Sholly* on the NRC's responsibility to hold formal adjudicatory hearings, if not reversed by the Supreme Court or legislatively, is potentially substantial, and is presently undergoing close scrutiny. Any increase in the number of formal hearings that are ultimately required to be held, above estimates made prior to *Sholly*, will obviously have fiscal and personnel impacts on the NRC's ability to hold informal hearings and meetings that are discretionary in nature. Accordingly, making informal hearings mandatory in all operating license proceedings, as the petitioners have requested here, is not appropriate at this time and we decline to formalize the use of informal hearings as a requirement in all such cases.

Information concerning proposed NRC licensing actions is and will continue to be available in the *Federal Register* and in NRC Public Document Rooms. Current NRC regulations already provide an avenue for members of the public to request that certain licensing or enforcement actions be taken. *See* 10 CFR 2.206. Interested members of the public may also request and intervene in adjudicatory hearings pursuant to Section 189 of the Atomic Energy Act providing they can demonstrate the requisite interest at stake, the "standing" requirement. 10 CFR 2.714; *see Portland General Electric Company* (Pebble Springs Nuclear Plant, Units 1 and 2), CLI-76-27, 4 NRC 610 (1976). Nonetheless, the Commission has discretion to order hearings upon request even where standing has not been shown. *Public Service Company of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), CLI-80-10, 11 NRC 438, 439 (1980). Limited participation in NRC proceedings by nonparties is also permitted under the current rules. 10 CFR 2.715(a).

In our view these avenues, together with the above-described NRC practice of holding informal hearings and meetings where circumstances warrant, are sufficient to ensure effective public participation in the NRC regulatory process. Mandatory diversion of additional NRC resources to informal public meetings or hearings in every type of license proceeding throughout the entire country, as petitioner seeks here, would adversely affect the ability of the NRC to fulfill its fundamental environmental, health, and safety statutory responsibilities. Substantial delays in the licensing of nuclear power plants could also result, since even minor license amendment actions would have to await conclusion of the informal hearings. The

Commission needs to maintain some measure of control in deciding when circumstances warrant the holding of informal public hearings or meetings and cannot allocate, in advance, the substantial resources necessary to meet the full breadth of the petitioner's request absent a stronger showing of deficiencies in our current practice and of substantially greater benefits to be gained. The TMI experience, with over 70 informal meetings in 1980 alone, demonstrates that the NRC can and will exercise its discretion to involve the public in a substantial manner in deserving situations.

(2) The Right of Persons Not Attempting to Intervene to Request Formal NRC Hearings.

Under current NRC rules, person seeking to intervene in NRC proceedings must meet a traditional threshhold requirement to show how their interests will be affected by the outcome of the proceeding. 10 CFR 2.714(a), (d). *See also, Portland General Electric Company, supra; Public Service Company of Indiana, supra.* The purpose of this requirement of "standing" is to establish that participants in the hearing process will contribute in a meaningful way to the development of a complete record on health, safety and environmental issues. This ensures that the public funds used to provide the adjudicatory resources for such proceedings will not be expended on matters which are of no relevance to the issues then being adjudicated.

Petitioner suggests, however, that any person should be able to request an adjudicatory hearing, irrespective of whether that person can show any personal stake in the outcome of the hearing and also irrespective of whether that person intends to participate in the hearing itself. We have difficulty in finding significant positive aspects in such a proposal, particularly in light of the above-mentioned practical benefits which result from employing the well-accepted standing requirement. The petitioner's suggestion would essentially abrogate the standing requirement entirely while creating certain anomalous and costly situations. For example, although at times intervenors pose views which differ from those of a licensee/applicant or the NRC staff, it is not difficult to conceive that adoption of petitioner's proposal would lead to empaneling a licensing board and holding a hearing without any participation by a party taking a differing position on the issues. Indeed, petitioner does not state what the issues might be at such a hearing, since the proposed rule change does not require the person requesting the hearing even to identify such issues with particularity. Adjudicatory hearings held under these circumstances would not contribute to the enhancement of the NRC's ability to protect the public health and safety and, in fact, would seem to be an expenditure of NRC resources without any benefit being gained.

(3) Furnishing of Docket-Related Papers and Notices of All Hearings, Conferences and Informal Proceedings.

Petitioner would amend 10 CFR 2.715 to provide for service of all docket-related papers ("all pleadings and papers of record") to any person so requesting, whether or not a particpant in the proceeding. This would be an expansion of current practice that seems unnecessary in light of the steps the NRC presently takes to ensure that members of the public have reasonable access to all docket-related papers, notices, and other material. NRC Local Public Document Rooms (LPDRs) are situated near the site of each licensed or proposed nuclear power plant in the United States. Each LPDR contains the entire file of docket-related papers for that site, along with other NRC documents of general public interest. The main NRC Public Document Room in Washington has all dockets and corresponding docket-related papers on file as well as most other publicly-available NRC documents. The LPDRs can usually obtain additional materials for persons requesting them on short notice. Finally, the Freedom of Information Act is available to persons desiring to obtain documents which would not ordinarily be placed in an LPDR or in the main NRC PDR in Washington.

We are confident that these methods adequately provide interested members of the public access to NRC documents, particularly those related to a specific proceeding. Nationwide service upon request would in our view be an unjustifiable expense and would not measurably add to public knowledge regarding NRC proceedings. In addition, the Comptroller General has recently ruled that Section 502 of the 1981 fiscal year Energy and Water Development Appropriations Act (P.L. 96-367) prohibits the NRC from providing certain documents and transcripts free of charge to nonapplicant parties in adjudicatory proceedings. *See* CG Opinion No. B-200585 (Dec. 3, 1980). Certain aspects of the petitioners proposal would, therefore, seem to be contrary to the Comptroller General's position.

In its present form, 10 CFR 2.715(b) requires that the Secretary of the Commission serve notices of hearing upon all persons requesting such notices. In practice, a docket list relating to each nuclear licensing matter is compiled by the Office of the Secretary, and persons on this list receive direct notification of all hearings, pre-hearing conferences, oral arguments, and other formal proceedings associated with that docket. Hence, a portion of petitioner's request in this area is already current practice. Moreover, all formal proceedings are noticed in the *Federal Register* and in local publications selected as being those reasonably calculated to provide the widest notice to the largest number of potentially affected people. Every effort, including advertisements and press releases, is made to notify the public of informal public meetings to be held near the site. We conclude that the thrust of petitioner's proposal to amend 10 CFR 2.715(b) as to the furnishing of notices of hearings is fully satisfied by current NRC rules and procedures.

(4) Location and Scheduling of Hearings and NRC-Licensee/Applicant Meetings.

Finally, petitioner requests that all hearings and NRC-licensee/applicant meetings be held at a site and at times which will maximize attendance by a majority of persons potentially affected. Most adjudicatory hearings are already held near the relevant nuclear reactor site, usually in the nearest sizeable city or town. Appellate oral arguments in adjudicatory proceedings are, however, generally heard in the Washington, D.C. area. Public meetings of a more informal nature are also ordinarily held near the site, particularly when they involve issues relating to the nuclear plant. Meetings with licensees or applicants may be held either near the plant or reactor site or in Washington, depending on the nature of the meeting, the convenience to the parties involved, and urgency of the meeting. Except for informal contacts between the NRC staff and the licensee or applicant (telephone coversations, discussions during site visits, etc.), NRC staff-licensee/applicant meetings are generally open to the public and are announced in advance. As noted above, this "Open Meetings" policy is fully described in a policy statement issued on June 28, 1978. 43 FR 28058. It provides that "All meetings conducted by the NRC technical staff as part of its review of a particular domestic license or permit application (including an application for an amendment to a license or permit) will be open to attendance by all parties or petitioners for leave to intervene in the case." The scheduling and location of such meetings is arranged, where possible, with the intent of allowing all interested parties to attend. This has, in some instances, resulted in meetings being held outside of normal business hours, as petitioner appears to suggest.

Requiring that all formal and informal hearings and meetings normally be held proximate to the actual or proposed nuclear plant site would not, in most cases, increase public convenience. Such reactor sites are not located, for obvious reasons, in high population density areas. Most interested members of the public would reside in a nearby population center, and this is where hearings are held under the current system. Furthermore, reactor sites are simply not properly equipped to accommodate large public meetings. We find no demonstrable merit in the petitioner's suggested change to our rules, and we will continue the current practice of holding the majority of adjudicatory hearings and public meetings in a city or town near the operating or proposed power plant site.

As to the petitioner's second point, most adjudicatory hearings are held during normal business hours. However, evening or even weekend sessions are occasionally held to permit intervenors to participate if they are unable to do so during business hours. Such sessions may also be held to hear statements offered by non-parties participating pursuant to 10 CFR 2.715(a). Special informal public meetings, such as those at Three Mile Island, are usually held during non-business hours.

We decline to accept the petitioner's approach for the timing of hearings and meetings, which implies that evenings and weekends should be preferred, as a matter of course, to regular business hours. Where the basic purpose of an informal meeting is to inform the public (as with meetings at Three Mile Island), evening hours have been used frequently, and we expect such practice to continue. Normal business hours, however, are more appropriate for the conduct of agency business in formal proceedings or in official meetings with the licensee or applicant. There are simply more hours and days available for the conduct of business if normal working hours are utilized. Licensing hearings can be a lengthy process, and would be even more time-consuming if petitioner's suggestion were adopted. We conclude that the efficient conduct of hearings and meetings requires that they generally take place during normal business hours on weekdays. As we have noted above, special arrangements are often made to accommodate members of the public wishing to appear at the hearing but unable to do so during business hours.

For the foregoing reasons, the Commission denies the petition for rulemaking filed by the Citizens Advisory Board. A copy of the Commission's letter of denial is available for public inspection and copying at the NRC Public Document Room, 1717 H Street N.W., Washington, D.C.

For the Nuclear Regulatory Commission

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.,
this 20th day of April, 1981.

Cite as 13 NRC 443 (1981) **CLI-81-6**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

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

(Petition for Relief Under 10 CFR 2.206)

PACIFIC GAS AND ELECTRIC CO.
(Diablo Canyon Nuclear Power Plant, Units 1 and 2)

May 8, 1981

The Commission affirms Part I of DD-81-3, 13 NRC 351 (1981), in which the Director of the Office of Nuclear Reactor Regulation denied a request under 10 CFR 2.206 to supplement the environmental record in this operating license proceeding on the ground that the requesting party had raised that same issue before the Licensing Board and no decision had yet been reached. The Commission also declines review of Parts II and III of the Director's Decision.

RULES OF PRACTICE: SHOW CAUSE PROCEEDING

Parties must be prevented from using 10 CFR 2.206 procedures for avoiding an existing forum in which the issues raised more logically should be presented. *Consolidated Edison Co. of New York* (Indian Point, Units 1, 2 and 3), CLI-75-8, 2 NRC 173, 177 (1975). Where an adjudicatory Board is presiding in a proceeding with jurisdiction to consider the matter, a party to that proceeding may not choose to avoid that forum by use of 10 CFR 2.206.

RULES OF PRACTICE: SHOW CAUSE PROCEEDING

If an adjudicatory Board has jurisdiction to consider an issue, a party to the proceeding before that Board must first seek relief from the Board. If a Board is clearly without jurisdiction, there is no need to present the matter to the Board for decision before seeking to institute a show cause proceeding under 10 CFR 2.206. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4), CLI-79-5, 9 NRC 607 (1979); *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 2), ALAB-579, 11 NRC 223, 226 (1980), *id.* DD-80-33, 12 NRC 598 (1980).

MEMORANDUM AND ORDER

On October 17, 1980, the Joint Intervenors in the NRC's proceeding to decide whether and on what conditions the power reactors at Diablo Canyon should operate filed a request under 10 CFR 2.206 with Harold Denton, Director of the Office of Nuclear Reactor Regulation, to supplement the environmental record in that proceeding so as to include a detailed evaluation of the environmental consequences of a catastrophic accident at that site. The Director denied the requested relief, in part because the Joint Intervenors had raised that same issue before the Licensing Board and no decision has been reached. DD-81-3, Part I, 13 NRC 351 (March 26, 1981). Because of the importance of this ruling to the Commission's administrative practice, the Commission specifically endorses the principle that 10 CFR 2.206 should not be used by a party to a licensing proceeding to request relief on a matter within the jurisdiction of the presiding officer in the proceeding. The Commission declines review of Parts II and III and, as a matter of course, expresses no views on the merits of those aspects of the Decision.

I

In May 1979, Joint Intervenors filed a motion to reopen and supplement the environmental record with regard to catastropic (hereinafter Class 9) accidents. This motion was based on alleged new information that called into question the validity of the prior NRC position on Class 9 accident considerations under the National Environmental Policy Act of 1969 (NEPA), 42 U.S.C. 4321 *et seq.* — principally the TMI-2 accident and the publication of the Risk Assessment Review Group (Lewis) Report. In addition, the Joint Intervenor called special attention to the issue whether the facility could comply with NRC seismic criteria in requesting additional consideration of Class 9 accidents. On motion of the NRC staff, the Board deferred consideration of Intervenors' request pending completion of the staff's evaluation of impacts of the TMI-2 accident for Diablo Canyon. In its most recent order considering those impacts, the Board continued to defer ruling on the motion to reopen until after the Appeal Board rules on the seismic issue. Prehearing Conference Order, at 3, 26-27 (February 13, 1981). The Joint Intervenors' motion must eventually be ruled upon by the Licensing Board before it completes its consideration of the Diablo Canyon license.

In October 1979, the Friends of the Earth (FOE) requested a supplement to the Diablo Canyon environmental record and the environmental record for two other facilities on Class 9 accidents, for almost identical reasons. In January 1980, before staff action was completed on this 10 CFR 2.206 petition, FOE sued the NRC in federal district court to compel the preparation of the requested supplement. *Friends of the Earth, et al.* v. *NRC, et al.*, N.D. Cal. No. C-80-0234-SW (*filed* January 30, 1980). In June, the Director, NRR, for the NRC staff, denied relief both because of the pendency of the same issue in the Diablo Canyon licensing proceeding and because FOE failed to make a suitable case to justify relief on the merits. *Arizona Public Service Co.* (Palo Verde Nuclear Generating Station, Units 1, 2 and 3), *et al.*, DD-80-22, 11 NRC 919 (1980). The Commission permitted the review time to expire without taking review. 10 CFR 2.206(c). No judicial review was sought for that decision.

In August 1980, the district court granted the NRC's and the applicant's motions to dismiss the *FOE* lawsuit. In an order entered in September, the district court explained that, because of the pendency of that issue in the Diablo Canyon proceeding and because the NRC had not yet decided the issue, "[j]udicial intervention now would interfere with the agency's consideration of Diablo Canyon, and would risk inconsistent results, likely produce duplication of effort and unnecessary expenditure of judicial resources, and encourage bypassing of congressionally sanctioned administrative procedures." *FOE* v. *NRC*, slip op. at 3, 11 *Environmental Rptr.* (BNA) 1035 (1980). "What the court finds compelling arises from the fact that the very issues underlying plaintiffs' concerns in this case, the issues which plantiffs argue deserve further study and analysis, are in fact presently being subjected to intensive structured investigation by the federal defendants, complete with the opportunity for public comment and criticism as well as subsequent judicial review." *Id.* at 2.

II

Notwithstanding this clear judicial endorsement of the Licensing Board proceeding as the proper forum in which to resolve this issue, Joint

Intervenors, in October 1980, filed the instant request for enforcement relief under 10 CFR 2.206, updating and restating their original legal theory and assertion of "special circumstances" warranting special treatment of Class 9 accidents. In the Commission's view, the Director properly denied relief in Part I of his Decision based on the pendency of the identical matter before the Licensing Board. In affirming that part of the Director's Decision, the Commission expressly adopts the view intimated six years ago in *Indian Point,* that parties must be prevented from using 10 CFR 2.206 procedures for avoiding an existing forum in which the issues raised more logically should be presented. *Consolidated Edison Co. of New York* (Indian Point, Units 1, 2 and 3), CLI-75-8, 2 NRC 173, 177 (1975). Where a Board is presiding in a proceeding with jurisdiction to consider the matter, a party to that proceeding may not choose to avoid that forum by use of 10 CFR 2.206.

This principle applies most strongly in a situation, like this case, in which petitioners attempt to raise under 10 CFR 2.206, the same issue on the same theory that "awaits the Board's consideration in the operating license proceedings." DD-81-3, 13 NRC, at 351. As noted above, the Licensing Board must decide the issue before it completes its action on Diablo Canyon. That "[t]he Board determined it will defer consideration of this motion [to reopen and supplement the record on Class 9 accidents] until the Appeal Board has ruled on the seismic issues before it" is no justification for attempting to circumvent the Board process for deciding the terms and conditions of contested license issuances. If anything, the Board's action holds open the prospect of relief based on the very issue raised in the Intervenors' motion. For these reasons, the Commission specifically affirms that Part of the Director's Decision which relies on these principles. The Commission does not elect to review any other aspect of the Decision.

This decision should not be taken as any weakening of the Commission's endorsement of the procedures available under 10 CFR 2.206. Those procedures permit any person to request enforcement action, if available, at any facility. The ruling today means that if a Board has jurisdiction to consider the issue, a party to the proceeding before that Board must first seek relief from the Board. If a Board is clearly without jurisdiction, there is no need to present the matter to the Board for decision. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4), CLI-79-5, 9 NRC 607 (1979). *See Virginia Electric & Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-551, 9 NRC 704, 708-09 (1979). Indeed, it is NRC practice that such motions are referred for action under 10 CFR 2.206. *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 2), ALAB-579, 11 NRC 223, 226 (1980), *id.* DD-80-33, 12 NRC 598 (1980).

That portion of DD-81-3 is, accordingly, affirmed.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, DC,
this 8th day of May, 1981.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS*

Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **Docket Nos. 50-247**
50-286

CONSOLIDATED EDISON
COMPANY OF NEW YORK, INC.
(Indian Point, Unit No. 2)

POWER AUTHORITY OF
THE STATE OF NEW YORK
(Indian Point, Unit No. 3) **May 12, 1981**

The Commission orders that the Indian Point operating licenses be modified, allowing the continued use of open-cycle cooling at the units in accordance with a settlement agreement approved by the Environmental Protection Agency.

EPA AUTHORITY: NUCLEAR PLANT COOLING SYSTEMS

By the terms of the Clean Water Act and Commission precedent, the NRC must defer to final decisions of the EPA with respect to the type of cooling system employed by nuclear power plants. 33 U.S.C. §1371(c)(2); *Public Service Company of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-78-1, 7 NRC 1 (1978); *Philadelphia Electric Company* (Peach Bottom Atomic Power Station, Unit 3), ALAB-532, 9 NRC 279 (1979).

*Chairman Hendrie disqualified himself from participation in this case.

ORDER

On November 15, 1978, the Commission on its own motion took review of the Appeal Board's decision in ALAB-487, and requested briefs from the parties. No party had requested review. The purpose of taking review was to clarify the status of license conditions requiring termination of once-through cooling at the Indian Point Unit 2 and Unit 3 facilities, in light of the pendency of an adjudicatory proceeding before the Environmental Protection Agency to determine the type of cooling system to be required of those two units and two fossil-fired generating plants located along the Hudson River. It is well established, by the terms of the Clean Water Act and Commission precedent, that the NRC must defer to final decisions of the EPA with respect to the type of cooling system to be employed by nuclear power plants. 33 U.S.C. §1371(c)(2); *Public Service Company of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-78-1; *Philadelphia Electric Company* (Peach Bottom Atomic Power Station, Unit 3), ALAB-532, 9 NRC 279 (1979). This case, however, presented the question of the status of NRC decisions with respect to the type of cooling system for a nuclear power plant while awaiting a final decision from EPA.

That legal issue has now been mooted by the execution of a settlement agreement that terminates the EPA proceeding with a decision permitting continued use of open-cycle cooling at the Indian Point units, coupled with a variety of measures to mitigate the adverse effects of the plants on the biota in the Hudson River and otherwise to benefit the environment of the area.

On February 27, 1981, Consolidated Edison Company of New York and the Power Authority of the State of New York, licensees of Units 2 and 3 respectively, filed a motion requesting that we issue an order (1) deleting the requirement, embodied in the license condition 2.E of licenses DPR-26 and DPR-64, that they terminate operation with once-through cooling; (2) directing that the settlement agreement supersedes and nullifies the stipulation, executed in 1975, requiring termination of once-through cooling at Indian Point Unit 3; and (3) directing the completion of all other action required under the Atomic Energy Act to make the licenses consistent with the settlement agreement relative to the condenser cooling systems for the two units.

With respect to the first two of these requests, all parties are in agreement that the relief requested should be granted. It should be noted that there is disagreement among the parties as to the legal basis for that grant of relief. The licensees, the New York State Department of Environmental Conservation, and the NRC staff agree that this action is compelled by Section 511(c)(2) of the Clean Water Act, 33 U.S.C. §1371(c)(2). The Hudson River

Fishermen's Association and Save Our Stripers, on the other hand, assert that EPA's action is not binding on the Commission, because the NRC license condition was imposed prior to EPA's taking final action with respect to the type of cooling system for Indian Point. HRFA/SOS urges that the Commission grant the requested relief, but that it premise the relief on the provision, stated expressly in the license condition, that permits the licensees to seek relief from the requirement that once-through cooling be terminated by showing data developed during the period of once-through operation. We cannot accept this suggestion, since it is patent to us that this agency is bound to follow final EPA decisions on water quality impacts, irrespective of whether they occur before or after NRC decisions on the same subject. However, we would note that the provision of the license condition cited by HRFA/SOS does indeed provide an alternative legal basis for granting the requested relief.

With respect to the licensees' third request, we are uncertain what further relief the licensees envision as being necessary with respect to the condenser cooling systems, beyond the purely ministerial alterations in the license required to conform the license to the settlement agreement, i.e., deletion of the requirement for termination of once-through cooling, and, once the State of New York has issued discharge permits pursuant to the settlement agreement, deletion of those portions of the license which are superseded by the terms and conditions of the discharge permits and the settlement agreement. Should further relief of a substantive nature be required, the licensees are at liberty to address to us a request for such further specific relief.

Accordingly, the Commission orders, effective upon the effective date of the settlement agreement, as set forth in Section 4.M.1 of the agreement, that:

1. Condition 2.E is deleted from licenses DPR-26 and DPR-64;
2. the stipulation entered into in the Indian Point Unit 3 proceeding on January 13, 1975 is superseded and nullified;
3. the NRC staff is directed to take such ministerial actions as may be needed to conform the Indian Point Unit 2 and Unit 3 licenses to the terms and conditions of this order and the settlement agreement.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 12th day of May, 1981.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of

STATEMENT OF POLICY ON CONDUCT OF LICENSING PROCEEDINGS **May 20, 1981**

The Commission issues a policy statement providing guidance to its licensing boards on the use of tools intended to reduce the time for completing licensing proceedings while still ensuring that hearings are fair and produce full records.

I. BACKGROUND

The Commission has reviewed the docket of the Atomic Safety and Licensing Board Panel (ASLBP) and the current status of proceedings before its individual boards. In a series of public meetings, the Commission has examined at length all major elements in its licensing procedure. It is clear that a number of difficult problems face the agency as it endeavors to meet its responsibilities in the licensing area. This is especially the case with regard to staff reviews and hearings, where requested, for applications for nuclear power plant operating licenses.

Historically, NRC operating licensing reviews have been completed and the license issued by the time the nuclear plant is ready to operate. Now, for the first time the hearings on a number of operating license applications may not be concluded before construction is completed. This situation is a consequence of the Three Mile Island (TMI) accident, which required a reexamination of the entire regulatory structure. After TMI, for over a year and a half, the Commission's attention and resources were focused on plants which were already licensed to operate and on the preparation of an action plan which specified changes necessary for reactors as a result of the accident.

Although staff review of pending license applications was delayed during this period, utilities which had received construction permits continued to build the authorized plants. The staff is now expediting its review of the applications and an unprecedented number of hearings are scheduled in the next 24 months. Many of these proceedings concern applications for operating licenses. If these proceedings are not concluded prior to the completion of construction, the cost of such delay could reach billions of dollars. The Commission will seek to avoid or reduce such delays whenever measures are available that do not compromise the Commission's fundamental commitment to a fair and thorough hearing process.

Therefore, the Commission is issuing this policy statement on the need for the balanced and efficient conduct of all phases of the hearing process. The Commission appreciates the many difficulties faced by its boards in conducting these contentious and complex proceedings. By and large, the boards have performed very well. This document is intended to deal with problems not primarily of the boards' own making. However, the boards will play an important role in resolving such difficulties.

Individual adjudicatory boards are encouraged to expedite the hearing process by using those management methods already contained in Part 2 of the Commission's Rules and Regulations. The Commission wishes to emphasize though that, in expediting the hearings, the board should ensure that the hearings are fair, and produce a record which leads to high quality decisions that adequately protect the public health and safety and the environment.

Virtually all of the procedural devices discussed in this Statement are currently being employed by sitting boards to varying degrees. The Commission's reemphasis of the use of such tools is intended to reduce the time for completing licensing proceedings. The guidelines set forth below are not to be considered all inclusive, but rather are to be considered illustrative of the actions that can be taken by individual boards.

II. GENERAL GUIDANCE

The Commission's Rules of Practice provide the board with substantial authority to regulate hearing procedures. In the final analysis, the actions, consistent with applicable rules, which may be taken to conduct an efficient hearing are limited primarily by the good sense, judgment, and managerial skills of a presiding board which is dedicated to seeing that the process moves along at an expeditious pace, consistent with the demands of fairness.

Fairness to all involved in NRC's adjudicatory procedures requires that every participant fulfill the obligations imposed by and in accordance with applicable law and Commission regulations. While a board should endeavor to conduct the proceeding in a manner that takes account of the special circumstances faced by any participant, the fact that a party may have personal or other obligations or possess fewer resources than others to devote to the proceeding does not relieve that party of its hearing obligations. When a participant fails to meet its obligations, a board should consider the imposition of sanctions against the offending party. A spectrum of sanctions from minor to severe is available to the boards to assist in the management of proceedings. For example, the boards could warn the offending party that such conduct will not be tolerated in the future, refuse to consider a filing by the offending party, deny the right to cross-examine or present evidence, dismiss one or more of the party's contentions, impose appropriate sanctions on counsel for a party, or, in severe cases, dismiss the party from the proceeding. In selecting a sanction, boards should consider the relative importance of the unmet obligation, its potential for harm to other parties or the orderly conduct of the proceeding, whether its occurrence is an isolated incident or a part of a pattern of behavior, the importance of the safety or environmental concerns raised by the party, and all of the circumstances. Boards should attempt to tailor sanctions to mitigate the harm caused by the failure of a party to fulfill its obligations and bring about improved future compliance. At an early stage in the proceeding, a board should make all parties aware of the Commission's policies in this regard.

When the NRC staff is responsible for the delay of a proceeding the Chief Administrative Judge, Atomic Safety and Licensing Board Panel, should inform the Executive Director for Operations. The Executive Director for Operations will apprise the Commission in writing of significant delays and provide an explanation. This document will be served on all parties to a proceeding and the board.

III. SPECIFIC GUIDANCE

A. Time

The Commission expects licensing boards to set and adhere to reasonable schedules for proceedings. The Boards are advised to satisfy themselves that the 10 CFR 2.711 "good cause" standard for adjusting times fixed by the Board or prescribed by Part 2 has actually been met before granting an extension of time. Requests for an extension of time should generally be in writing and should be received by the Board well before the time specified expires.

B. Consolidated Intervenors

In accordance with 10 CFR 2.715a, intervenors should be consolidated and a lead intervenor designated who has "substantially the same interest that may be affected by the proceedings and who raise[s] substantially the same questions .... " Obviously, no consolidation should be ordered that would prejudice the rights of any intervenor.

However, consonant with that condition, single, lead intervenors should be designated to present evidence, to conduct cross-examination, to submit briefs, and to propose findings of fact, conclusions of law, and argument. Where such consolidation has taken place, those functions should not be performed by other intervenors except upon a showing of prejudice to such other intervenors' interest or upon a showing to the satisfaction of the board that the record would otherwise be incomplete.

C. Negotiation

The parties should be encouraged to negotiate at all times prior to and during the hearing to resolve contentions, settle procedural disputes, and better define issues. Negotiations should be monitored by the board through written reports, prehearing conferences, and telephone conferences, but the boards should not become directly involved in the negotiations themselves.

D. Board Management of Discovery

The purpose of discovery is to expedite hearings by the disclosure of information in the possession of the parties which is relevant to the subject matter involved in the proceeding so that issues may be narrowed, stipulated, or eliminated and so that evidence to be presented at hearing can be stipulated or otherwise limited to that which is relevant. The Commission is concerned that the number of interrogatories served in some cases may place an undue burden on the parties, particularly the NRC staff, and may, as a consequence, delay the start of the hearing without reducing the scope or the length of the hearing.

The Commission believes that the benefits now obtained by the use of interrogatories could generally be obtained by using a smaller number of better focused interrogatories and is considering a proposed rule which would limit the number of interrogatories a party could file, absent a ruling by the Board that a greater number of interrogatories is justified. Pending a Commission decision on the proposed rule, the Boards are reminded that they may limit the number of interrogatories in accordance with the Commission's rules.

Accordingly, the boards should manage and supervise all discovery, including not only the intitial discovery directly following admission of contentions, but also any discovery conducted thereafter. The Commission again endorses the policy of voluntary discovery, and encourages the boards, in consultation with the parties, to establish time frames for the completion of both voluntary and involuntary discovery. Each individual board shall determine the method by which it supervises the discovery process. Possible methods include, but are not limited to, written reports from the parties, telephone conference calls, and status report conferences on the record. In virtually all instances, individual boards should schedule an initial conference with the parties to set a general discovery schedule immediately after contentions have been admitted.

E. Settlement Conference

Licensing boards are encouraged to hold settlement conferences with the parties. Such conferences are to serve the purpose of resolving as many contentions as possible by negotiation. The conference is intended to: (a) have the parties identify those contentions no longer considered valid or important by their sponsor as a result of information generated through discovery, so that such contentions can be eliminated from the proceeding; and (b) to have the parties negotiate a resolution, wherever possible, of all or part of any contention still held valid and important. The settlement conference is not intended to replace the prehearing conferences provided by 10 CFR 2.751a and 2.752.

F. Timely Rulings on Prehearing Matters

The licensing boards should issue timely rulings on all matters. In particular, rulings should be issued on crucial or potentially dispositive issues at the earliest practicable juncture in the proceeding. Such rulings may eliminate the need to adjudicate one or more subsidiary issues. Any ruling which would affect the scope of an evidentiary presentation should be rendered well before the presentation in question. Rulings on procedural matters to regulate the course of the hearing should also be rendered early.

If a significant legal or policy question is presented on which Commission guidance is needed, a board should promptly refer or certify the matter to the Atomic Safety and Licensing Appeal Board or the Commission. A board should exercise its best judgment to try to anticipate crucial issues which may require such guidance so that the reference or certification can be made and the response received without holding up the proceeding.

G. Summary Disposition

In exercising its authority to regulate the course of a hearing, the boards should encourage the parties to invoke the summary disposition procedure on issues where there is no genuine issue of material fact so that evidentiary hearing time is not unnecessarily devoted to such issues.

H. Trial Briefs, Prefiled Testimony Outlines and Cross-Examination Plans

All or any combination of these devices should be required at the discretion of the board to expedite the orderly presentation by each party of its case. The Commission believes that cross-examination plans, which are to be submitted to the board alone, would be of benefit in most proceedings. Each board must decide which device or devices would be most fruitful in managing or expediting its proceeding by limiting unnecessary direct oral testimony and cross-examination.

I. Combining Rebuttal and Surrebuttal Testimony

For particular, highly technical issues, boards are encouraged during rebuttal and surrebuttal to put opposing witnesses on the stand at the same time so that each witness will be able to comment immediately on an opposing witness' answer to a question. Appendix A to 10 CFR Part 2 explicitly recognizes that a board may find it helpful to take expert testimony from witnesses on a round-table basis after the receipt in evidence of prepared testimony.

J. Filing of Proposed Findings of Fact and Conclusions of Law

Parties should be expected to file proposed findings of fact and conclusions of law on issues which they have raised. The boards, in their discretion, may refuse to rule on an issue in their initial decision if the party raising the issue has not filed proposed findings of fact and conclusions of law.

K. Initial Decisions

Licensing proceedings vary greatly in the difficulty and complexity of issues to be decided, the number of such issues, and the size of the record compiled. These factors bear on the length of time it will take the boards to issue initial decisions. The Commission expects that decisions not only will continue to be fair and thorough, but also that decisions will issue as soon as practicable after the submission of proposed findings of fact and conclusions of law.

Accordingly, the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel should schedule all board assignments so that after the record has been completed individual Administrative Judges are free to write initial decisions on those applications where construction has been completed. Issuance of such decisions should take precedence over other responsibilities.

IV. CONCLUSION

This statement on adjudication is in support of the Commission's effort to complete operating license proceedings, conducted in a thorough and fair manner, before the end of construction. As we have noted, that process has not, in the past, extended beyond completion of plant construction. Because of the considerable time that the staff had to spend on developing and carrying out safety improvements at operating reactors during 1979-1980, in the wake of the Three Mile Island accident, this historical situation has been disrupted. To reestablish it on a reliable basis requires changes in the agency review and hearing process, some of which are the subject of this statement.

As a final matter, the Commission observes that in ideal circumstances operating license proceedings should not bear the burden of issues that ours do now. Improvement on this score depends on more complete agency review and decision at the construction permit stage. That in turn depends on a change in industrial practice: submittal of a more nearly complete design by the applicant at the construction permit stage. With this change operating license reviews and public proceedings could be limited essentially to whether the facility in question was constructed in accordance with the detailed design approved for construction and whether significant developments after the date of the construction permit required modifications in the plant.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 20th day of May, 1981.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **45 Fed. Reg. 65521-38 (October 3, 1980)**

URANIUM MILL LICENSING REQUIREMENTS (10 CFR Parts 30, 40, 70 & 150) **May 26, 1981**

The Commission denies petitions filed by three operators of uranium mills and the State of New Mexico to stay the Commission's Uranium Mill Licensing Requirements issued pursuant to the Uranium Mill Tailings Radiation Control Act of 1978.

RULES OF PRACTICE: STAY OF AGENCY ACTION

In ruling on a motion for a stay, the Commission considers the following four factors: (1) whether petitioners will suffer irreparable harm without a stay; (2) whether petitioners are likely to prevail on the merits; (3) whether other interested parties would not be substantially harmed by a stay; and (4) whether the public interest supports a stay. *Virginia Petroleum Jobbers Ass'n* v. *FPC,* 259 F.2d 921, 925 (D.C. Cir. 1958); *Washington Metropolitan Area Transit Commission* v. *Holiday Tours,* 559 F.2d 841, 843-44 (D.C. Cir. 1977).

RULES OF PRACTICE: STAY OF AGENCY ACTION (IRREPARABLE INJURY)

The expense of an administrative proceeding is not usually considered irreparable injury. *Meyers* v. *Bethlehem Shipbuilding Corp.,* 303 U.S. 41, 51

(1938); *Hornblower & Weeks-Hemphill Noyes, Inc.* v. *Csaky,* 427 F. Supp. 814 (S.D.N.Y. 1977).

NRC: ENFORCEMENT OF HEALTH AND ENVIRONMENTAL STANDARDS FOR URANIUM MILL TAILINGS

The Atomic Energy Act of 1954, as amended, authorizes the Administrator of the Environmental Protection Agency to issue health and environmental standards for the protection of the public health, safety, and the environment from radiological and non-radiological hazards associated with the processing, possession, transfer, and disposal of uranium mill tailings at active processing sites or disposal sites. Section 275.b.(1). Once the EPA standards are issued and become effective, the NRC is reponsible for their implementation and enforcement in non-Agreement States through the conduct of licensing activities under the Act. Section 275.d. In Agreement States, the implementation and enforcement of EPA's standards is the responsibility of the Agreement States. Sections 2.75.d and 274.o.

MEMORANDUM AND ORDER

On April 13, 1981, three operators of uranium mills (Operators) petitioned the Nuclear Regulatory Commission (NRC) to stay the Uranium Mill Licensing Requirements[1] issued by the NRC pursuant to the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA).[2] Operators state that their motion was precipitated by the NRC's recent correspondence with Agreement States[3] requesting that they adopt the NRC's Uranium Mill Licensing requirements as "minimum national standards" as contemplated by UMTRCA.[4] Operators contend that because this NRC letter allegedly requires Agreement State compliance by August 1, 1981, they will be irreparably injured by a lack of opportunity to participate in state proceedings to adopt regulations for uranium milling. They also claim among other things that the NRC's regulations are unlawful because they were issued before EPA promulgated health and safety standards pursuant

[1]45 *Fed. Reg.* 65521-38 (October 3, 1980).
[2]Pub. L. 95-604, 92 Stat. 3021-3043 (November 8, 1978).
[3]Agreement States are states which have entered into an agreement with the NRC for the purpose of discontinuing NRC regulatory authority over certain classes of radioactive material and the assumption of that regulatory authority by the State. See, generally, Section 274 of the Atomic Energy Act of 1954, as amended.
[4]These Operators and the American Mining Congress previously petitioned the United States Court of Appeals for the Tenth Circuit to review these requirements. *Kerr-McGee Corporation, et al.* v. *NRC* (Nos. 80-2043, 80-2229, 80-2269, and 80-2271). These consolidated petitions are still pending.

to UMTRCA and that the public interest will be adequately protected by current state regulation of uranium milling.

The State of New Mexico has also requested that the effective date of the requirements be stayed until six months after the final court decision in *Kerr-McGee Corporation, et al.* v. *NRC.* New Mexico appears to contend that unless a stay is granted it may be irreparably injured if it is required to hold two hearings on the adoption of milling regulations: one to comply with the current regulations and a second to comply with any changes which might result from the litigation. New Mexico also claims that its current regulations are adequate to protect the public health and safety in the interim.

Subsequently, on April 27, 1981 Operators moved for a stay in the U.S. Court of Appeals for the Tenth Circuit (Motion), and on May 6, 1981 New Mexico also moved that Court for a stay. To the extent that those motions amplify or clarify the movants' arguments before the Commission, they are responded to in the Commission's filing "Respondents' Opposition To Motion For A Stay" (Opposition) which is incorporated by reference herein.

For the reasons discussed below, we find no merit in petitioners' contentions. Accordingly, the stay requests are denied.

The NRC's Uranium Mill Licensing Requirements establish a comprehensive regulatory regime for protection of the public health and safety and the environment from hazards associated with uranium milling and the large quantities of mill tailings that generate. The Commission's authority to establish this regulatory regime is provided by UMTRCA. It was enacted by Congress in recognition of the hazards presented by uranium milling and the need to protect the public from these hazards by authorizing the NRC to regulate uranium milling and to set minimum national standards for the regulation of uranium milling by those Agreement States who wish to continue regulatory uranium milling activities.[5] In order to give those Agreement States time to upgrade their regulation of uranium milling activities, UMTRCA provides the Agreement States until November 8, 1981 to modify their regulations to include the minimum national standards. Section 204(h)(1) as amended, 93 Stat. 799-800 (November 9, 1979).

The NRC's Uranium Milling Licensing Requirements were developed over a period of over four years during which time the NRC solicited and received substantial public comments. On June 3, 1976 the NRC announced the initiation of the preparation of a Generic Environmental

[5]H.R. Rep. No. 95-1480, Part I, 95th Cong., 2d Sess. 13 (1978).

Impact Statement (GEIS) on uranium milling. A draft GEIS was published for public comment in April 1979; and a proposed rule was issued on August 24, 1979.[6] Subsequently, the NRC held public hearings in Denver and Albuquerque in October, 1979. In addition, 99 written comments were received. In September, 1980, the NRC issued the Final GEIS, and on October 3, 1980 issued the final rule. As is discussed below, to the extent the Operators' motion is based on comments made during the rulemaking process the Statement of Consideration accompanying the final rule addressed them.

In ruling on a motion for a stay, the Commission considers the following four factors: (1) whether petitioners will suffer irreparable harm without a stay; (2) whether petitioners are likely to prevail on the merits; (3) whether other interested parties would not be substantially harmed by a stay; and (4) whether the public interest supports a stay. *Virginia Petroleum Jobbers Ass'n.* v. *FPC*, 259 F.2d 921, 925 (D.C. Cir. 1958); *Washington Metropolitan Area Transit Commission* v. *Holiday Tours*, 559 F.2d 841, 843-44 (D.C. Cir. 1977).[7] Petitioners' conclusory filings utterly fail to sustain their burden on any of these factors.

I. Irreparable Injury

Operators contend they will suffer irreparable injury if the regulations are not stayed. They allege that they will not have an opportunity to participate meaningfully in any rulemaking proceeding initiated by an Agreement State for its adoption of federal regulations pursuant to UMTRCA because the NRC has set an August 1, 1981 deadline for Agreement State compliance. They also allege that they will not have an opportunity to participate meaningfully in any proceeding initiated by EPA for the establishment of general standards. Finally, they claim that they will sustain heavy expenses and unnecessary disruption to their operations if the stay is not granted.

UMTRCA provides that if an Agreement State wants to regulate uranium milling after November 8, 1981, then that State by that date shall require compliance with standards for the protection of the public health, safety, and the environment from hazards associated with such material

[6]44 Fed. Reg. 50020-50022.

[7]Although petitioners contend that the four-fold test is not appropriate for an agency's analysis of a request for a stay, the only caselaw relied on is *Holiday Tours*, *supra*. As petitioners assert, that case supports the proposition that administrative action may be stayed upon a showing of substantial question regarding its validity, *i.e.*, upon a showing of substantial probability of success on the merits and a need to minimize hardship, *i.e.*, irreparable injury. Thus, there is no merit to the petitioners' argument questioning the applicability of the well-established four-fold test of request for a stay.

which are equivalent, to the extent practicable, or more stringent than standards adopted and enforced by the Commission for the same purpose.[8] Section 274.o.(3) of the Atomic Energy Act of 1954, as amended, (Act) provides that in the case of rulemaking on uranium mill regulations Agreement States shall provide an opportunity for public participation through written comments or a public hearing. Section 274.o. of the Act also provides that no State shall be required under Section 274.o.(3) to conduct proceedings concerning any regulation which would duplicate proceedings conducted by the Commission.

These provisions show that the Act only requires States to provide an opportunity for public comments on uranium milling rules, it leaves details of public participation to State law. Operators have neither informed us how the various Agreement States will implement the opportunity for public comment[9] nor provided us with support for their contention that such implementation will require more time than currently available. However, some Agreement States which intend to modify their agreements to include the regulation of uranium milling have provided the NRC with timetables for implementing those modifications. Those timetables include a period for public participation consistent with the States meeting a November 8, 1981 deadline for completing the processes for modifying the agreements.

Moreover, it has been manifestly clear since the enactment of UMTRCA in 1978 that Agreement States have until November 8, 1981 to implement uranium mill tailings regulations at least equivalent to those issued by the NRC. The only new circumstance referred to by the Operators is the NRC's recent letter to Agreement States suggesting that they submit completed applications for amendment of their agreements by August 1, 1981. These letters cannot reasonably be characterized as containing a threat to revoke Agreement State regulatory authority over uranium mill tailings on August 1, 1981. Thus, we find no merit in Operators' argument that they will not have an adequate opportunity to participate in State proceedings to adopt uranium milling regulations.[10]

Section 275.c.(1) of the Act establishes a procedure for EPA's promulgation of standards generally applicable to uranium mill tailings. The procedure includes an opportunity to present written comments and to participate in an oral hearing. Operators have not provided any explanation of how the NRC's promulgation of uranium milling regulations adversely

[8]Pub. L. 95-604, Sections 204(e)(h), 92 Stat. 3034-38 (Nov. 8, 1978).

[9]The NRC has suggested that under Sec. 274.o the states could rely on the record compiled by NRC because the states need not duplicate NRC proceedings.

[10]For the reasons given in the Commission's Opposition, this conclusion is not changed by Operators' elaboration of this argument as contained in their Motion.

affects their ability to participate in EPA proceedings. Moreover, petitioners have not explained why they have raised this issue now several months after it was clear that the NRC's regulations would precede EPA's standards.

Finally, Operators offer no factual support for their bald allegations that failure to grant their request for a stay will result in heavy expenses and unnecessary disruption to their operations. The Commission cannot grant a stay on the basis of mere claims unsupported by facts.[11] For these reasons, and those in the Commission's opposition, the Commission finds that denial of the request for a stay will not irreparably injure the Operators.

New Mexico's claim of irreparable injury is similarly without merit. New Mexico has not provided the Commission with any reasons to doubt the validity of the requirements. Therefore, New Mexico's claim of injury from the possible need to hold a second hearing is purely speculative. *Pharmaceutical Manufacturers Ass'n.* v. *Weinberger,* 401 F. Supp. 444, 449 (D.C. 1975). In any event, the expense of an administrative proceeding is not usually considered irreparable injury. *Meyers* v. *Bethlehem Shipbuilding Corp.,* 303 U.S. 41, 51 (1938), *Hornblower & Weeks-Hemphill Noyes, Inc.,* v. *Csaky,* 427 F. Supp. 814 (S.D.N.Y. 1977). Thus, New Mexico presents no valid argument that it will be irreparably injured if it is required to achieve the November 8, 1981 deadline set by UMTRCA.[12]

II. Probability of Success on the Merits

Operators contend that the NRC regulations are unlawful because they were issued in advance of EPA's promulgation of standards pursuant to Section 275.b. of the Atomic Energy Act of 1954, as amended, and invalid because they are based upon information not disclosed for public comment.

Section 275.b. of the Atomic Energy Act of 1954, as amended (Act) authorizes the Administrator of the Environmental Protection Agency (EPA) to issue health and environmental standards for the protection of the public health, safety, and the environment from radiological and non-radiological hazards associated with the processing, possession, transfer, and disposal of uranium mill tailings at active processing sites or disposal sites. Section 275.b.(1). Once the EPA standards are issued and become effective, the NRC is responsible for their implementation and enforcement in non-Agreement States through the conduct of licensing activities under the Act. Section 275.d. In Agreement States, the implementation and

[11]Although Operators' Motion provides some examples of alleged expense and disruption, the analysis in the Commission's Opposition explains why these examples do not support a stay.

[12]For the reasons contained in the Commission's Opposition, this conclusion is unchanged by New Mexico's *amicus* filing before the Tenth Circuit.

enforcement of EPA's standards is the responsibility of the Agreement States. Sections 275.d. and 274.o.

The argument that the Commission should delay promulgation of mill tailings regulations pending EPA's issuance of general standards is not new. It was made by commentors in response to the NRC's proposed mill tailings regulations. The NRC's reasons for rejecting that interpretation of UMTRCA are clearly spelled out in the Statement of Consideration (Statement) accompanying the final rule.[13]

In that Statement, the Commission stated that an analysis of the UMTRCA and its legislative history show that the NRC has the immediate duty to ensure that the management of uranium mill tailings is carried out in a manner that will protect the public health and safety and the environment. Moreover, the Commission observed that any delay would have made it difficult, if not impossible, for the Agreement States to issue equivalent standards by November 8, 1981 as required by UMTRCA. Section 275.o.(2). Finally, the Commission noted that NRC is aware that its regulations must be compatible with any generally applicable standards established by EPA; and that in recognition of this situation the NRC staff closely coordinates its activities with EPA on this matter.

Operators have provided no new information on this issue.[14] Thus, the Commission has no reason to question its prior belief that it acted lawfully in promulgating uranium milling regulations prior to EPA's issuance of general standards. The NRC's decision to issue regulations is clearly consistent with Congress' concerns regarding the health hazard posed by uranium milling and the need to upgrade State regulations in that area. Moreover, effectiveness of the coordination between EPA and NRC is clearly demonstrated by the consistency between EPA's recently proposed standards for inactive mill tailings piles[15] and NRC regulations for the long-term management of tailings piles. Finally, if any aspects of the NRC's regulations prove to be inconsistent with EPA's standards, the NRC can always modify its regulations. Any such modifications would probably not be significant because of the effective coordination between EPA and NRC. For these reasons, and the reasons in the Commission's Opposition, the Commission finds no merit in petitioners' contention.

In their motion before the Commission, Operators provided no support for their allegation that the regulations are based on information not disclosed for public comment. The regulations are supported by a three-

[13]See, 45 *Fed. Reg.* at 65523.
[14]Operators Motion also provided no new information on this issue.
[15]46 *Fed. Reg.* 2556 (January 9, 1981).

volume Final Generic Environmental Impact Statement on Uranium Milling (GEIS).[16] That GEIS is based on the reports of several technical studies made publicly available by the NRC, and was publicly reviewed in draft form. Under these conditions, Operators' contention is utterly groundless.[17]

New Mexico has not provided any reasons to believe that the Operators will succeed on the merits of their claims. Thus, there is nothing in New Mexico's filing to support this element in the analysis of its stay request.

III. Harm to Other Interested Parties

The public could be harmed by a stay of the NRC regulations on uranium milling. Radioactive releases from existing uranium mills constitute the largest potential source of routine releases from the nuclear fuel cycle. Members of the public who live in the vicinity of uranium mills are exposed to radiation from those mills; and most of the population of the United States is exposed to radioactive radon from uranium mill tailings piles. Thus, many persons could be unnecessarily exposed to excess levels of radiation if the Agreement States delay implementation of the regulations.

Operators contend that the NRC has admitted that the risks posed by uranium milling are *de minimis*. This is simply wrong. Operators rely on the proposed narrative explanation for Table S-3, Uranium Fuel Cycle Environmental Data[18] and a recent study entitled *Radon Releases From Uranium Mining and Milling And Their Calculated Health Effects* (Radon Report).[19] On April 14, 1978 the Commission deleted from Table S-3 the value for releases of radon from the uranium fuel cycle and noted that radon releases could be considered in individual licensing proceedings pending a generic determination of the radon release value.[20] Thus, neither Table S-3 nor its proper narrative explanation consider the environmental impacts of radon. Perforce, the proposed narrative explanation of Table S-3 does not support the claim that the risks posed by uranium milling are *de minimis*. The Radon Report also does not support that conclusion. Radon is the primary source of long-term public exposure to radiation resulting from uranium milling. If adequate measures are not taken to control radon emissions from mill tailings piles, the public exposure to that source would

[16]NUREG-0706.
[17]Subsequently, Operators' Motion identified certain documents which were not available until the comment period closed. These documents do not support a stay for the reasons given in the Commission's Opposition.
[18]46 *Fed. Reg.* 15154 *et seq.* (March 4, 1981).
[19]NUREG-0757 (1981).
[20]43 *Fed. Reg.* 15613.

exceed its exposure to all the other radiation sources associated with uranium fuel cycle. The Radon Report states that the radiation exposure due to untreated uranium mill tailings piles will be one hundred times the exposure from piles stabilized in accordance with NRC regulation.[21] Thus, contrary to Operators' allegation, the Radon Report supports the NRC's conclusion that the public health, safety and the environment require regulation of mill tailings piles.

IV. Public Interest

The public interest in protection of the public health and safety also warrants denial of a stay. Congress enacted the UMTRCA because of its concern over the public's exposure to radioactive radon gas and other radioactive materials associated with uranium milling.[22] Moreover, Congress decided that the NRC should have clear authority to regulate uranium milling activities, that Agreement States regulations or uranium milling should at least be equivalent to that of the federal government, and that Agreement States regulations should be upgraded no later than November 1, 1981.[23] These Congressional actions clearly demonstrate that it is in the public interest to implement the Commission's Uranium Mill Licensing Requirements on the time table Congress established.

For all these reasons, and those in the Commission's Opposition, petitioners' motion for a stay is denied.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 26th day of May, 1981.

[21]NUREG-0757 at 2-12.
[22]H.R. Rep. No. 95-1480, Part 2 Uranium Mill Tailings Radiation Control Act of 1978, 95th Cong., 2d Sess. 25 (1978).
[23]*Id.* at 43.

Cite as 13 NRC 469 (1981) **ALAB-639**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:
Richard S. Salzman, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket Nos. 50-498 OL**
50-499 OL

HOUSTON LIGHTING
& POWER COMPANY *et al.*
(South Texas Project, Units 1 and 2) **May 8, 1981**

Acting on the staff's motion, the Appeal Board reviews on certification and reverses the Licensing Board's order directing the staff to give intervenors the names of individuals who, after receiving a pledge that their identities would be held confidential, reported questionable construction practices at the South Texas facility to the Commission.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

As a general matter discovery orders are interlocutory and accordingly not reviewable as of right before the end of the case.

RULES OF PRACTICE: DISCOVERY (PRIVILEGED MATTER)

The Government enjoys a privilege to withhold from disclosure the identity of persons who furnish information on violations of law to officers charged with enforcement of that law. *Rovario* v. *United States,* 353 U.S. 53, 59 (1957). This "informer's privilege" obtains not only in criminal cases but exists in civil cases and is embodied as well in the Freedom of Information Act, 5 U.S.C. § 552(b)(7)(D).

RULES OF PRACTICE: DISCOVERY (PRIVILEGED MATTER)

The "informer's privilege" applies in Commission proceedings. 10 C.F.R. §§ 2.744(d), 2.790(a)(7) and 21.2; *Northern States Power Co.* (Monticello Plant, Unit 1), ALAB-16, 4 AEC 435, *affirmed by the Commission*, 4 AEC 440 (1970).

RULES OF PRACTICE: DISCOVERY (PRIVILEGED MATTER)

The privilege to withhold the names of confidential informants is not absolute; it must yield where the informer's identity is relevant and helpful to the defense of an accused, or is essential to a fair determination of a cause. *Rovario, supra*, 353 U.S. at 60-61.

RULES OF PRACTICE: DISCOVERY (PRIVILEGED MATTER)

The burden to obtain the names of confidential informants is not met by speculation that their identification might be of some assistance to those who seek disclosure.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Denials of discovery requests are interlocutory orders; the right to appellate review of such rulings must abide the end of the case and an appeal of the licensing board's initial decision. 10 C.F.R. § 2.730(f); *Toledo Edison Co.* (Davis-Besse Station), ALAB-300, 2 NRC 752, 759 (1975).

APPEARANCES

Messrs. Edwin J. Reis and **Jay M. Gutierrez** for the Nuclear Regulatory Commission staff, *petitioners*.

Mr. William S. Jordan, III, Washington, D.C., and **Ms. Pat Coy** and **Mr. Robert Hager**, San Antonio, Texas, for intervenors Citizens Concerned About Nuclear Power, Inc., and Citizens for Equitable Utilities, *respondents*.

Messrs. Jack R. Newman, Maurice Axelrad, and **David B. Raskin**, Washington, D.C., and **Finis E. Cowan** and **Thomas B. Hudson, Jr.**, Houston, Texas, for Houston Lighting & Power Company *et al.*, *applicants*.

DECISION

Opinion of the Board by Mr. Salzman and Dr. Buck:

I.

The Licensing Board has ordered the staff to give intervenors the names of individuals who, after receiving a pledge that their identities would be held confidential, reported questionable construction practices at the South Texas facility to the Commission. We grant the staff's motion for review of that order and set it aside as an unjustified departure from Commission policy that "the identify of anyone so reporting will be withheld from disclosure." 10 C.F.R. § 21.2.

1. The background of this case has previously been recounted elsewhere[1] For purposes of the matters before us the following points are salient: Houston Lighting and Power Company has been licensed to construct the "South Texas Project," a nuclear power plant near Bay City, Texas. The Power Company in turn engaged Brown and Root, architects-engineers, to design and build the plant. A lengthy NRC staff investigation into construction practices at that facility led the Director of Inspection and Enforcement on April 30, 1980, to stop work on the project because of widespread noncompliance with Commission requirements. In addition, the Director ordered the Power Company to "show cause" why civil penalties of $100,000 should not be imposed against it and construction work remain suspended until corrective measures were taken. At the core of the Director's charges was the allegation that the Power Company had failed to supervise its contractor with the result (among other things) that quality control of safety-related construction was inadequate and the quality control inspectors harassed and otherwise discouraged from doing their jobs properly.

The Power Company did not contest the charges. Instead, it paid the penalty and instituted at the Director's behest a broad series of management changes designed to correct the situation. Two private groups, Citizens Concerned About Nuclear Power, Inc., and Citizens for Equitable Utilities (hereinafter collectively, "Citizens"), questioned the efficacy of the changes called for by the Director, alleged that the Power Company's failure to supervise the construction of its plant cast doubt on its fitness for a license to operate it, and demanded a public hearing on the show cause order. In lieu of that separate hearing, the Commission directed the

[1]See, *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), CLI-80-32, 12 NRC 281 (1980).

Licensing Board presiding over the ongoing operating license proceeding to consider those allegations promptly.[2] Citizens had previously intervened in that proceeding as parties opposed to licensing the plant.

2. This brings us to the specific matter at issue. In the proceeding below, Citizens demanded the names of the individual Brown and Root or Power Company employees who informed the staff about harassment at the South Texas Project as well as the names of those who had harassed and intimidated them. The staff declined to reveal its confidential informants and Citizens moved on March 16, 1981 to compel the staff to do so. On March 21, over the staff's objection, the Board ordered the informants disclosed to intervenors, subject to a protective order that the names not be revealed to officials of the Power Company, Brown and Root, or any of their subcontractors. (The names would be supplied, however, to the Power Company's counsel.)[3]

The Licensing Board did not make its order immediately effective to allow the staff time to seek appellate review.[4] On April 3rd, the staff filed a "Notice of Appeal" to this Board coupled with a "Motion for Direct Certification" under 10 C.F.R. § 2.785(d) asking us to send the Licensing Board's order to the Commission immediately for its direct review without passing on it ourselves.[5]

II.

As a general matter discovery orders are interlocutory and accordingly not reviewable as of right before the end of the case.[6] Whether a disclosure order of the kind in question is sufficiently distinct from the main proceeding so as to be appealable now under the "collateral order doctrine"[7] is an issue about which the federal courts are themselves divided.[8] We need not address that question here, however; the disclosure order is of sufficient general importance in the scheme of Commission

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[2]CLI-80-32, *supra*, 12 NRC at 290-92.

[3]Memorandum and Order of March 24, 1981. A formal protective order has not yet been drawn up by the Board; it contemplates issuing an order similar to one it previously approved.

[4]*Id.* at 8.

[5]The applicants generally support the staff's position in this case.

[6]See 10 C.F.R. § 2.730(f); *Pennsylvania Power & Light Co.* (Susquehanna Station, Units 1 and 2), ALAB-613, 12 NRC 317, 321 (1980).

[7]See *Cohen* v. *Beneficial Loan Corporation*, 337 U.S. 541, 546 (1949).

[8]*Compare, e.g., In re United States*, 565 F.2d 19, 21 (2nd Cir. 1977), *certiorari denied sub nom. Bell* v. *Socialist Workers Party*, 436 U.S. 962 (1978), with *Southern Methodist Univ. Ass'n.* v. *Wynne & Jaffe*, 599 F.2d 707, 711-12 (5th Cir. 1979).

operations to merit review on certification under our decisions, particularly because it must be examined now or not at all.[9] We find no occasion, however, to pass the matter along to the Commission without ruling on it. To do so would shirk our own responsibilities and sanction a practice we frown upon ourselves.[10]

III.

1. The Supreme Court has recognized "the Government's privilege to withhold from disclosure the identity of persons who furnish information of violations of law to officers charged with enforcement of that law. The purpose of the privilege is the furtherance and protection of the public interest in effective law enforcement. The privilege recognizes the obligation of citizens to communicate their knowledge of the commission of crimes to law-enforcement officials and, by preserving their anonymity, encourages them to perform that obligation." *Roviaro* v. *United States,* 353 U.S. 53, 59 (1957) (citations omitted). This "informer's privilege" obtains not only in criminal cases but exists (perhaps even more strongly) in civil cases,[11] and is embodied as well in the Freedom of Information Act.[12]

That the privilege applies in this agency's proceedings is not disputed; long-standing decisions[13] and express Commission regulations eliminate any debate. 10 C.F.R. §§ 2.744(d), 2.790(a)(7). Indeed, Part 21 of the regulations ("Reporting of Defects and Non-Compliance") explicitly invites individuals who become aware of nuclear safety-related problems to report them to the appropriate Commission office, tendering assurance that, "as authorized by law, the identity of anyone so reporting will be withheld from disclosure." 10 C.F.R. § 21.2.[14]

The privilege to withhold the names of confidential informants is not absolute; it must yield where the informer's identity "is relevant and helpful to the defense of an accused, or is essential to a fair determination of a

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[9]See *Kansas Gas & Electric Co.* (Wolf Creek Station, Unit 1), ALAB-327, 3 NRC 408, 413 (1976); *The Toledo Edison Co.* (Davis-Besse Station), ALAB-300, 2 NRC 752, 759 (1975).

[10]The staff also asks us to review the Licensing Board's ruling permitting Citizens to file their motion to compel six weeks late. For reasons we explained in declining to take up similar procedural objections made by Citizens in this case, the staff's complaints do not merit the exercise of our certification jurisdiction. *Houston Lighting & Power Co.* (South Texas Project), ALAB-637, 13 NRC 367 (April 16, 1981).

[11]See, *In re United States, supra,* 565 F.2d at 22, and cases there cited.

[12]5 U.S.C. § 552(b)(7)(D); *Church of Scientology* v. *Department of Justice,* 612 F.2d 417 (9th Cir. 1979); *Nix* v. *United States,* 572 F.2d 998 (4th Cir. 1978).

[13]*Northern States Power Co.* (Monticello Plant, Unit 1), ALAB-16, 4 AEC 435, *affirmed by the Commission,* 4 AEC 440 (1970).

[14]The regulation includes telephone numbers in each region that may be called collect for this purpose.

cause." *Roviaro, supra,* 353 U.S. at 60-61. In the matter before us the Power Company, not Citizens, is the "accused;" we must therefore focus on the second prong of the test.[15] The issue is thus whether the Licensing Board abused its discretion in ordering the staff's confidential informants revealed to these intervenors as "necessary to a proper decision in this proceeding."[16]

A fuller appreciation of the basis for the privilege being asserted is necessary to answer that question. For present purposes, we may simply restate the Second Circuit's authoritative and succinct rationale (*In re United States, supra,* 565 F.2d at 22 (citations omitted)):

> The question of informer privilege is, of course, not one of first impression. It is an ancient doctrine with its roots in the English common law, founded upon the proposition that an informer may well suffer adverse effects from the disclosure of his identity. Illustrations of how physical harm may befall one who informs can be found in the reported cases. However, the likelihood of physical reprisal is not a prerequisite to the invocation of the privilege. Often, retaliation may be expected to take more subtle forms such as economic duress, blacklisting or social ostracism. The possibility that reprisals of some sort may occur constitutes nonetheless a strong deterrent to the wholehearted cooperation of the citizenry which is a requisite of effective law enforcement.
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> Courts have long recognized, therefore, that, to insure cooperation, the fear of reprisal must be removed and that "the most effective protection from retaliation is the anonymity of the informer." "By withholding the identity of the informer, the government profits in that the continued value of informants placed in strategic positions is protected, and other persons are encouraged to cooperate in the administration of justice." Congress, also, has recognized the importance of this protective measure.

The need to protect confidential informants is not an academic concern to the NRC. That this is so finds confirmation not only in the Commission's regulations (to which we have already alluded), but in Congressional enactments designed specifically to safeguard from retaliation those who assist the NRC in carrying out its safety responsibilities.[17] And, as we have ourselves recognized:

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[15]See 10 C.F.R. §§ 2.744(d) and 2.790(a)(7).

[16]Order of March 24, 1981 at 7.

[17]Energy Reorganization Act of 1974, as amended, § 210, 42 U.S.C. § 5851 (1980).

Common sense tells us that a retaliatory discharge of an employee for "whistleblowing" is likely to discourage others from coming forward with information about apparent safety discrepancies. Yet, the Commission's safety inspectors cannot be everywhere; to an extent they must depend on help of this kind to do their jobs.[18]

2. The staff was thus on firm ground in promising confidentiality to these informants. It cannot be gainsaid that the individuals who cooperated with the staff's South Texas investigation risked not only financial and social penalties in doing so, but physical abuse as well. We need not belabor the point; Citizens candidly conceded this to the board below.[19]

To overcome the acknowledged importance of the need for confidential treatment of informants, the burden was on the intervenors to demonstrate the need for their disclosure.[20] We have reviewed Citizens' submissions in this regard with care. Their arguments boil down to the assertion that, without those names, they "have no way of judging whether what has been provided to them in any way reflects an accurate rendition of what was said by the inspectors in question."[21] But a moment's reflection reveals that this is an inadequate reason. It is always the case that, without questioning the informants themselves, one cannot learn *directly* whether all the reliable information they conveyed has been fully disclosed. But the staff made available to intervenors not only its detailed statement of charges against the Power Company — these alone run some 20 single-spaced typed pages — but its even lengthier underlying inspection report as well. The show cause order, it is to be remembered, was drafted before the staff could know

[18]*Union Electric Company* (Callaway Plant, Units 1 and 2), ALAB-527, 9 NRC 126, 134 (1979).

[19]Citizens' Motion to Compel NRC Staff to Provide Information, filed March 16, 1981, at 3-4. See also, Order to Show Cause of April 30, 1980, Appendix A, *passim*.

[20]10 CFR § 2.744(d); *Northern States Power Co.* (Monticello Plant, Unit 1), ALAB-10, 4 AEC 390, 395, 399; *Id.*, ALAB-16, 4 AEC 435, 436, *affirmed*, 4 AEC 440; *In re United States*, *supra*, 565 F.2d at 23 and cases there cited. The staff's assertion that the information was obtained on a pledge that the informants' names would not be disclosed was not challenged by intervenors nor questioned by the Board below. There is, therefore, no need for us to investigate whether there may be "loopholes" in the arrangement. See *Monticello*, *supra*, ALAB-10, 4 AEC at 394-95.

[21]Citizens' Opposition to NRC Staff Appeal, etc., at 7. The quote in the text is from counsel's brief on intervenors' behalf. Previously, the same point was made by intervenors' representatives in a document entitled "Opposition to NRC's Notice of Appeal" etc. (page 4), dated April 13, 1981, and in Citizens' Motion to Compel cited in fn. 19, *supra*.

The dissent fears that in reducing intervenors' arguments to essentials we have lost their essence. We think not. Even the fuller exposition of Citizens' position quoted in the dissent does little more than reassert that "the intervenors must have an opportunity to examine [the accuracy of the staff's information]." (*Infra*, p. 20.) Still missing from their argument, however, is the key ingredient — their basis (other than surmise) for believing that the staff has not made a full disclosure of its informants' revelations.

whether the Power Company would challenge the Director's allegations; there is no basis to believe that it "pulls any punches." Indeed, given the staff's demand for the maximum civil penalty, there is no reason to assume that the charges omitted any significant incident the staff thought sustainable.

That information has been in Citizens' hands for some considerable time. (The Order to Show Cause has been available for more than a year now.) They assertedly have their own confidential sources of information. Yet intervenors have never even attempted a showing that the staff did not spread its full case on the record. It is settled law that the burden to obtain the names of confidential informants is not met by speculation that their identification "might" be of some assistance to those who seek disclosure.[22] To accept intervenors' argument would render NRC pledges of anonymity to informants meaningless — if not disingenuous. Intervenors in every such case could then be expected to demand their names to see if the staff had disclosed the informants' confidences "accurately." There is no justification for that impedance to obtaining the cooperation of informants.

3. Citizens pressed the additional argument below that, in directing the Licensing Board to take up their allegations, the Commission had ordered the intervenors be given the names of the staff's informants.[23] The Licensing Board rejected that contention, holding that the Commission had simply allowed intervenors to seek to learn those names by invoking the usual mechanisms provided in the Rules of Practice.[24] On this point we agree with that Board. We are unpersuaded that the Commission would have disregarded established evidentiary privileges and departed from its own published policies without saying so in unmistakable terms.

4. The Licensing Board's explanation for its ruling provides no greater enlightenment about why disclosure of the staff's confidential informants is "necessary to a proper decision in this proceeding." The Board simply states without findings or analysis its belief that

> it is important to know whether the individuals who allegedly reported harassment to the intervenors are the same as those who allegedly reported it to the Staff. If they are different, the scope of the harassment questions which we must adjudicate may be far broader and widespread than if the individuals reporting to the intervenors and to the Staff are identical.[25]

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[22] *United States* v. *Prueitt*, 540 F.2d 995, 1003-04 (9th Cir. 1976), *certiorari denied sub nom. Temple* v. *United States*, 429 U.S. 1063 (1977); *United States* v. *D'Amato*, 493 F.2d 359, 366 (2nd Cir.), *certiorari denied*, 419 U.S. 826 (1974).

[23] See CLI-80-2, *supra*, 12 NRC 281.

[24] Memorandum and Order of March 24, 1981 at 5.

[25] Memorandum and Order of March 24, 1981 at 8. The relevant portion of that memorandum

It by no means follows from those conclusions that the names of the staff's informants must perforce be disclosed to intervenors. Not the individuals but their information is of significance to the proceeding. Had Citizens demonstrated that their own informants tell a significantly different story than the one reported by the staff, we might have a different situation. But that is not the case here. We stress again that Citizens make no such claim. Rather, their papers indicate only a wish to interview the informants on the possibility that the staff may not have obtained all the information they possessed.

For the reasons we have already explained, unsupported speculation of this sort is legally insufficient to override the informer's privilege. To accept it as a basis for revealing the names of individuals promised anonymity in exchange for information renders the privilege virtually useless as a means of encouraging employees to volunteer information about defects that might otherwise pass unnoticed. See pp. 475-76, *supra*. With all deference, the Licensing Board's refusal to recognize the privilege here has the practical effect of negating important Commission policy without cause.

The Board's action is no less abusive of its discretion because it directed the informants' identities disclosed subject to a protective order. Such an order does not cure the vice in releasing their names. The intervenors' stated purpose is to interview as many of the informants as possible. Their doing so would make it immediately obvious that the NRC's pledge of anonymity had been broken. The intervenors' well-meant actions would nevertheless be instrumental in undoing the reason for recognizing the informer's privilege in the first place. Clairvoyance is not needed to appreciate that word of the breach of confidentiality would spread and the likelihood of informants coming forward with safety-related information in future cases be diminished.

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reads in its entirety as follows (*id.* at 7-8):

> As announced at the prehearing conference, we find that the names of the particular QA/QC inspectors who supplied information to the Staff concerning harassment, and the names of the employees who allegedly harassed and intimidated those inspectors, are not only directly relevant to CCANP's contentions but are also necessary to a proper decision in this proceeding. Specifically, it is important to know whether the individuals who allegedly reported harassment to the intervenors are the same as those who allegedly reported it to the Staff. If they are different, the scope of the harassment questions which we must adjudicate may be far broader and widespread than if the individuals reporting to the intervenors and to the Staff are identical. Moreover, proper development of the record in this proceeding in an expeditious manner suggests that the names of individuals should be made available on discovery rather than waiting for the evidentiary hearing. Finally, by its very nature, the information can only be obtainable from the staff.

It is very easy when focusing on the immediate concerns of the case at bar to take the short view and err on the side of disclosing confidential sources of information. But this is neither the sole reactor under construction nor the only one in which informers may play an important role in bringing potentially dangerous situations to the Commission's attention. The informer's privilege, as it has been developed and refined by the courts over the years, is an attempt to balance the government's recognized need for information over the long range with the necessities of a fair hearing and a full record in a particular case. We deem it both Commission policy and the soundest course to insist that this balance be struck in accordance with established jurisprudence. Citizens failed to demonstrate more than a speculative need for revealing the staff's confidential sources.[26] Consequently, the order directing that disclosure must be set aside.[27]

[26]Disagreement on this point essentially fuels the dissenting arguments. Because of this, we see no gain in lengthening the opinion to address matters that, but for that disagreement, we probably would not dispute. On the main question, however, our colleague, like the intervenors, provides no basis for believing that the staff came forward with less than all the information necessary for "a full adjudicatory hearing." The need to protect confidential informants is not overcome by speculation about why the staff needed "two-and-one-half years, 12 separate investigations, and numerous conferences with the applicants," before initiating formal action. (*Infra*, p. 482.) If something could be drawn from this, an equally logical — and in the circumstances more likely — answer would be the difficulty of getting knowledgeable individuals to speak out in light of their very real fear of retaliation. More to the point is the fact that the nature of the staff's investigation was fully known to the Commission when it instructed the Licensing Board to turn to the intervenors' allegations. Nevertheless, as previously noted, the Commission did not order the staff's confidential informants disclosed — although the same arguments now pressed on us were then presented to it. The short answer, we think, is that it does not follow rationally from the length of the staff's investigation and the persistence of its investigators that the ultimate report is either inaccurate or incomplete. And without some more concrete showing, there is no occasion for the government to renege on promises to private individuals in this case that, in exchange for information otherwise unobtainable, their identities would be withheld from disclosure.

Nor is it a "curiosity," as the dissent suggests (p. 483), that the staff did not object to disclosure of the private intervenors' informants; the informer's privilege inures only to law enforcement officials. *Roviaro*, *supra*, 353 U.S. at 59. Intervention in one Commission proceeding does not entitle Citizens to privileged information that, if disclosed, might jeopardize the NRC's likelihood of receiving similar reports in future cases involving other plants. It is the NRC's continuing need for confidential informants that made the Licensing Board's failure to recognize the importance of the privilege both shortsighted and arbitrary.

[27]Nor would there be virtue in accepting Citizens' suggestion to the Board below that the names of those informants who have left Brown and Root's employment be disclosed. As the Court of Appeals explained in *Hudgson* v. *Charles Martin Inspectors of Petroleum, Inc.*, *supra*, 459 F.2d at 306: "The possibility of retaliation, however, is far from being 'remote and speculative' with respect to former employees for three reasons. First, it is a fact of business life that employers almost invariably require prospective employees to provide the names of their previous employers as references when applying for a job. Defendant's former employees could be severely handicapped in their efforts to obtain new jobs if the defendant should brand them as 'informers' when references are sought. Second, there is the possibility that a former employee may be subjected to retaliation by his new employer if that employer finds out that

IV.

Citizens responded to the staff's papers seeking review of the informer's privilege point with a "conditional cross-appeal" questioning other discovery rulings rendered by the Licensing Board. They ask that their appeal be considered "if permission to take [the staff's] interlocutory appeal is not promptly denied * * *."

We need not decide whether the Rules of Practice permit a cross-appeal in these circumstances, for *denials* of discovery requests are interlocutory orders not appealable at this stage of the proceedings in any event. 10 C.F.R. § 2.730(f); *Davis-Besse, supra* ALAB-300, 2 NRC at 758-59. The right to appellate review of such rulings must abide the end of the case and an appeal from the Licensing Board's initial decision. Unlike the disclosure order to which the staff objected, a matter we were obliged to take up now or not at all,[28] the matters Citizens complain about can be reviewed and redressed at the end of the case if erroneous. Our certification jurisdiction was not conferred to permit interlocutory review of discovery rulings of this nature. *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-634, 13 NRC 96 (February 19, 1981).

Treating the staff's papers as a motion for directed certification under 10 C.F.R. § 2.718(i), the motion is *granted* and the Licensing Board's order of March 24, 1981, insofar as it directs the staff to disclose the names of confidential informants to the intervenors, is *vacated*.[29] The intervenors' conditional cross-appeal is *dismissed* for want of jurisdiction; treating their papers as a motion for directed certification, the motion is *denied*.

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the employee has in the past cooperated with the Secretary. Third, a former employee may find it desirable or necessary to seek reemployment with the defendant. In such a case the former employee would stand the same risk of retaliation as the present employee.

There is no ground for affording any less protection to defendant's former employees than to its present employees. *Wirtz* v. *B.A.C. Steel Products, Inc.*, 312 F.2d 14 (4th Cir. 1962)."

[28]See, *Wolf Creek, supra*, ALAB-327, 3 NRC at 413.

[29]In light of our disposition of the matter, we see no cause to certify the staff's complaint to the Commission.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

[The opinion of Ms. Kohl, dissenting in part, follows.]

Ms. Kohl, dissenting in part:

While I do not disagree with the principles stated in the majority opinion concerning the informer's privilege, I believe their application to the facts of this case warrants affirmance of the Licensing Board's ruling. Accordingly, I dissent from the majority's decision insofar as it concludes that Citizens have not demonstrated that disclosure of the staff's confidential informants "is necessary to a proper decision in this proceeding." I also disagree that the protective order "does not cure the vice in releasing their names."

1. The majority finds Citizens' reasons for needing the requested information to be inadequate and speculative.[1] The majority "boils down" Citizens' arguments "to the assertion that, without those names, they 'have no way of judging whether what has been provided to them in any way reflects an accurate rendition of what was said by the inspectors in question.' " But distilling Citizens' arguments in this manner deprives them of their real substance. Citizens provided, *inter alia*, the following significant elaboration on their need for the names of the staff's informers (CEU Opposition to NRC Staff Appeal at 7-9):[2]

> As adverse parties in this proceeding, the intervenors must have an opportunity to examine [the accuracy of the staff's information]. Otherwise, the NRC Staff will have taken on the role of the Licensing Board itself, and the Board will be unable to make an independent judgment on the validity of the Staff's assertions. The Staff can hardly be allowed to assume the role of determining what facts shall be heard when it is also an adversary party to this proceeding.
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> * * *
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> Most important, the individuals whose identities are at issue here are precisely those people who have the best information concerning what has been happening at the South Texas Project for the past several years. Their information forms the basis for the NRC's stopwork order. By necessity, their information will form a major part of the basis for

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[1]Neither the staff nor the majority disputes the Licensing Board's other findings pursuant to 10 C.F.R. § 2.744(d) that the information is relevant and unobtainable elsewhere.

The fact that the names of the staff's informers are obtainable only from the staff distinguishes this case from a number of others upon which the staff and majority rely, where the information sought to be discovered was either within the knowledge of the requesting party or available from other sources. See, *e.g.*, *Suarez* v. *United States*, 582 F.2d 1007, 1012 (5th Cir. 1978); *Usery* v. *Local 720, Laborers' Int'l*, 547 F.2d 525, 528 (10th Cir.), *cert. denied*, 431 U.S. 938 (1977); *Brennan* v. *Engineered Products, Inc.*, 506 F.2d 299, 303 (8th Cir. 1974); *Hodgson* v. *Charles Martin Inspectors of Petroleum, Inc.*, 459 F.2d 303, 307 (5th Cir. 1972).

[2]See also CCANP Motion to Compel at 3; CCANP Opposition to NRC's "Notice of Appeal" at 4; CCANP Brief on Appeal at 7.

the Licensing Board's ultimate decision. These issues are too important to be allowed to proceed on the NRC Staff's hearsay statements of what they have learned from the actual QA/QC inspectors who were subjected to harassment. That is particularly the case if the NRC Staff is to take a position in favor of continued participation in the South Texas Project by Houston Lighting and Power and by Brown and Root. Given that position by the NRC Staff, reversal of the Order to Compel would eliminate the ability of the only parties adverse to Houston Lighting and Power and Brown and Root to determine the truth of factual assertions made by those favorable to the Houston Lighting and Power and Brown and Root positions, and it would seriously damage, if not destroy, the intervenors' ability to participate effectively in this proceeding.

I believe that this explanation amply satisfies Citizens' burden of showing a need for disclosure, particularly in light of other salient points in this case not addressed by the majority.[3]

For instance my colleagues emphasize that the staff has provided Citizens with a detailed statement of the charges against the applicants in the related show cause proceeding, as well as a lengthy inspection report. In their view, this, plus the staff's demand for the maximum civil penalty, afford "no basis to believe that [the show cause order] 'pulls any punches.'" But the majority gives undue weight to these matters. First, statements of charges in connection with show cause orders routinely must be made public under 10 C.F.R. § 2.790(a). Second, the staff did not turn over its inspection report until December 1980 and then only in response to intervenors' request. More significant, in my view, is the fact that it was only after two-and-one-half years, 12 *separate* investigations, and numerous conferences with the applicants, that the staff finally initiated formal action against applicants. CLI-80-32, 12 NRC 281, 283, 291 (1980).[4] This fact, in conjunction with the Commission's pledge to Citizens of "a full adjudicatory hearing" (*id.* at 291) with a " 'full airing of all relevant information regarding the safety of the ... plant' " (*id.* at 290), give Citizens more than adequate justification for its assertion of need to pursue independently the staff's sources.

[3]The Supreme Court in *Roviaro* v. *United States*, 353 U.S. 53, 62 (1957), directs decisionmakers to take account of "the particular circumstances of each case" and all "relevant factors" when balancing the public interest in protecting confidential sources against an individual's right to a fair chance to present his case.

[4]See *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-283, 2 NRC 11, 13-14 (1975), another case involving quality assurance questions, in which the staff's original investigation unfortunately proved to be incomplete.

The Licensing Board correctly recognized that if the staff's informers are not the same as intervenors', "the scope of the harassment questions which we must adjudicate may be far broader and widespread than if the individuals reporting to the intervenors and to the Staff are identical." Memorandum and Order at 7. It is a licensing board's responsibility, particularly in cases like this which involve serious safety questions, to make certain that the record is fully developed and that all parties have a fair opportunity to participate in that record development. The finding here that disclosure of the staff's informers to intervenors is "necessary to a proper decision" is in fulfillment of that responsibility.

2. Once Citizens established their need for the informers' names, it was incumbent on the staff to bolster its claim of privilege with details about its "blanket pledge of confidentiality." The mere invocation of the informer's privilege and the corresponding assertions of impediments to other investigations and reprisals if the pledge is broken are inadequate, under the balancing test of *Roviaro*,[5] to support a finding that the pledge outweighs a litigant's need for information. (Indeed, the policy concerns that underlie the privilege exist in every case; if they alone were adequate, the privilege would be absolute, rather than qualified.) But, curiously, the staff tells us virtually nothing about the pledge.

For example, it would be useful to know: (1) what the exact nature of the pledge is; (2) who made it and under what circumstances; (3) how often such pledges are made; (4) how often it is necessary to rely on informers promised confidentiality — as opposed to anonymous tips; (5) whether the promise of confidentiality is a well-established policy;[6] and (6) whether there were any efforts to get the informers to waive the privilege.

An additional curiosity is the staff's absence of objection to the Licensing Board's order compelling Citizens to disclose their confidential sources to applicants' counsel.[7] While Citizens, of course, do not enjoy even a qualified privilege, presumably some of the same concerns about reprisals pertain to disclosure of these informants as well. Thus, if the staff's real interest is encouraging plant employees and inspectors to reveal information

[5]See note 3, *supra*.

[6]The majority points to Part 21 of the Commission's regulations, "Reporting of Defects and Noncompliance," as evidence of the agency's longstanding reliance on the informer's privilege. See 10 C.F.R. § 21.2. The staff, however, does not rely on this provision, perhaps because the pledge of confidentiality promised in Part 21 contains its own explicit qualification: the identity of anyone reporting defects will be withheld "as authorized by law." Thus, the pledge embodied in Part 21 is limited by, and cannot take precedence over, other relevant Commission regulations (such as those governing disclosure of information, 10 C.F.R. § 2.790, 2.744), and the fundamental legal principles concerning this qualified privilege (*Roviaro* and its progeny).

[7]Memorandum and Order of March 18, 1981.

about possible safety violations without fear of intimidation — whether to the staff or concerned members of the public — the staff should object as strongly to disclosure of intervenors' informers as to disclosure of its own sources.

3. I believe that the Licensing Board properly weighed all the relevant factors in this case, in accordance with the mandate of *Roviaro* and the Commission's regulations set forth at 10 C.F.R. § 2.744. The Board is, after all, charged with the important function of rendering the initial decision in this case. It follows that its determination that certain information is "necessary to a proper decision" is entitled to substantial weight.[8] The Board's reasoning here that disclosure is necessary for a proper decision is persuasive and falls far short of the abuse of discretion the majority finds. Indeed, in *Virginia Electric and Power Co.* (North Anna Station, Units 1 and 2), CLI-74-16, 7 AEC 313 (1974), the Commission reached the same conclusion in a similar case involving a request for disclosure of privileged matter. The Commission found that "maximum reliance on the proper discretion of a Licensing Board is essential." *Id.* at 314. It also noted, as an independent basis for affirming the Board, that the policy considerations underlying the assertions of privilege should yield when safety issues are involved. *Id.* at 315. Applying these views to the instant case should result in affirmance, rather than reversal, of the Licensing Board's ruling.[9]

4. My views should not be interpreted as unsympathetic to the staff's well-founded concerns about the potentially adverse effect of disclosure of informers' identities on future investigations.[10] Indeed, my conclusion might well have been just the opposite had not the Licensing Board cautiously and appropriately made disclosure subject to a protective order.[11] But I do not share my colleagues' view that a protective order

[8] It is worth noting, in this regard, that an earlier version of 10 C.F.R. § 2.744 required licensing boards to give "great weight" to any staff objection to the production of privileged documents. See *Consumers Power Co.* (Midland Plant, Units 1 & 2), ALAB-33, 4 AEC 701, 704, 706, 708 (1971). The Commission deleted that provision (10 C.F.R. § 2.744(e)) in 1975 "to permit the presiding officer *in his discretion* to compel the attendance and testimony of AEC personnel and require the staff to answer written interrogatories." 40 Fed. Reg. 2973 (Jan. 17, 1975) (emphasis added).

[9] The majority finds the Board's decision to be "without findings or analysis." While its findings are stated briefly, I believe that the Board's "path may reasonably be discerned." *Bowman Transportation, Inc.* v. *Arkansas-Best Freight System, Inc.*, 419 U.S. 281, 286 (1974). When compared with the findings upheld by the Commission in *North Anna, supra*, the Board's decision is a veritable treatise. See LBP-74-16, 7 AEC 302, 303 (1974).

[10] At least one court, however, rejects the notion that individuals will be less likely to provide information concerning violations of the law simply because "in a few instances that information may be used in a later lawsuit." *Crawford* v. *Dominic*, 469 F. Supp. 260, 264 (E.D. Pa. 1979).

[11] See 10 C.F.R. § 2.740(c), 2.744(h).

"does not cure the vice in releasing [the informants'] names."[12] The fact that the Commission's regulations permit disclosure of sensitive information subject to protective orders reflects a fundamental faith in the adequacy of these instruments. Yet implicit in the majority's decision is the unwarranted concern that Citizens, and perhaps applicants' counsel as well, would not obey such an order. This is clearly at odds with the view we expressed in *Houston Lighting and Power Co.* (Allens Creek Station, Unit 1), ALAB-535, 9 NRC 377, 400 (1979), that "this Commission and its adjudicatory boards have always proceeded on the assumption that the terms of all protective orders will be scrupulously observed by everyone who acquires confidential information under such an order." It also overlooks our belief that "the law knows no *presumption* that anyone will disregard a protective order." *Pacific Gas and Electric Co.* (Diablo Canyon Plant, Units 1 and 2), ALAB-504, 8 NRC 406, 411 n.8 (1978) (emphasis in original).

On the contrary, in this instance there is every reason to presume that the parties would obey a protective order. The Board below indicated that the order would be "comparable to that under which [Citizens] have been directed to identify the names of certain of their informants." Memorandum and Order at 2. See also Memorandum and Order of March 18, 1981. This is significant for at least two reasons. First, that protective order was designed to guard the identities of Citizens' informants for reasons similar to those offered by the staff — *i.e.,* fear of harassment and reprisal.[13] Thus, because Citizens and the staff *both* now have a stake in the nondisclosure of the identities of their informers, there would be an element of mutual deterrence to violation of either protective order.[14] Second, the order that protects Citizens' sources requires disclosure only to *counsel* for applicants and the staff, presumably because the Commission can more effectively sanction attorneys who violate such orders.[15] Citizens' pleadings again show representation by legal counsel. Thus, the Board's protective order

[12]Both the majority and the staff find support for their view that disclosure here is not appropriate in *Northern States Power Co.* (Monticello Plant, Unit 1), ALAB-10, 4 AEC 390; ALAB-16, 4 AEC 435, *aff'd,* 4 AEC 440 (1970), despite the fact that the regulations at that time contained no provision for discovery of staff documents comparable to 10 C.F.R. § 2.744. See 10 C.F.R. § 2.740-2.743 (1970). But even assuming *arguendo* the relevance of *Monticello* to the instant case, it is noteworthy that in ALAB-16, 4 AEC at 436-437, we found disclosure of informants' names *in camera* to be acceptable. This being so, it is difficult to understand why a protective order — comparable to an *in camera* review — cannot likewise be acceptable here.

[13]See LBP-80-11, 11 NRC 477, 479-480 (1980). Of course, Citizens cannot claim even a qualified privilege against disclosure.

[14]And as the majority itself notes, Citizens are fully aware of and sensitive to the possibility of reprisals against informers whose identities become known publicly.

[15]See 10 C.F.R. § 2.713(c)(3), 45 *Fed. Reg.* 69877, 69879 (Oct. 22, 1980).

could similarly limit disclosure of the staff's informers only to *counsel* for Citizens and applicants.[16]

I believe that, in the circumstances of this case, the Licensing Board reached a reasonable accommodation of the competing interests involved, and I would affirm its decision.

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[16]A protective order and other provisions such as *in camera* review admittedly may not afford a "failsafe" shield from public exposure. But as the District of Columbia Circuit stated in *Westinghouse Electric Corp.* v. *City of Burlington, Vermont,* 351 F.2d 762, 771 (D.C. Cir. 1965), "no informer can ever be certain of anonymity."

Cite as 13 NRC 487 (1981) **ALAB-640**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Administrative Judges:*

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Richard S. Salzman
Dr. W. Reed Johnson
Thomas S. Moore

In the Matters of

PHILADELPHIA ELECTRIC COMPANY, *et al.* **(Peach Bottom Atomic Power Station, Units 2 and 3)**	**Docket Nos. 50-277 50-278**
METROPOLITAN EDISON COMPANY, *et al.* **(Three Mile Island Nuclear Station, Unit No. 2)**	**Docket No. 50-320**
PUBLIC SERVICE ELECTRIC AND GAS COMPANY (Hope Creek Generating Station, Units 1 and 2)	**Docket Nos. 50-354 50-355**
	May 13, 1981

Following a consolidated evidentiary hearing concerning the environmental release of radioactive radon gas attributable to the mining and

*The Appeal Panel members listed are on one or more of the Boards assigned to hear the captioned proceedings; their collective designation is simply a convenience in issuing this decision.

milling of uranium for nuclear reactor fuel, the Appeal Boards adopt radon release values to be factored into the cost-benefit analyses for the Peach Bottom, Hope Creek and Three Mile Island reactors.

RULES OF PRACTICE: COLLATERAL ESTOPPEL

Participants in a proceeding cannot be held bound by the record adduced in another proceeding to which they were not parties.

APPEARANCES

Messrs. Jay E. Silberg, Matias F. Travieso-Diaz, Troy B. Conner, Jr., and **Robert M. Rader,** Washington, D.C., jointly appearing on behalf of the applicants, Metropolitan Edison Co. *et al.,* Philadelphia Electric Co. *et. al.,* and Public Service Electric and Gas Co.

Ms. Sue Reinert, Oswego, New York, as a representative of intervenors, Ecology Action of Oswego.

Dr. Chauncey Kepford, State College, Pennsylvania, as a representative of the *Peach Bottom-Three Mile Island* intervenors, Citizens for a Safe Environment and the Environmental Coalition on Nuclear Power.

Mr. Bernard M. Bordenick for the Nuclear Regulatory Commission staff.

CONTENTS

DECISION

I. PROCEDURAL HISTORY

The National Environmental Policy Act (NEPA) requires that the Commission consider the environmental effects associated with the uranium fuel cycle and factor them into the cost-benefit analyses underlying its reactor licensing decisions. One effect of that fuel cycle is the release of radioactive radon gas (radon-222) to the atmosphere as a result of the mining and milling of uranium for reactor fuel. This decision concerns the environmental impact of those radon emissions to be factored into the cost-benefit analyses of the Peach Bottom, Three Mile Island and Hope Creek reactors.

Until recently, the Commission's regulations specified in tabular form (Table S-3) a value representing the environmental effects of radon gas to be employed in all agency cost-benefit analyses.[1] As originally promulgated in 1974, Table S-3 contained a value for emissions of radon-222 principally from uranium mills.[2] In November 1975, the New England Coalition on Nuclear Pollution filed a petition for rulemaking to amend that table. It asserted, among other things, that the table seriously understated radon emissions by disregarding long-term releases of radon gas from mill tailings piles and that, consequently, any use of the table would misrepresent the resultant effects on human health and safety from radon gas.[3]

The Commission published notice of the filing of the petition and received numerous public comments. It effectively granted the petition with respect to several elements other than radon by promulgating a revised interim Table S-3 in March of 1977.[4] Subsequently, on April 11, 1978 the Commission dealt with the radon aspects of the petition. It determined that the radon value in Table S-3 was incorrect and deleted that value from the table. Rather than immediately initiate a rulemaking proceeding on radon

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[1]See 10 CFR Part 51, Table S-3, "Table of Uranium Fuel Cycle Environmental Data." See also 10 CFR 51.23(c) (governing the use of Table S-3 in the staff's environmental impact statements for light water reactors).

[2]See 10 CFR Part 51, Table S-3 (1974 rev.).

[3]Petitioner advanced similar arguments regarding Table S-3 release values and health effects of krypton-85, tritium, and carbon-14. Supporting the petition was the affidavit of Dr. Robert O. Pohl, a professor of physics at Cornell University and an expert witness for the intervenors in this proceeding. With respect to the radon release value, Dr. Pohl asserted that the total curies of radon-222 released should be 2×10^7 (twenty million) rather than 75 as set forth in the table.

[4]That same year, Dr. Walter H. Jordan, a member of the Atomic Safety and Licensing Board Panel, also raised questions regarding radon gas releases in a memorandum to James R. Yore, then Chairman of the Panel. Among other things, that memorandum gave a detailed explanation of the author's technical grounds for considering the radon release value in Table S-3 to be in error.

releases, however, the Commission deferred any decision on a new rulemaking until the completion of the generic environmental impact statement on uranium milling. 43 Fed. Reg. 15613 (April 14, 1978). It announced instead that it would permit the radon question to be litigated in individual licensing proceedings. The Commission instructed the licensing and appeal boards to reopen the records in pending cases "to receive new evidence on radon releases and on health effects resulting from radon releases." 43 Fed. Reg. at 15615-16.

Soon after the Commission issued that directive, the NRC staff moved to consolidate some 17 proceedings pending before us for the purpose of receiving new evidence and determining the environmental effects of radon releases. Because of the generic character of the radon question, the staff urged that such consolidation would be the most efficient means of resolving those questions while curtailing the likelihood of inconsistent decisions. All of the applicants and intervenors who responded opposed the motion, generally doubting the staff's claims of efficiency and arguing that consolidation would be financially burdensome.

We attempted to fashion a feasible and fair alternative to consolidation. Noting that the Licensing Board in the *Perkins* construction permit proceeding[5] had recently held an evidentiary hearing on the radon question, we decided to adopt a "lead case" approach which incorporated the *Perkins* record and decision into the records of the other pending proceedings. Because the parties to those cases (with the exception of the NRC staff) had not participated in *Perkins*, we allowed them an opportunity "to supplement, contradict, or object to" the *Perkins* record and decision. ALAB-480, 7 NRC 796, 804-806 (1978).

The Licensing Board decided *Perkins* on July 14, 1978. LBP-78-25, 8 NRC 87. A number of intervenors were dissatisfied with that decision and the underlying record. They suggested, *inter alia*, that supplemental evidence was needed on the question of radon emission rates. We agreed, perceiving a need for greater specificity concerning the parties' objections. Accordingly, on December 1, 1978 we modified the procedure set out in ALAB-480 and asked intervenors to particularize their assertions concerning radon releases and concentration levels. We also requested all parties to brief the adequacy of the Licensing Board's "*de minimis* theory of health effects." ALAB-509, 8 NRC 679, 684 (1978). The *Perkins* Board had found that, compared to natural radon releases, the radon emissions associated with the mining and milling of uranium fuel for that facility were so small as to be completely undetectable and their health effects insignificant. 8

[5] *Duke Power Co.* (Perkins Nuclear Station, Units 1, 2 and 3), Docket Nos. STN 50-488, 50-489, and 50-490.

NRC at 100. We explained that if we were to find the *Perkins* radon emission figures substantially correct, we would then confront the acceptability of the Board's approach to the health effects question.

In response to ALAB-509, intervenors filed various general objections to the *Perkins* record and the Board's rationale.[6] The NRC staff and applicants in eleven proceedings responded. They generally supported the sufficiency of the *Perkins* record and approved the use of the *de minimis* theory to decide the issue of health effects. See ALAB-540, 9 NRC 428, 431-32 (1979).

Upon consideration of the papers submitted, we decided "to consolidate and hear first the cases where intervenors [were] actively participating and to hold the remainder in abeyance for the time being." 9 NRC at 433 (footnote omitted). We based that decision primarily on the generic nature of the radon issue, the manageable number of litigants involved, and the judgment that "moving along in the actively contested cases first [would] help insure against our overlooking relevant considerations when we [came] to review the remaining proceedings on our own initiative." *Id.* at 434. We then elected to hear all but three of the various intervenors' contentions, rejecting as beyond the scope of the proceeding two deficiencies concerning the cost of nuclear fuel and one dealing with radon released from the fly ash of coal. Finally, noting suggestions that some issues might be amenable to summary disposition, we permitted such motions before fixing a hearing date.

The applicants jointly filed a motion for summary disposition of all issues. The staff, however, moved to dispose summarily of only two. We ruled on those motions in ALAB-562, 10 NRC 437 (September 10, 1979), where we organized our discussion in terms of the "twenty-six deficiencies" which the *Sterling* and *Tyrone* intervenors had previously alleged. (See fn. 6,

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[6]Intervenors in *Sterling* and *Tyrone* — Ecology Action of Oswego and Northern Thunder, respectively — jointly alleged twenty-six specific deficiencies in *Perkins*. The *Peach Bottom-Three Mile Island* intervenors — Citizens for a Safe Environment and the Environmental Coalition on Nuclear Power — generally challenged the Licensing Board's approach to health effects and raised some specific objections to the *Perkins* record and decision. The *Hope Creek* intervenor, Mr. David Caccia, also questioned the adequacy of the Board's treatment of health effects. The remaining cases were either uncontested or elicited no response. As explained in the text, we decided to consolidate the five contested cases and hear them first. Both the *Tyrone* and *Sterling* projects were subsequently cancelled by their respective owners and we dismissed those proceedings for that reason. The *Tyrone* intervenors thereafter curtailed their participation in the radon proceeding. See our unpublished order of August 30, 1979. The *Sterling* intervenors continued to participate fully at our invitation. See *Rochester Gas and Electric Corp.* (Sterling Power Project, Nuclear Unit No. 1), ALAB-596, 11 NRC 867 (June 17, 1980); and our unpublished memorandum and order of June 23, 1980. The *Hope Creek* intervenor did not attend the evidentiary hearing; his sole concern involved the *Perkins* theory of health effects, which was beyond the scope of that evidentiary proceeding. See ALAB-562, 10 NRC 437, 441 fn. 10 (1979).

supra). We also noted that those deficiencies included all the relevant points which the *Peach Bottom-Three Mile Island* intervenors had raised. We held that an evidentiary hearing would be required on several matters where genuine issues of material fact remained. Those included radon emissions from mill tailings piles, underground mines, and open pit mines, as well as radon releases associated with water pathways and phosphate residues. We disposed summarily of the remaining issues for reasons explained there. *Id.* at 441, 443-47.

Because we intended to focus first on the matter of radon release rates and concentration levels, we deferred consideration of certain health effects aspects of one of the intervenors' deficiencies and reminded the parties that the forthcoming evidentiary hearing would not deal with that subject. *Id.* at 444-45. Perceiving no compelling reason for all the Appeal Panel members assigned to these proceedings to be present for the taking of further evidence, we selected three to preside at the hearing.[7] We noted, however, that once the evidentiary hearing was completed, all would participate in the consideration of the issues to be decided. *Id.* at 447-48.

We held a three-day evidentiary hearing in Harrisburg, Pennsylvania. Applicants participated jointly and made a consolidated evidentiary presentation. Intervenors consolidated their direct case but conducted cross-examination separately. The applicants and the staff submitted testimony concerning all of the alleged deficiencies in *Perkins;* intervenors presented evidence on some of those deficiencies and limited themselves to cross-examination on others.

Thereafter, the parties filed proposed findings of fact and conclusions of law. As requested, they addressed not only those issues on which summary judgment was denied, but the effect of more recent evidence received at the hearing bearing on one issue which we had previously disposed of summarily in ALAB-562.[8] The parties have also advised us of their views concerning how we should proceed with the health effects issue. Before turning to the question, we must decide the matter of radon releases and concentrations.

II. TECHNICAL INTRODUCTION

To provide a solid foundation for later discussion we begin with a brief technical introduction to the radon release issue.

[7]They were: Mr. Rosenthal, who served as Chairman; Dr. Buck; and Dr. Johnson. No party objected to our proceeding in this manner.
[8]Tr. 522-23, 527-28. This point is discussed at pp. 41-47, *infra*.

A. Basic Radiological Concepts

The isotope of interest in this proceeding, radon-222, is a member of the natural radioactive series that begins with uranium-238 and, through successive decay events, ends with the stable lead-206 isotope. The decay events in this series, which are presented graphically in Figure 1,[9] involve the emission of alpha or beta particles.[10] As can be seen, fourteen successive isotopes are formed in the decay of the original parent. The symbols within the boxes show the isotope present at each point. The Greek letters in Figure 1 indicate the decay process; *i.e.,* "α" and "β" symbolize radioactive decay by emission of alpha and beta particles, respectively.

Also listed in Figure 1 are the chemical symbols of the uranium decay series isotopes and their half-lives. The half-life of any nuclide is the time it takes a given quantity of radioactive atoms to decay to one-half that number. As Figure 1 shows, the half-life of uranium-238, the parent of the series, is 4.5×10^9 years (4.5 billion years) whereas the half-life of radon-222, the seventh isotope in the series, is 3.8 days.

The decay rate of any given isotope is equal to its decay constant times the number of atoms of that isotope which are present. The decay constant is the probability per unit of time that a nucleus of the isotope will decay. The radioactivity of an isotope is commonly expressed as the number of disintegrations of a given quantity of that isotope per unit of time. A unit commonly used in quantifying radioactivity is the curie, which is defined as 3.7×10^{10} (37 billion) disintegrations per second. For example, the radioactivity of one gram of uranium-238 is approximately 0.33 millionths of a curie or 0.33 microcuries.[11]

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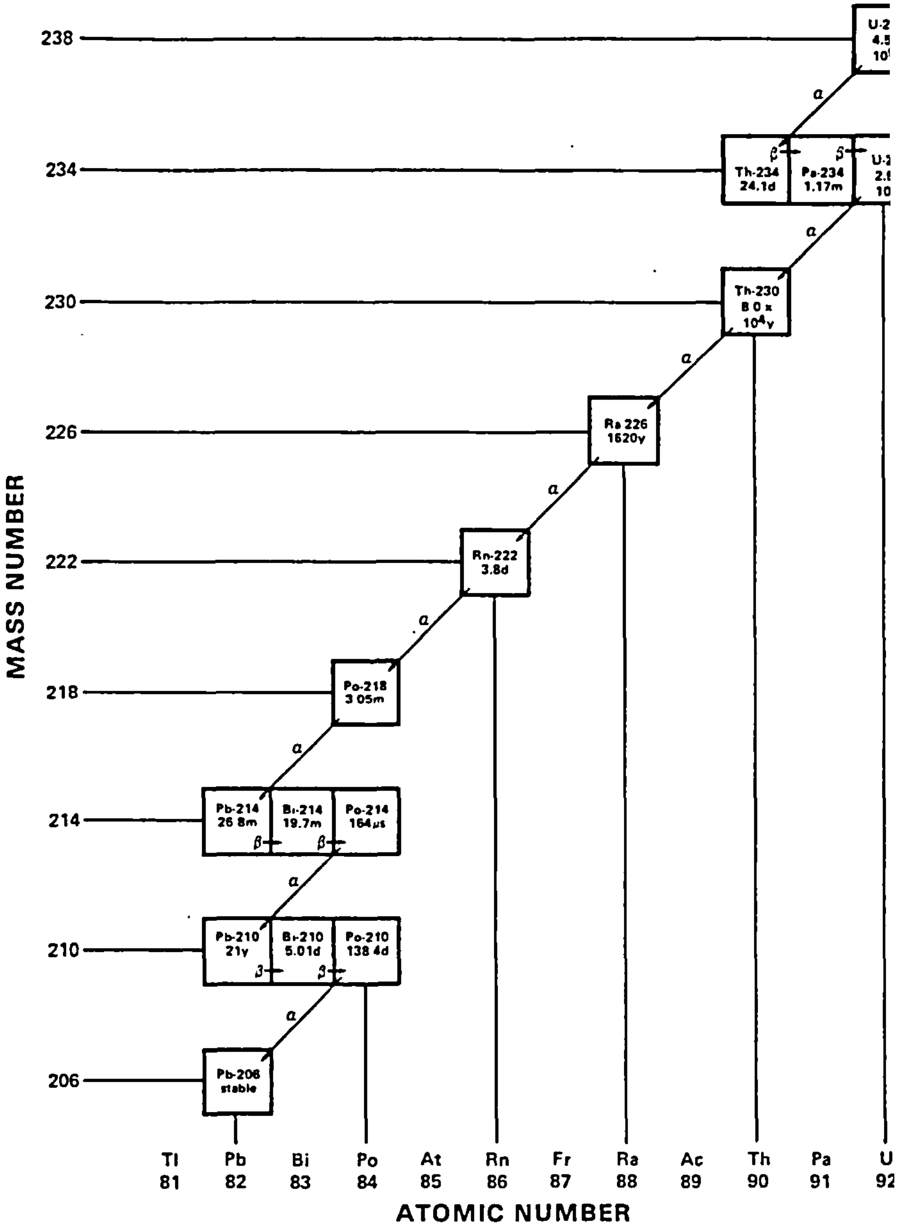
[9]This chart is adopted from the staff's "Draft Generic Environmental Impact Statement on Uranium Milling," NUREG-0511 (April 1979), Volume II, p. C-3. Although the document provides background information relevant to the staff's testimony before us, it includes subjects not at issue in these proceedings and, therefore, was not offered into evidence by the staff. We have taken official notice of the document. Tr. 521-22.

[10]An alpha particle is a positively charged nuclear subdivision of matter identical with the nucleus of a helium atom. It consists of 2 protons and 2 neutrons and is ejected at high speed in certain nuclear transformations. A beta particle is an electron ejected from the nucleus of an atom during radioactive decay.

[11]The development and discussion of these relationships can be found in any basic textbook on the subject or related subjects. See, for example, S. Glasstone and A. Sesonke, *Nuclear Reactor Engineering,* Chapter 2 (D. Van Nostrand Co., Inc., 1963). For a more fundamental discussion see I. Kaplan, *Nuclear Physics,* Chapter 10 (Addison-Wesley Publishing Co., Inc., 1955).



The number below the symbol for the isotope in each box is the half-life of that isotope in years (y), days (d), minutes (or seconds (s). The prefix μ stands for micro- and indicates multiplication by a factor of one one-millionth (10^{-6}).

Fig. 1 The Uranium-238 Decay Series.

If a parent isotope such as uranium-238 has very long half-life relative to those of the daughter isotopes produced in the series, in time there may exist a state of "secular equilibrium" in which the various daughter products are being produced as fast as they are decaying. In the absence of any physical or chemical separation,[12] the amount of each daughter (and hence its radioactivity) will remain constant and the radioactivity of each member of the decay chain will equal that of the long-lived parent. Because the half-life of uranium-238 is many orders of magnitude larger than those of other isotopes in its associated radioactive decay chain, the condition for secular equilibrium is met. Assuming that no physical or chemical separation has taken place, all of the decay series isotopes associated with 1 gram of uranium-238 should exist at the same level of radioactivity (*i.e.*, 0.33 μCi).

B. Characteristics and Sources of Radon

Radon-222 is the only member of the uranium-238 decay chain that is a gas. It is, moreover, a noble gas, which means that it does not combine with or attach to other elements. This property of radon allows it to migrate and escape from the soil — even without the assistance of activities such as mining and milling.[13] It also permits radon to be dispersed from its source by air currents.

As a noble gas, radon is not retained in the body long enough to be of major concern. But the radon daughter products are also radioactive and attach to particles suspended in the air.[14] When this air is inhaled, the daughter isotopes are deposited in the passageways of the lungs, where they may release the energy produced by their own decay. The absorption of this decay energy constitutes a radiation dose.

It must be recognized, however, that in the natural environment everyone is enveloped in radon. This is because uranium is widely distributed over the earth in the soils, rocks and oceans in varying concentrations and, as a consequence, radon is similarly dispersed in

[12]A number of non-nuclear natural physical processes (such as weathering and sedimentation) may contribute to differential movement of the various elements of the chain and can thus disrupt this equilibrium; perfect secular equilibrium is, therefore, not always found in nature.

[13]Radium-226 is the immediate precursor of radon-222 in the uranium-238 decay series. When an atom of radium decays in the soil, it emits an alpha particle and is transformed into an atom of radon. If the radium atom is sufficiently close to the surface of the grain of soil in which it is imbedded, the radon atom may recoil out of the grain and into the air which permeates the soil. The radon then begins its travels to the surface, primarily through the processes of diffusion and convection.

[14]The decay of radon results in a series of four relatively short-lived daughter products plus several longer-lived daughters. The short-lived daughters are: polonium-218 (half-life of 3.05 minutes), lead-214 (26.8 minutes), bismuth-214 (19.7 minutes) and polonium-214 (0.16 thousandths of a second).

various materials. Indeed, radon is released from such ordinary building materials as stone, brick, cement and gypsum board. It is also released from the burning of fossil fuels such as coal. Although radon is widespread in the earth's crust and routinely escapes from ordinary soils, mining and milling activities enhance the release of radon to the environment — largely by increasing the exposure of uranium ore to the atmosphere. These activities are the major contributors to releases of radon attributable to the uranium fuel cycle.

Mining of uranium for nuclear fuel is either done by open-pit or underground mining methods. As the names imply, open-pit mining consists of extracting ore directly from the surface of the earth, whereas underground mining involves tunneling to deeper ore bodies.

The milling process involves three major stages: (1) ore handling and preparation (including grinding and interim storage); (2) mill concentration (extraction of uranium from the ore by means of a leaching technique); and (3) product recovery (a process which utilizes chemical precipitation and results in a uranium concentrate called "yellowcake"[15]).

Radon releases occur at all stages of the milling process. Because the mill tailings (*i.e.*, residue of the milling process) contain substantial amounts of thorium and radium, as well as about ten percent of the uranium originally present in the ore, releases from the mill tailings piles occur both during and following the active milling stage.

Finally, because the calculation of the rate at which radon migrates and escapes from solid materials such as soil or mill tailings piles is a matter of considerable importance in this decision, a brief discussion of the methods used in estimating and describing radon release rates may be helpful.

C. Estimation of Radon Releases

Theoretical estimates of radon migration and its rate of release from soil masses and mill tailings piles depend on several variables. The quantity of radon escaping from soil or a similar substance to the atmosphere is usually described in terms of a "flux", *i.e.*, the amount of radon emanating per unit area per unit of time. The radon flux can be calculated by use of the well-established techniques of diffusion theory.[16] In applying the theory to radon diffusion, the soil or other material is viewed as a matrix of

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[15]Yellowcake is a term of art given to the product of the refining process used in the milling operation. Yellowcake contains a high concentration (90 percent or more) of U_3O_8.

[16]Diffusion theory deals with the spreading or scattering of matter under the influence of a concentration gradient; *i.e.*, variations of concentration in space. See, *e.g.*, 4 *McGraw-Hill Encyclopedia of Science and Technology*, 154-59 (1971).

impermeable solid particles containing a maze of gas or liquid filled capillaries through which the diffusion of radon gas takes place.[17]

There are several variables in the diffusion theory equation for computing the rate of migration of radon through materials. Of interest here are the concentration of radon available for diffusion, the void fraction of the medium, and the effective bulk diffusion coefficient. The concentration of radon available for diffusion is partially dependent on the concentration of its immediate parent, radium-226, which is generally assumed to equal the secular equilibrium concentration of the uranium-238 present in a given grade of ore. The radon concentration also depends on the void fraction and the "emanating power" of the medium. The void fraction (or porosity) is that part of the total volume of the medium which is empty of solid particles. The emanating power, often called the emanation coefficient, represents that fraction of radon produced within the grains of surrounding material which escapes into the void space and is thus free to migrate or diffuse. The diffusion coefficient relates the rate of flow and the concentration of a material within a medium. When the concentration of the material varies within the medium, the diffusion coefficient will determine the net rate of flow of that material within the medium.

Throughout our decision, radon emissions are frequently expressed in terms of curies released per annual fuel requirement (AFR). An AFR is defined as the amount of fuel required to operate the 1000 MWe model light water reactor at 80 percent capacity for one year.[18] The AFR can be expressed in units of tons of uranium ore, yellowcake, or U_3O_8.[19] Actual reactors may use more or less than one AFR each year depending on their size and characteristics of operation. For example, a 500 MWe reactor operating for one year (at 80 percent capacity) would require approximately one-half of the fuel needed to operate the model 1000 MWe plant for one year — thus it would need only one-half of an AFR to operate for one year. Expression of radon releases in terms of AFRs permits the allocation to a single nuclear power reactor of a representative portion of the radon releases from the uranium fuel cycle for purposes of NEPA cost-benefit analyses.

[17]See *e.g.*, M. Sears *et al.*, "Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing 'As Low as Practicable' Guides — Milling of Uranium Ores," ORNL-TM-4903, Vol. 1, pp. 144-50 (Oak Ridge National Laboratory, 1975).

[18]See 10 CFR Part 51, Table S-3. The AFR is synonymous with another term — the reference reactor year (RRY).

[19]See fn. 15, *supra*. As used in *Perkins* and in this proceeding, one AFR is equal to 271,000 metric tons of uranium ore at 0.1 percent ore grade (which can also be expressed as 271 metric tons of U_3O_8), or a mill output of 245 metric tons of U_3O_8.

III. EMISSIONS ATTRIBUTABLE TO URANIUM MINING

Intervenors advanced several deficiencies with regard to the *Perkins* record on mining. They questioned the evidentiary basis for the Licensing Board's assumptions that underground mines would be sealed and open pit mines reclaimed. They also pointed out that the *Perkins* proceeding did not address the extent to which unsealed or improperly sealed underground mines could continue to emit radon. Finally, intervenors pointed out that the *Perkins* record on open pit mines was inadequate because it gave no consideration to emissions from mine overburden and contained nothing more than the witnesses' rough approximations of what radon releases might be. We discuss these issues according to the type of mine involved, beginning with underground mines.

A. Underground Mines

In *Perkins,* staff witness Ralph M. Wilde[20] estimated that a model underground mine would release 4060 curies of radon per AFR during active mining. He calculated that value from an estimate of the concentration of radon gas in the air ventilated from several underground mines during the time required to extract sufficient ore to produce the uranium for one AFR. He testified that, after the mine was closed down, ventilation would cease and radon emissions would "essentially go to zero." *Perkins* Tr. 2541-42. The applicant's witness, Dr. Morton I. Goldman,[21] stated that he had reviewed the staff's estimate and considered it to be reasonable. Goldman, p. 1, fol. *Perkins* Tr. 2266. The Licensing Board adopted Mr. Wilde's estimate and accepted his suggestion that underground mines would cease emitting radon when mining operations stopped. 8 NRC at 90.

The *Perkins* release rate of 4060 curies per AFR from operating underground mines was not at issue in this proceeding because we had summarily disposed of intervenors' only deficiency touching upon that subject in ALAB-562. For reasons we explain in the next two sections,

[20]Mr. Wilde also testified for the staff at the consolidated radon hearing. He joined the AEC in 1973 and is currently a program assistant to the Director of the Division of Fuel Cycle and Material Safety in the NRC's Office of Nuclear Material Safety and Safeguards. He holds a B.S. degree in chemistry from Brigham Young University. His experience includes four years as an analytical chemist and metallurgist for uranium ore processing mills and fourteen years as Radiation Safety Director for the Anaconda Company's uranium mining and milling operations in New Mexico.

[21]Dr. Goldman also testified for the applicants at the consolidated radon hearing. He joined the Nuclear Utilities Services Corporation in 1961 and is currently its Senior Vice President and Corporate Technical Director. He holds advanced degrees in sanitary and nuclear engineering from the Massachusetts Institute of Technology. For some 30 years, he has engaged in original research and technical consultation on a variety of nuclear health, environmental and safety issues, including radioactive waste management and treatment.

however, we must reject the *Perkins* finding in favor of a more recent and more soundly based value which was fully explored by all parties at the evidentiary hearing (see pp. 508, 510-512, *infra*). We must also reject the *Perkins* Board's conclusion regarding radon emissions from inactive underground mines. As we noted in ALAB-562, the *Perkins* record failed to reveal the extent to which underground mines could and would be sealed; moreover, it did not contain sufficient information on the extent to which an unsealed mine could continue to emit radon.[22] Before us, witnesses for the applicants and staff presented testimony on all of these topics; intervenors participated by way of cross-examination of those witnesses.

To calculate a value for radon releases from operating underground mines, Mr. Wilde used survey data from a sample of 27 underground mines in New Mexico, Wyoming, Colorado, and Utah, representing approximately 63 percent of the total production of underground uranium mines in the United States. Tr. 345-46. He had obtained those data shortly before the hearing and presented the parties and the Board with a revised version of his prefiled testimony so as to be able to include the new information.[23] Tr.

[22]Intervenors advanced Deficiency 3 on those subjects, which asserts:

> 3. In the long run, radon emissions depend on the extent to which underground [mines] are sealed and open pit mines are reclaimed. The NRC has no jurisdiction over mines. In Perkins Staff and Applicant [witnesses referred] to state laws which require sealing and reclamation as adequate to insure the cessation of emissions after [the mines'] useful lives. In testimony on June 27, 1978, before the House Subcommittee on Energy and Environment, Betty Perkins from the New Mexico Energy and Mineral Department, indicated in New Mexico abandoned mines have been improperly sealed, have contaminated the soil, and have left ore storage piles exposed. Measurement at abandoned mines shows gamma radiation levels 10 to 100 times above background, a fact which demonstrates the existence of radiologic pathways for radon. In view of the actual facts regarding abandoned mines, it is [incumbent] upon the NRC to make a detailed examination of the statutory standards imposed on the operators of mines, the penalties [for] failure to comply with such standards, and each state's enforcement experience before leaping to unwarranted conclusions regarding the efficacy of state regulation of mines.

Response of Ecology Action of Oswego and Northern Thunder, Inc. to ALAB-509, pp. 9-10 (filed Feburary 19, 1979).

[23]Mr. Wilde's prefiled testimony referenced two reports prepared for the NRC by Battelle Pacific Northwest Laboratory as the source of the underlying data for his calculated release rates from underground and open pit mines. They are: P. Jackson *et al.*, "Radon-222 Emissions in Ventilation Air Exhausted from Underground Uranium Mines," NUREG/CR-0627 (September 1979); and K. Nielson *et al.*, "Prediction of the Net Radon Emission from a Model Open Pit Uranium Mine," NUREG/CR-0628 (September 1979). The revised report on underground mines (containing additional data) is: P. Jackson *et al.*, "An Investigation of Radon-222 Emissions from Underground Uranium Mines," NUREG/CR-1273 (Progress Report 2, February 1980). These reports are not in the record, but the reliability of the data which Mr. Wilde obtained from them was not challenged. The same can be said for other reports from which expert witnesses for all parties obtained source information upon which to base their testimony.

344-45. Mr. Wilde calculated an average radon release rate from active underground mines by dividing "the total annual release from all of the mines sampled ... by the total annual production from those same mines" and multiplying by the amount of U_3O_8 needed to produce one AFR. Tr. 346. The total radon released from the 27 mines studied was about 150,000 curies per year, the annual production from those mines was 5760 tons of U_3O_8 (or 5225 metric tons of U_3O_8), and the size of the AFR was 271 metric tons, Tr. 394. Assuming a 30-year lifetime for an underground mine, Mr. Wilde obtained a release rate for active mining of 260 curies per year per AFR. Wilde, p. 12, fol. Tr. 355. He explained that an additional 10 curies per year per AFR would be released from waste rock stored on the surface near the mine for a total release rate of 270 curies per year per AFR, or "aproximately 8000" curies per AFR. *Id.;* Tr. 363.

Mr. Wilde testified before us that radon would be released from an unsealed inactive underground mine as a result of air circulation due to natural convection.[24] Wilde, p. 11, fol. Tr. 355. He explained that because of uncertainties in computing the amount and direction of natural convection air flow,[25] it is extremely difficult to predict radon release rates for underground mines once mining operations have ceased. Consequently, he preferred to make the very conservative assumption that those mines would exhaust radon at the same rate as active underground mines under conditions of forced ventilation. *Id.* at 12. He therefore attributed that same release rate of 270 curies per year per AFR to unsealed, inactive underground mines. *Ibid.*

Further record evidence suggests that Mr. Wilde's attribution of such a release rate to unsealed underground mines is overly conservative because it is based on an unrealistic assumption that emissions from inactive mines by natural convection will equal those from active mines with forced ventilation. In response to cross-examination about radon emissions from existing underground mines, Mr. Wilde gave release rates for two unsealed inactive mines, called the "Barbara J" and "Mesa Top," which he considered representative of best and worst case conditions. Tr. 357-61.

Mr. Wilde used measured natural convection air flows given for those two mines in an EPA study, together with ore production data he obtained from the Grand Junction office of the Department of Energy, to calculate a release rate of 1-2 curies per year per AFR for the Barbara J mine (Tr. 358) and 70-80 curies per year per AFR for the Mesa Top mine (Tr. 361). He

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[24]Natural convection occurs because of variations in air density caused primarily by differences in temperature.

[25]The natural convection pattern within a mine depends on such factors as differences in elevation between mine openings and the temperature differential between the air inside and outside the mine. Wilde, p. 11, fol. Tr. 355.

explained that air flows from the Barbara J. were so small that they had to be measured by injecting a puff of smoke and recording how long it took to rise to the top of the hoisting and ventilation shafts. Tr. 357-58. In contrast, the Mesa Top mine represented a "near-worst case for an abandoned inactive mine" because of its interconnection with several other active and inactive mines and the existence of open paths for air flow through "adit" entries.[26] Tr. 359-61. Mr. Wilde stated that radon releases were measured while air was flowing out of the Mesa Top shaft but that during the summer he would expect the flow to be reversed and the emissions to cease. Tr. 362. He explained that the flow could be reversed twice daily in response to the appropriate day-to-night temperature changes. *Ibid.* This suggests that the release rate of 70-80 curies per AFR per year may overstate actual radon emissions from that particular mine, because it has not been adjusted for periods when the air is not flowing out of the mine shaft.

Dr. Goldman's testimony on this subject further supports the conclusion that Mr. Wilde's estimate of 270 curies per year per AFR is extremely conservative. Using data from an EPA study of 2100 inactive underground mines,[27] Dr. Goldman calculated a radon release rate representative of inactive, unsealed underground mines of 36 curies per year per AFR. To reach this figure, he assumed that all radon released to the air inside the mine was exhausted to the atmosphere by natural convection. Goldman, pp. 25-26, fol. Tr. 441.

Dr. Goldman and Mr. Wilde each indicated that estimates of radon releases from abandoned underground mines would be considerably overestimated if based on the assumption that actual releases would equal those from an active underground mine with the ventilation fans in operation. Dr. Goldman stated that he would expect such a value to be "greatly in excess of that actually released." Goldman, p. 26, fol. Tr. 441. Mr. Wilde expressed his opinion that actual releases would be "appreciably" lower. Wilde, p. 16, fol. Tr. 355. He stated that in his judgment, actual radon releases from abandoned, unsealed underground mines would be "much closer to 10 curies per year per AFR than 270 curies per year per AFR." *Ibid.* He considered it highly improbable that natural air flows would be sufficiently large to exhaust to the atmosphere the same amount of radon that would be released under conditions of forced ventilation. *Id.* at 14. He acknowledged that at Mesa Top, conditions were such that about half the amount of radon released during normal operation would escape due to natural convection; the rest would decay underground. Tr. 419. He

[26]An adit is a horizontal tunnel driven into the side of a hill (Tr. 359.)

[27]R. Blanchard *et al.*, "Potential Health and Environmental Hazards of Uranium Mine Wastes — Draft Report," Office of Radiation Programs, U.S. EPA (September 1979).

continued to maintain that his original estimate of 267 curies per year per AFR was "conservatively high." Tr. 429.

We think that value is more appropriately viewed as a conservatively calculated upper limit than as a reasonably accurate estimate of radon emissions from abandoned and unsealed underground mines. Dr. Goldman's value of 36 curies per year per AFR, calculated from data on 2100 inactive underground mines in the United States, is a more soundly based and realistic estimate. Indeed, the reliability of that figure is further supported by Mr. Wilde's "worst case" estimate, which was only 70-80 curies per year per AFR. Thus, on the basis of the present record, we think that the reasonably conservative approach is to adopt 80 curies per year per AFR as the radon release rate for unsealed, inactive underground mines. To that value must be added the 10 curies per year per AFR that Mr. Wilde testified was attributable to the waste rock pile on the surface near the mine (see p. 501, *supra*), yielding a total of 90 curies per year AFR. Adopting this result is clearly more reasonable than accepting a figure several times larger based on what amounts to an unwarranted assumption that the ventilation fans will continue to be powered and run in underground mines after operations have ceased.

According to the testimony of both Mr. Wilde and Dr. Goldman, abandoned underground mines can be sealed easily and effectively by filling their hoisting and ventilation shafts with soil or waste rock. In addition, shaft openings can be sealed with concrete plugs. Wilde, pp. 5-6, fol. Tr. 355; Goldman, p. 24, fol. Tr. 441. Mr. Wilde stated that radon emission rates from properly sealed mines would be negligible; the only measurable radon releases associated with sealed underground mines would be the 10 curies per AFR per year from waste rock remaining on the surface. Wilde, pp. 8-9, 12, 15, fol. Tr. 355.

There is nothing in the record to contradict Mr. Wilde's opinion that "the technical feasibility of reclaiming worked out mines has already been adequately demonstrated." Wilde, p. 6, fol. Tr. 355. Similarly, no party challenged his figure of 10 curies per year per AFR for radon releases attributable to properly sealed underground mines and their associated wastes. We are inclined to accept that value as reasonable. We cannot adopt it, however, because the record contains no reliable information about the extent of reclamation currently being required or practiced.

The NRC has no regulatory authority over uranium mining or mine reclamation.[28] Mr. Wilde observed that responsibility for regulation of

[28]*See Homestake Mining Co.* v. *Mid-Continent Exploration Co.*, 282 F.2d 787, 791 (10th Cir. 1960); *Rochester Gas and Electric Corp.* (Sterling Power Project, Unit 1), ALAB-507, 8 NRC 551, 554 (1978), *aff'd. sub nom. Ecology Action of Oswego* v. *NRC*, No. 78-1885 (D.C. Cir. Mar. 12, 1980).

uranium mines (including their reclamation) rests primarily with the state in which the mine is located. Dr. Goldman testified for the applicants that his staff had surveyed five mining companies in Colorado, Wyoming, and New Mexico. None of them had sealed mines; some had made commitments to their State agencies to seal underground mine shafts, or planned to do so in the future. Goldman, p. 24, fol. Tr. 441. Neither witness was qualified to testify concerning mining reclamation laws; moreover, no party presented testimony from state regulatory officials or mine operators to demonstrate that reclamation efforts can reasonably be anticipated. We simply do not know, for example, what percentage of ore production in the United States has come from currently inactive underground mines which have been or will be sealed, or what percentage of ore production is now being obtained from operating mines subject to laws or regulatory requirements ensuring that they will eventually be sealed. Lacking the necessary facts, we can only assume that the uranium taken from underground mines to produce fuel for the reactors under consideration is attributable to mines that have not been or will not be reclaimed.

B. Open Pit Mines

Mr. Wilde testified in *Perkins* that there was "no reliable information available upon which to base estimates" of radon releases from open pit mines. Wilde, p. 7, fol. *Perkins* Tr. 2369. Because open pit mining then constituted about half of all uranium mining activity,[29] the *Perkins* Board insisted that at least an upper bound figure for emissions from that source be placed in the record. 8 NRC at 90-91. Mr. Wilde made a number of assumptions and calculated a radon release rate for a hypothetical open pit mine of approximately 100 curies per year per AFR. Dr. Goldman made a similar calculation and arrived at a release rate ranging from 100 to 200 curies per year per AFR. *Perkins* Tr. 2640.

Based on those concededly rough estimates, the Licensing Board assumed that the amount of radon released from open pit mines could be as high as 200 curies per year per AFR. 8 NRC at 91. It reasoned that, if the hypothetical open pit mine was reclaimed at the end of its 20 year operating life, the total radon releases attributable to the active mining period would be 4,000 curies per AFR. The Board then concluded that because the release rates attributable to active mining, whether from underground or

[29]Mr. Wilde had testified in *Perkins* that he expected the amount of uranium ore obtained from open pit mines to decline in the future. *Perkins* Tr. 2551. Based on more recent data, he estimated in this proceeding that only 40 percent of the total production of U_3O_8 now comes from open pit mines; the remaining 60 percent comes from underground mines. Wilde, p. 9, fol. Tr. 355.

open pit mines, were nearly the same, it would not matter which type of mine produced the uranium to fuel a particular plant. *Ibid.*

With respect to long-term releases from inactive open pit mines, Mr. Wilde had testified in *Perkins* that he was not qualified to address the extent of reclamation then being required. He added that he was therefore unable to quantify what those long-term radon emissions might be. *Perkins* Tr. 2550-56.[30] Testifying for the intervenors in *Perkins*, Dr. Chauncey Kepford[31] assumed that the pits would remain open forever. He then calculated that over the next ten billion years, for example, 6×10^{13} curies of radon would be released if all of the fuel needed to operate the *Perkins* plant during its 30 year life were to come from unreclaimed open pit mines. Kepford, Table 1, fol. *Perkins* Tr. 2820. The Licensing Board declared that it considered Dr. Kepford's assumptions unrealistic because it was unreasonable to suppose "that society will permit such open sores on our landscape for all future time." 8 NRC at 92. Rather, the Board expressed its own speculative judgment that reclamation would probably occur within 100 years after active mining had ceased. *Id.* at 91-92.

As intervenors pointed out, the *Perkins* record on open pit mines is very "sketchy."[32] We stated in ALAB-562 that uncertainty remained regarding

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[30]Dr. Goldman stated that of the five states in which significant amounts of uranium are mined, only three had surface mining reclamation requirements applicable to uranium mines. *Perkins* Tr. 2639.

[31]Dr. Kepford cross-examined applicant and staff witnesses on behalf of intervenors in *Perkins;* he also furnished testimony for that proceeding in deposition form. He represents that he has devoted himself full time since 1970 to "the problems of nuclear power" and has testified before various state and federal legislative bodies and administrative agencies. He holds a Ph.D. in chemistry from the University of Calgary, Alberta, Canada. His prior experience includes two years as a research scientist for United Aircraft Research Laboratories and two years as an assistant professor of chemistry at the York Campus of Pennsylvania State University.

[32]Intervenor's Deficiency 3, concerning reclamation of underground and open pit mines, appears at fn. 22, *supra.* Deficiencies 4 and 5 on open pit mines are as follows:

> 4. The testimony in Perkins regarding emissions from open pit mines is extremely sketchy. Mr. Wilde at page seven of his affidavit states, "For open pit mines ... there is just no reliable information available upon which to base estimates of radon release." Pages 2543 through 2558 [of] the transcript enumerate many of the [uncertainties] regarding emissions from open pit mines. Nevertheless, at page 2610 of the transcript, Mr. Wilde performs a "quick and dirty" computation of emissions using a model open pit mine. He makes what is an apparently completely arbitrary choice of a mine which covers one square mile. He computes a release of 100 curies/yr/AFR. Apparently the Board in Perkins was somewhat skeptical about Mr. Wilde's calculation since in paragraph 13 of the Perkins decision the rate of emission from open pit mines was doubled to 200 curies/yr/AFR.
>
> The Sweetwater DES indicates a release rate of 6090 curies per year. The Sweetwater mine will have a capacity sufficient to produce 410 MT [of] yellow cake per year during its estimated 15 year life. Using the Staff figure of 245 MT [of]

radon emissions from both reclaimed and unreclaimed open pit mines. In particular, we were concerned that "releases from reclaimed mines [might] be higher than expected, due to the physical rearrangement of the overburden as it is replaced in the pit." 10 NRC at 442-43 (footnote omitted). We also noted that one of intervenors' deficiencies apparently concerned radon releases from operating open pit mines as well as from those that had been shut down and left unreclaimed. *Id.* at 443, fn. 21.

Mr. Wilde testified before us that, since the *Perkins* proceeding, the data base for radon releases from open pit mines had improved considerably and now included information that could be used to develop representative mine models and radon release rates for both the active mining stage and the period following shutdown of the mines. Wilde, pp. 7-8, fol. Tr. 355. Based on the new data, Mr. Wilde calculated a revised estimate for radon emissions from open pit mining and combined it with the 8000 curies per AFR released from underground mining to obtain a release rate of approximately 5200 curies per AFR as the emissions attributable to the uranium mining industry as a whole. Tr. 348-49. He explained that the new industry value was based on the assumption that 60 percent of all uranium ore was derived from underground mines and that the remaining 40 percent came from open pit mines. Tr. 349. He also indicated that it represented an increase of about 25 percent from the *Perkins* estimate of 4060 curies per AFR. Tr. 349.

Mr. Wilde neglected to specify the staff's new estimate for radon emissions from open pit mining. But he did explain how he calculated that value, and it can easily be derived from the figures he presented in his testimony. Mr. Wilde testified that the radon release rate for open pit mining given in the Battelle report (fn. 23, *supra*) was 630 curies per AFR. Tr. 384. But he indicated that all of the Battelle figures are based on a smaller AFR; *i.e.*, 182 metric tons of ore rather than the 271 metric tons

yellow cake per AFR would result in an annual release rate for the Sweetwater mine of approximately 250 curies/yr/AFR. This is another example of the actual facts deviating from the Staff's assumptions regarding radon emissions.

5. Also with respect to open pit mines, the Perkins record gives no consideration to emissions from overburden. Testimony before the Senate Subcommittee on Energy Production and Supply on July 24 and 25, 1978, indicates the overburden has a volume of 8 to 35 times the volume of the mine. Therefore all of the overburden cannot be returned to the mine. The overburden has as much as 10% of the radioactive concentration of mill tailings. South Dakota, with a mine reclamation law on the books, has former mining areas that are now sterile and bare. The [overburden] has been indiscriminately piled on the landscape just like mill tailings.

Response of Ecology Action, fn. 22, *supra*, at 10-11.

which Mr. Wilde used throughout his testimony. Tr. 364-65. Accordingly, Mr. Wilde explained that the Battelle release rates must be increased in direct proportion to the relationship between the different AFRs (271 to 182 or about 1.5 to 1) in order to obtain the staff's release rates. Tr. 363-65, 391-92. Thus, the 630 curies per AFR corresponds to about 945 curies per AFR, which Mr. Wilde then apparently rounded to 1000 in calculating his release rate of 5200 curies per AFR for the uranium mining industry as a whole.

For inactive open pit mines, Mr. Wilde calculated radon releases from both reclaimed and unreclaimed mines because of the lack of assurance that all worked-out mines would be reclaimed. Wilde, pp. 6-7, fol. Tr. 355. For the same reason we were unable to adopt the staff's estimate of the release rate for sealed underground mines (pp. 503-504, *supra*), we are unable to make a finding for reclaimed open pit mines. Because the record does not accurately reflect the degree to which open pit mines are reclaimed, we simply have insufficient information to make appropriate use of radon release rates for reclaimed open pit mines. Therefore, we must assume that all inactive open pit mines will remain unreclaimed. Our discussion of the relevant testimony on this point merely serves to illustrate the degree of reduction in radon emissions that can be achieved in the future if open pit mines are properly reclaimed.

For a reclaimed open pit mine, Mr. Wilde used a model mine consisting of seven successively mined pits and their associated wastes. In this model, approximately 85 percent of the mine volume is refilled with overburden[33] containing 20 parts per million (ppm) of U_3O_8; the remaining overburden (about 15 percent) is piled on the surface near the mine. There is also a surface pile of sub-ore[34] containing 150 ppm of U_3O_8 that is being saved for possible commercial use in the future. *Id.* at 8. Mr. Wilde explained that radon emissions from the model reclaimed mine are attributable to overburden fill in six pits, sub-ore and overburden exposed in the last unfilled pit, and sub-ore and overburden dump piles. *Ibid.* He computed the combined long-term radon releases from those sources to be about 40 curies per year per AFR. *Ibid.*[35]

Using the same model and assuming that the worked-out pits of the model mine were not refilled, Mr. Wilde also prepared an estimate

[33]Overburden is the material that lies on top of the ore body being mined.

[34]The sub-ore contains low-grade uranium not now economically recoverable.

[35]Dr. Goldman explained that the model mine Mr. Wilde described was created from averaged characteristics of eight major open pit mines in the Casper, Wyoming area and included radon release measurements made at one of those mines. Based on the same model, Dr. Goldman calculated the resulting radon emissions for the partially reclaimed model open pit mine to be 39 curies per year per AFR; essentially the same result as Mr. Wilde reached. Goldman, p. 28, fol. Tr. 441.

representative of long-term radon releases from an unreclaimed open pit mine. That model projects radon releases from the overburden and sub-ore exposed in seven unfilled pits, seven overburden piles, and a sub-ore pile. Mr. Wilde calculated a radon release rate of approximately 80 curies per year per AFR for that source. *Id.* at 9-11, 15.

Dr. Goldman's calculations are derived from an EPA study of over 900 abandoned open pit mines.[36] Goldman, p. 28, fol. Tr. 441. He used a model surface mine from which all of the sub-ore was placed on top of the overburden pile. From EPA estimates of the amounts of ore, sub-ore and waste removed from the mine, Dr. Goldman calculated a long-term release of 100 curies per year per AFR for a completely unreclaimed open pit mine and its associated wastes. To reach this figure, he assumed an average ore grade during the life of the mine of 0.29 percent and a 95 percent recovery rate. *Id.* at 29.

Mr. Wilde's and Dr. Goldman's present estimates are more soundly based than those in their *Perkins* testimony. As mentioned previously, they had offered some impromptu calculations for open pit mines only at the Licensing Board's insistence. The value of 4060 curies per year which Mr. Wilde gave in *Perkins* for active mining was based on data from underground mines only, but he nevertheless applied it to open pit mines. Thus, with respect to radon releases from active open pit mines, we find no basis for adopting either the staff's estimate of 4060 curies per AFR or the Licensing Board's assumption of 4000 curies per AFR from active open pit mines (see pp. 499, 504, *supra*). In contrast, the staff's new value of 1000 curies per AFR is based on considerably better data for active open pit mining. Wilde, p. 7, fol. Tr. 355; Tr. 384-86. We adopt it for use in this proceeding.

Regarding radon releases from abandoned, unreclaimed open pit mines, Mr. Wilde's and Dr. Goldman's revised conclusions are reasonably similar. Mr. Wilde derived his estimate of 80 curies per year per AFR from a model representing averaged characteristics of eight major open pit mines, whereas Dr. Goldman calculated his figure of 100 curies per year per AFR from data in an EPA study of over 900 unreclaimed open pit mines. No evidence was presented to controvert either estimate. Because it is slightly more conservative (*i.e.*, higher) and based on a more representative sample, we shall adopt Dr. Goldman's value of 100 curies per year per AFR for radon emissions from unreclaimed open pit mines.

We must reject Mr. Wilde's estimate for radon releases from a reclaimed open pit mine. As we have seen (p. 507, *supra*), the record contains no

[36]Blanchard *et al.*, fn. 27, *supra.*

reliable information about the extent to which abandoned mines will be reclaimed. Therefore, we must assume that all open pit mines are unreclaimed.

In summary, we adopt (1) radon releases of 1000 curies per AFR attributable to open pit uranium mining and (2) releases of 100 curies per AFR per year for inactive and unreclaimed open pit mines and their associated wastes.

C. Changes in Radon Release Values

From the preceding sections, it is clear that the staff's values for radon releases from mining have undergone a number of revisions. In fact, it was in response to the discovery of a large discrepancy between actual radon emissions attributable to the uranium fuel cycle and the value originally set forth in Table S-3 (of 10 CFR Part 51) that the Commission deleted the radon value from that table and ordered that further proceedings be undertaken. Our analysis of the changing values suggests that as more data have become available, better and more reliable estimates have evolved.

For example, in *Perkins*, the staff presented Homer Lowenberg and Jack E. Rothfleisch to explain the basis for the radon value in Table S-3.[37] Mr. Rothfleisch testified that the 75 curies per AFR set forth there represented releases from mill tailings piles only during active milling and did not include any estimate of radon emissions from mining. Rothfleisch, p. 3, fol. *Perkins* Tr. 2369. According to Messrs. Rothfleisch and Lowenberg, that approach was adopted because, although data for radon releases were sparse (especially for releases from mines), the information that was available at that time suggested that the releases were not significant. Lowenberg at 2, 7; Rothfleisch at 4. The staff later provided more complete estimates of radon emissions from uranium mining and milling for the *Perkins* proceeding.

In response to the concerns set forth in ALAB-562 and intervenors' allegations of deficiencies in the *Perkins* record, the staff made additional changes in its estimates of radon releases. The more significant changes involve releases from uranium mines, both underground and open pit. New survey data, which were not available for the *Perkins* proceeding; revealed that the staff's estimate of radon emissions from open pit mines was far in excess of actual release rates. Consequently, the staff recalculated that

[37]Mr. Lowenberg was then Assistant Director for Operations and Technology for the Division of Fuel Cycle and Material Safety in the NRC's Office of Nuclear Material Safety and Safeguards. Mr. Rothfleisch was then a senior chemical engineer with the Fuel Processing and Fabrication Branch of that same office. Mr. Rothfleisch participated in the calculation of the original radon release value for Table S-3 under Mr. Lowenberg's supervision. Rothfleisch, p. 1, and Lowenberg, p. 1, both fol. *Perkins* Tr. 2369.

value. Tr. 384. For underground mines, the staff's *Perkins* estimate included the assumption that the amount of radon released was proportional to ore grade, which later data proved to be incorrect. *Ibid.* In these circumstances, we do not view the changes from prior values as providing serious cause to doubt the accuracy of the more recent release rate estimates.

This brings us to a matter to which we alluded briefly at the outset of our discussion on underground mining, where we noted that we were rejecting the *Perkins* estimate of 4060 curies per AFR from active mining in favor of a more recent value explored at the evidentiary hearing. In Deficiency 1, intervenors had claimed that the staff's methodology in calculating the *Perkins* value was flawed because "it is not possible to demonstrate a fixed correlation between [tons of uranium] ore mined and [curies of] radon released."[38] In other words, intervenors maintained that calculating a value

[38]The complete text of Deficiency 1 is as follows:

> 1. Staff testimony, e.g., affidavit of R. M. Wilde, assumes a fixed correlation between uranium ore mined and curies of radon released. As per Wilde, the release per MT of ore is 1.48×10^{-2} curies Rn-222. In fact, it is not possible to demonstrate a fixed correlation between ore mined and radon released. A report dated August 4, 1978, from Battelle Pacific Northwest Laboratories to Dr. Harry Landon of the NRC states:
>
> It is evident that there is a much [closer] relationship between approximate areas of the mine ventilated and radon 222 emitted than between ore production and radon 222 emission. Thus, a simple [extrapolation] on the basis of curies per ton of ore could lead to erroneous conclusions about the total emission rate from mines.
>
> [Another] report to Mr. Landon from Battelle, this one dated February 6, 1978, and entitled *Literature Review of Radon 222 Emission Rates from United States Uranium Mines*, indicates the radon release rate from mining varies from 1.8 curies to 48 curies per ton of yellow cake. Using Magno's formula of 245 MT yellow cake per AFR, the higher figure would result in a release rate from mining in excess of 10,000 curies per AFR. The February 6, 1978, Battelle report includes a paper entitled *Radon and its Daughter Products in Uranium Mining Ventilation Exhaust Air*, by Walter Enderlin. This paper states:
>
> To date there are not sufficient data available to correlate mine production rates with the concentration of radon daughters in the ventilation exhaust plume.
>
> The paper goes on to list the following nine factors which influence radon emissions: (1) grade of ore, (2) [fluctuations] in atmospheric pressure, (3) rate of advance and size [of] broken ore, (4) quantity of ground water entering mine, (5) quantity of exposed rock surface which varies with type of mining method and age of mine, (6) resident time of ventilation air, (7) amount of ore handling underground, (8) type of ventilation system, and (9) [porosity] and permeability of mine rock. The Perkins record at pages 2541 and 2542 also [suggests] the difficulty in correlating radon releases to ore production. This is a specific instance of a deficiency in Perkins which results from using models [rather] than data from actual mines. The evidence referred to above indicates radon emission from mining can only be determined on a mine by mine basis. The environmental assessment for

for curies released per ton of ore mined and then using that value to predict radon releases from other mines could lead to erroneous results. They did not argue that the staff's *Perkins* calculations were incorrect or that the underlying data were insufficient. Rather, they challenged the staff's methodology, maintaining that because radon releases vary from mine to mine, emissions must be measured on a mine-by-mine basis and then attributed to each reactor after determining where its fuel will be obtained.

We rejected that argument in ALAB-562, finding that the staff's averaging technique was both realistic and reasonable and that, in contrast, the intervenors' suggested approach would be unworkable. 10 NRC 437, 447 (1979). Contrary to intervenors' assertion, the staff's methodology had not assumed a correlation between curies released and ore produced. Rather, the staff used averages to derive an estimate of curies released per AFR that would be representative of all underground uranium mines. Accordingly, we granted the applicants' motion for summary disposition of intervenors' Deficiency 1.

By virtue of that disposition, the matter of radon releases from active underground mines was no longer before us at the time of the hearing. Consequently, in our order of March 7, 1980 (unpublished), we asked the parties to address whether the new information explored at the hearing materially affected our summary disposition of that deficiency.

The intervenors advocated different approaches in their response to our request. One group would have us reverse our summary disposition of Deficiency 1 and "find that no correlation between radon releases and AFRs has been demonstrated."[39] Because the issue was explored at the hearing in the context of the staff's testimony on inactive underground mines, this intervenor saw no need for further hearings on the matter. It counseled against our simply accepting the staff's new figure of 8,000 curies per AFR, however, arguing that if the correlation itself is suspect, the new number should be no more reliable than the old. The other intervenors argued that we should await publication of the final Battelle report before addressing either our summary disposition of Deficiency 1 or the magnitude of radon releases from underground mining.[40] They stated that they did not foresee any immediate need for hearings on those issues, but would reevaluate that need following publication of the final report. Finally, they

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Sterling and Tyrone cannot be completed until inquiry is made into the actual mines which will produce their uranium.

Response of Ecology Action, fn. 22, *supra*, at 7-8.

[39]Ecology Action's Response to Board Request on Deficiency No. 1, p. 2 (filed June 29, 1980).

[40]Proposed Findings of Fact, Conclusions of Law, and Response to Appeal Board Questions on Battelle Report, Submitted by the TMI-2/Peach Bottom Intervenors, p. 9 (filed June 20, 1980); Intervenors' Response to Details of NRC Staff's Reply Brief, p. 2 (filed July 23, 1980).

asserted that we should place no reliance on any affidavits filed by the staff and applicants without affording intervenors an opportunity for cross-examination.

The staff took the position that the new information does not in any way invalidate the arguments advanced in support of our summary disposition of Deficiency 1. In the staff's view, the fundamental assertion of that deficiency — that the methodology used to determine radon emissions from underground mining was flawed — had not been materially affected and our rejection of it, valid at the time, remained proper.

The applicants agreed, emphasizing that nothing in the revised report affected our conclusions about the staff's methodology. Indeed, the report used the same averaging technique of adding together the radon released from all mines sampled and dividing by the total annual ore production for those mines. Both applicants and the staff urged us to adopt the new estimate as better and more reliable than that available to the Licensing Board in *Perkins*.

We see no reason to await the final Battelle report, as some intervenors have suggested. In the first place, it is not clear whether or when the final report might be available. Tr. 434. Moreover, the new figure was calculated from a sample representing a large proportion (63 percent) of the total underground mine production of uranium in the United States. Mr. Wilde testified that we now have a "mature" population of mines with an equal number of new mines opening and old mines closing. Tr. 412-13. As a mine is worked-out, its average cumulative production increases, which means that if there were more younger than older mines, radon emissions could be expected to increase in the future. Tr. 412. But if, on the average, new mines are opened to replace the older mines which have reached their maximum potential for radon releases, total emissions from the mining industry would change only in the event of a major increase or decrease in production. For that reason, Mr. Wilde considered it unlikely that any final version of the Battelle study would contain results significantly higher than those appearing in the current report. Tr. 411-12.

We remain convinced that summary disposition was appropriate. The staff's methodology is reasonable and, indeed, appears to be the only practical way to estimate radon emissions from uranium mining. The staff's expression of a radon release rate per AFR simply establishes an industry average that can then be attributed to individual reactors according to the number of AFRs they require. It was not intended to represent a fixed correlation between radon releases from any particular mine and the amount of uranium ore obtained from that mine, nor was it used to predict such releases based on the amount of ore produced. Tr. 396-97.

Finally, there is no need for further hearings on this matter because it has already received extensive treatment. The new values presented in the staff's testimony were the subject of thorough cross-examination at the hearing. In these circumstances, we perceive no unfairness in our adopting 8,000 curies per AFR as the radon release rate for underground mines from which uranium ore is actively being produced.

IV. EMISSIONS ATTRIBUTABLE TO URANIUM MILLING

After uranium ore is mined, it is delivered to a mill where, by a combination of physical and chemical processes, the U_3O_8 is separated from the impurities contained in the ore. During the period of active milling, radon gas is released from the ore while it is being processed. After the bulk of the uranium is extracted, the residue is deposited in tailings piles. Because the residual material still contains the progenitors of radon, tailings piles emit radon both during and after the active milling period. The amount of radon released depends on what steps are taken to stabilize the piles.

Our discussion of radon emissions attributable to uranium milling draws on both the *Perkins* record and testimony presented before us. The intervenors objected to the adequacy of the *Perkins* record on milling in three major respects. First, they questioned the accuracy of the long-term radon release rates from mill tailings piles. Second, they challenged the Licensing Board's conclusion on the adequacy of regulatory authority to control mill tailings disposal. Third, they disputed the Board's finding that mill tailings piles can and will be stabilized, and questioned whether the stability of such piles could be assured over hundreds of thousands of years. We discuss these issues in the context of the various sources of radon released during and immediately after the active milling period, as well as the long term emissions from protected and unprotected tailings piles after milling operations have ceased.

A. Radon Releases During Active Milling Operations

In *Perkins,* staff witness Paul J. Magno[41] calculated radon releases from a model uranium mill for each of the distinct stages of the active milling process and for the tailings "dry-out" period immediately after active milling.[42] He found radon releases of 30 curies per AFR during the processing of uranium ore and 750 curies per AFR from mill tailings piles during the 26-year active life of the model uranium mill. He also calculated that 350 curies of radon per AFR would be released from mill tailings piles during the 5-year drying-out period following mill shutdown. The Licensing Board essentially adopted the staff's conclusions.[43]

First, Mr. Magno calculated a release of some 30 curies of radon per AFR resulting from the crushing, grinding, and chemical processing of uranium ore. To arrive at this figure, he assumed that all of the radon in secular equilibrium with the uranium in the ore is available for release to the atmosphere. He further assumed an ore grade of 0.1 percent U_3O_8 and a 90 percent mill recovery fraction.[44] He stated that the emanation coefficient for mill tailings was 0.2 based on experimental data. In order to account for the possible effects of various mill processing steps on the emanating power of the ore particles, however, he used a value of 0.4 for uranium ore during active milling. Magno, pp. 2-3, fol. *Perkins* Tr. 2369. Mr. Magno's analysis was not contradicted. We adopt 30 curies per AFR as the radon release rate for the processing of uranium ore.

Second, to determine radon releases from mill tailings piles during the period of active milling, Mr. Magno employed a model tailings pile representative of two commonly used milling processes. Near the end of the operating life of the model mill, about three-fourths of the mill tailings pile consists of ponds or wet beaches (deposits of solids with significant moisture content); the remainder is composed of essentially dry material. For the wet tailings areas, Mr. Magno estimated a radon release rate of 1.04

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[41]When he prepared his prefiled testimony for *Perkins,* Mr. Magno was a health physicist with the Division of Fuel Cycle and Material Safety in the NRC's Office of Nuclear Material Safety and Safeguards. At the time of the *Perkins* hearing, however, he was employed as an environmental scientist in the Environmental Protection Agency's Office of Radiation Programs. *Perkins* Tr. 2361. He holds a B.S. degree in chemistry from Boston College and has had over 24 years of experience in the nuclear field. Before joining the NRC in 1975, Mr. Magno worked in various capacities for the U.S. Public Health Service, EPA, Northeastern Radiological Health Laboratory, Massachusetts Insititute of Technology, Brookhaven National Laboratory, and the former AEC.

[42]When discharged as mill effluent, the tailings contain a considerable amount of water. The drying-out period (which, for the model mill, is five years) is the amount of time required for evaporation of essentially all of the moisture in the pile.

[43]LBP-78-25, 8 NRC 87, 92-93 (1978).

[44]The recovery fraction represents the percentage of uranium in the ore converted to U_3O_8 in the milling process. The remainder of the uranium is considered unrecoverable and, for the most part, ends up in the tailings pile.

curies per year for each acre of pile surface area. For the dry pile material, he calculated a radon release rate of 38.2 curies per year per acre.[45] By assuming a 38 foot pile depth, Mr. Magno calculated that for each AFR, 2.72 x 10^5 metric tons of tailings would be produced and that each AFR would contribute 2.9 acres to the tailings pile.[46] After apportioning the 2.9 acres per AFR into wet and dry tailings pile components, he concluded that 29 curies would be released annually for each AFR produced at the mill, for a total release of 750 curies per AFR during the 26-year life of the mill. *Id.* at 3-5 and Table 1. That value was uncontested, and intervenors alleged no deficiencies with regard to it. We accept it with the reservation noted, and adopt it as corrected.

Third, Mr. Magno calculated the amount of radon released during the tailings dry-out period once milling operations cease. He assumed that the tailings would achieve an essentially dry condition in five years and that the drying-out process would occur linearly (*i.e.*, at the same rate throughout the five year period). He then calculated a radon release of 350 curies per AFR attributable to that five year period. *Id.* at 6. This value, like that for the active milling phase, was neither controverted nor the subject of any of intervenors' deficiencies. We adopt it with the same previously noted correction.[47]

Thus, Mr. Magno's analysis yields a total radon release rate of 1130 curies per AFR for the initial processing of the uranium ore, the 26 years of operation of the model uranium mill, and the 5 years during which its mill tailings are drying out. No party challenged Mr. Magno's conclusions. As noted previously, however, Dr. Goldman identified an error of about 25 percent due to Mr. Magno's use of an incorrect density value for the model mill tailings pile. Consequently, his figure of 750 curies per AFR from mill tailings during active milling must be increased to 940 curies per AFR, and

[45]Wet tailings piles release less radon then dry tailings because of the presence of water rather than air in the void spaces of the tailings medium. Diffusion coefficients for materials in water are typically much smaller than those for materials in air. See, *e.g.*, 4 McGraw Hill Encyclopedia of Science and Technology 154-55 (1971).

[46]To arrive at that figure, Mr. Magno necessarily assumed a value for the density of the tailings, which he neglected to mention in his *Perkins* testimony. As applicants' witness Dr. Goldman pointed out in his testimony before us, however, Mr. Magno apparently used a density value of 1.92 grams per cubic centimeter for the tailings pile material to calculate that each AFR would contribute 2.9 acres to the tailings pile. Dr. Goldman testified that the value Mr. Magno used is appropriate only for wet tailings, whereas the value for dry tailings is 1.6 grams per cubic centimeter. Goldman, p. 8, fn. 4, fol. Tr. 441. Using the density value of 1.6 grams per cubic centimeter, as Dr. Goldman suggested, the tailings area should be 3.6 acres per AFR. This means that wherever Mr. Magno used the smaller area, the correct value would be obtained by increasing his result by 25 percent. For ease of reference to *Perkins*, we shall continue to use Mr. Magno's calculated values and apply the appropriate correction at the conclusion of this section. See pp. 515-516, *infra*.

[47]See fn. 46, *supra*.

his figure of 350 curies per AFR from mill tailings during the 5-year drying out period must be increased to 430 curies per AFR. With those corrections made, we adopt 1400 curies per AFR, comprised of the following, as the release rate for active milling: (1) 30 curies per AFR from the processing of uranium ore;[48] (2) 940 curies per AFR from mill tailings piles during the 26-year active milling period; and (3) 430 curies per AFR from mill tailings piles during the 5-year drying-out period after active milling has ceased.

B. Long-Term Radon Releases from Unprotected Mill Tailings Piles

In *Perkins*, Mr. Magno also testified for the staff about long-term radon emissions from unstabilized mill tailings piles. If completely uncovered, the staff's model tailings pile would release 110 curies of radon per year per AFR. Based on that testimony, the Licensing Board found that tailings piles which are uncovered or protected by only a few feet of soil would continue to emit radon at a rate of about 100 curies per year per AFR for tens of thousands of years. 8 NRC at 93. Intervenors questioned the accuracy of this value and alleged in Deficiency 10 that measured emissions at actual mills are larger than the staff computed.[49] At the hearing, they presented Dr. Robert O. Pohl[50] who testified in support of that assertion.

Using a model abandoned tailings pile from an Environmental Protection Agency study of existing abandoned tailings piles,[51] Dr. Pohl calculated that the atmospheric release of radon from unprotected mill tailings would be 330 curies per AFR per year.[52] Pohl, p. 1, fol. Tr. 24. To demonstrate further the staff's underestimation of the release rate for

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[48]This value need not be corrected because it does not concern the model tailings pile to which the error pertains.

[49]Deficiency 10 states:

> 10. The affidavit of P.J. Magno calculates radon emissions of 1,130 curies per AFR through the [active] milling period. Following [stabilization], Magno's affidavit indicates an emission rate of between 1 and 100 curies per year. [Intervenors] are prepared to submit evidence, based on government documents, that measured emissions at actual mills are greater than computed in Mr. Magno's affidavit.

Response of Ecology Action, fn. 22, *supra*, at 12. Although intervenors referred to Mr. Magno's estimate for the active milling period, the evidence they presented dealt with radon releases from dry, unprotected tailings piles.

[50]Dr. Pohl is a professor of physics at Cornell University. He obtained his Ph.D. in physics from the University of Erlangen in 1957. His experience includes research on energy-related applications of solid state physics and work in the area of nuclear waste disposal.

[51]J. Swift *et. al.*, "Potential Radiological Impact of Airborne Releases and Direct Gamma Radiation to Individuals Living Near Inactive Uranium Mill Tailings Piles," EPA/520/1-76-001, U.S. Environmental Protection Agency, Office of Radiation Programs (January 1976).

[52]Although the detailed analysis underlying Dr. Pohl's estimate was not repeated in his testimony before us, it appears in his affidavit, supporting Ecology Action's response to the applicants' motion for summary disposition (filed July 16, 1979).

unprotected piles, he also referred to the model pile presented in the Staff's Draft Generic Environmental Impact Statement on Uranium Milling (GEIS).[53] Using that model, he calculated a release rate of 200 curies per year per AFR, nearly twice the staff's *Perkins* estimate. The Draft GEIS assumed a model tailings pile generated at the model mill with a height of 8 meters and an area of 80 hectares (or 8 x 10^5 square meters). Dr. Pohl computed that the model pile would contain 10^7 metric tons of tailings representing ore originally containing 1.5 x 10^4 metric tons of U_3O_8. Using the staff's value of 245 metric tons of U_3O_8 per AFR, Dr. Pohl calculated that the pile represented tailings from the production of 55 AFRs. These tailings would release 1.1 x 10^4 curies per year, or 200 curies per year per AFR. Pohl, p. 1, fol. Tr. 24.

Staff witness Hubert J. Miller[54] and applicants' witness Dr. Goldman fully endorsed the staff's *Perkins* estimate of 110 curies per year per AFR, with minor adjustments. They explained that radon releases per AFR are inversely proportional to average tailings depth; *i.e.*, lesser pile depths produce greater radon emissions per AFR. Thus, most of the discrepancy between Dr. Pohl's figure (330 curies per AFR per year) and Mr. Magno's estimate (110 curies per AFR per year) resulted from the different values they used for the depth of the tailings pile. Miller, p. 27, fol. Tr. 150; Goldman, pp. 10-13, fol. Tr. 441. Mr. Miller pointed out that in the EPA report Dr. Pohl used, the average depth of tailings at the inactive sites surveyed was estimated to be 4.8 meters. Miller, p. 28, fol. Tr. 150. Mr. Magno had derived the staff's *Perkins* estimate from the same model tailings pile used for active milling, with the additional assumption that it was completely dry. Using Mr. Magno's depth figure of 38 feet (or 11.6 meters), Mr. Miller recalculated Dr. Pohl's estimate and obtained 137 curies per AFR per year. Mr. Miller then attributed the remaining difference primarily to the specific flux coefficients used.[55]

[53] NUREG-0511, fn. 9, *supra*.

[54] Mr. Miller, who joined the NRC staff in 1976, is a section leader for uranium mill licensing in the waste management division of the NRC's Office of Nuclear Material Safety and Safeguards. He holds a B.S. in civil engineering from the University of Notre Dame and an M.S. in environmental engineering from the University of North Carolina. His experience includes 5 years with the U.S. Navy, during which he was assigned to the AEC's Division of Naval Reactors. He also served as project manager for the staff's Generic Environmental Impact Statement on Uranium Milling.

[55] The specific flux coefficient describes the relationship between the surface emanation of radon and the concentration of radium in a body of material. It is closely related to the diffusion coefficient of the medium derived from diffusion theory (see p. 18, *supra*). Mr. Miller considered Mr. Magno's specific flux coefficient to be "conservatively realistic," and pointed out that it was consistent with the results of NRC-supported research as well as measurements of radon emissions discussed in the Draft GEIS on uranium milling. Miller, pp. 29-30, fol. Tr. 150.

Both Dr. Goldman and Mr. Miller supported the staff's choice of 11.6 meters for the average pile depth. According to Mr. Miller, that value was confirmed by the results of a staff survey of 18 active mills which yielded an average value in the range of 12 to 13 meters. Dr. Goldman's more limited survey suggested an average pile depth of 13 meters. Goldman, p. 12, fol. Tr. 441.

Mr. Miller also provided an estimate of radon emissions for a source not previously considered in the staff's *Perkins* testimony. That source is the release of radium in particulate form as a result of wind blown "dusting" of dry tailings. Radium dispersed from the tailings pile falls on the ground nearby and remains a separate source of radon regardless of whether the tailings piles are later stabilized. Using Mr. Magno's composite model, Mr. Miller estimated a radon release rate of 1.4 curies per AFR per year from this source. He considered the *Perkins* release rates (for both covered and uncovered piles) to be sufficiently conservative to encompass such a small value. Consequently, he did not increase Mr. Magno's estimate by that amount and concluded that radon releases from uncovered piles would be 110 curies per AFR per year.

Dr. Goldman assumed a median pile depth of 12.5 meters and calculated a conservative estimate of radon releases from dry uncovered tailings of 135-160 curies per AFR per year. Based on his analysis, he expected the actual release rate to be about 75-80 curies per AFR per year, about half his conservative estimate. Goldman, p. 12, fol. Tr. 441.

Dr. Goldman also recalculated Mr. Magno's *Perkins* estimate of radon emissions of 110 curies per year per AFR from uncovered tailings piles. Using the correct value for the density of dry tailings (see fn. 46, *supra*), Dr. Goldman arrived at a corrected radon release value of 140 curies per year per AFR. *Id.* at 10. This figure agrees well with his conservatively calculated estimate, and uses an average tailings depth representative of tailings piles at operating mills. We therefore adopt Dr. Goldman's corrected version of Mr. Magno's estimate that the radon release rate from uncovered tailings piles will be 140 curies per year per AFR.

Both abandoned tailings piles and those which will be created in the future must be covered in accordance with recent legislative and regulatory requirements described in the next section. Once those actions have been taken, our finding of a release rate for uncovered tailings piles should not be applicable to any existing piles. As we shall see (pp. 530-31, *infra*), it is thus more appropriately viewed as a conservative upper limit for the long-term effects of erosion and other uncertainties.

C. Long-Term Radon Releases from Protected Mill Tailings

Several recent developments have affected the Licensing Board's findings in *Perkins* concerning radon releases from covered mill tailings piles. They are: (1) passage of the Uranium Mill Tailings and Radiation Control Act; (2) completion of the staff's Generic Environmental Impact Statement on Uranium Milling; and (3) issuance of final NRC regulations governing uranium mill licensing and tailings management. Staff witnesses had testified in *Perkins* that uranium mill operators would be required to stabilize mill tailings piles so that radon emissions would not exceed twice the natural background release rate and the need for continuing, active maintenance would be eliminated. They further stated that the staff was obtaining commitments for equivalent stabilization requirements from those "Agreement States"[56] that regulated uranium mills and tailings. Based on Mr. Magno's calculations, the Licensing Board found that tailings piles stabilized according to the staff's performance objective would emit less than one curie of radon per AFR per year. LBP-78-25, 8 NRC at 93-94.

When we issued ALAB-562, the Commission had not yet promulgated its final rules for uranium mill operation and tailings disposal. We noted in that decision that intervenors had extensively questioned the adequacy of regulatory control over mill tailings and the degree of assurance for long-term stability of tailings impoundments. Drawing from intervenor's allegations of deficiencies in the *Perkins* record, we stated:

> [T]he claim that the piles will be covered or stabilized, and can be maintained in that fashion, has not been sufficiently well established. In this respect, the de-stabilizing effects of erosion, tails migration, and the sheer volume of the pile remain to be fully considered. Nor has there yet been demonstrated the requisite assurance that regulatory control of mill tailings can be maintained for an appropriate length of time. And the effect of the guidelines under which such control is now exercised is not clear.

10 NRC at 441-42 (footnotes omitted). We accepted those issues for litigation because there were no final rules for tailings disposal.[57] As we shall see, the new statute and rules generally provide the basis for decision on the issues we previously identified.

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[56]Pursuant to Section 274 of the Atomic Energy Act, 42 U.S.C. § 2021, the Commission may enter into agreements providing for state regulation of certain types, quantities and uses of radioactive materials.

[57]See 10 CFR 2.758(a), which provides that, except in special circumstances, Commission rules and regulations shall not be subject to attack in adjudicatory licensing proceedings.

1. Regulatory Authority

The Uranium Mill Tailings and Radiation Control Act of 1978 (UMTRCA[58] established programs for (1) remedial action at inactive, unreclaimed mill tailings sites and (2) stronger regulatory control of currently active and future mill sites. Title I of that Act requires the Department of Energy (DOE) to bring inactive sites into conformance with generally applicable criteria to be established by the Environmental Protection Agency (EPA), with NRC concurrence in the specific remedial action to be taken at each site.[59]

Title II of the Act added mill tailings as licensable byproduct material under the Atomic Energy Act of 1954, thereby removing any uncertainty about the NRC's authority to require tailings stabilization after mill operations have ceased. In addition, Congress ratified the staff's past practice of requiring licensees to post security to insure proper mill decommissioning and tailings reclamation. Where necessary or desirable to protect the public health or safety, the Act requires state or federal ownership of tailings disposal sites.

In order to assure uniformity, the Act also set forth requirements for the licensing and regulation of uranium mills and mill tailings by Agreement States.[60] After November 8, 1981, those states must have a program for tailings management that includes standards for protection of public health, safety and the environment "which are equivalent, to the extent practicable, or more stringent than, standards adopted and enforced by the Commission."[61] In the interim, the Commission must ensure that state regulatory

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[58]Pub. L. No. 95-604, 92 Stat. 3021, 42 U.S.C. §§ 7901 *et seq.* (November 8, 1978), as amended by Section 22 of the Surface Transportation Assistance Act of 1978, Publ. L. No. 96-106, 93 Stat. 799 (November 9, 1979).

[59]We note that EPA has issued immediately effective interim cleanup standards and proposed disposal standards for inactive uranium processing sites, although it has not yet promulgated final standards. See 45 Fed. Reg. 27366 (April 22, 1980) and 46 Fed. Reg. 2556 (January 9, 1981), respectively.

[60]Intervenors advanced the following deficiency concerning the adequacy of regulatory control by Agreement States:

> 16. Staff testimony indicates that in agreement states mill tailings will be adequately isolated and stabilized. However, a notice on page [17879 of vol. 43] #81 of the Federal Register (April 26, 1978) captioned Assessment of Environmental Impact of Uranium Mills in Agreement States, suggests concern on the part of the NRC as to the environmental review procedure used in agreement states and the capability of such states to insure the isolation and stabilization of tailings.

Response of Ecology Action, fn. 22, *supra*, at 14. As explained in the text, the provisions of UMTRCA and a clarifying amendment (fns. 61 and 62, *infra*) have removed any past uncertainty about tailings management by Agreement States.

[61]UMTRCA, fn. 58, *supra*, Section 204(e)(1) (adding a new subsection (o) to Section 274 of the Atomic Energy Act, 42 U.S.C. § 2021(o)).

authority is "exercised in a manner which, to the extent practicable, is consistent with [the Commission's requirements]."[62]

In April 1979, the staff published a draft version of its GEIS on uranium milling to be used in support of proposed regulations on uranium milling and mill tailings disposal.[63] Those proposed regulations were issued for comment in August 1979[64] and were explored at the hearing before us.[65] Thereafter, the staff published its final GEIS on uranium milling[66] and the Commission adopted in final form the regulations supported by that document.[67] The Commission's statement of considerations published with the final rules specifically directs us "to base [our] decision [in the radon release proceedings] on the adjudicatory record of those proceedings." It also permits us to make findings "different from those contained in the GEIS and reflected in the final rule" to the extent they may be called for by that record.[68] We not that the record before us does not dictate any findings that are inconsistent with the new rules.[69]

The regulations set forth criteria for the management and disposal of mill tailings. They establish "technical, financial, ownership, and long term site surveillance requirements relating to the siting, operation, decontamination, decommissioning, and reclamation of mills and tailings or waste systems and sites at which such mills and systems are located."[70] Disposal of tailings or wastes at milling sites "should be such that ongoing active maintenance is not necessary to preserve isolation" (Criterion 12). Site inspections must be conducted at least annually to verify the stability of tailings or waste systems and to determine the need, if any, for monitoring or maintenance. (*Ibid.*) Tailings are to be located to eliminate or reduce disruption and dispersion by natural forces, and primary emphasis is to be given to isolation of tailings or wastes in view of their potential long-term effects (Criterion 1). The "prime option" for tailings disposal is below grade burial in mines or specially excavated pits (Criterion 3); where this is

[62]Sections 22(a) and (b) of Pub. L. No. 96-106, fn. 58, *supra*, amending Section 204(h) of UMTRCA (governing State authority to regulate uranium mills and mill tailings).

[63]See 44 Fed. Reg. 24963 (April 27, 1979).

[64]44 Fed. Reg. 50012 (August 24, 1979).

[65]At the time of the hearing, the regulations were not yet final; hence, they had no legal effect. Accordingly, staff witnesses testified about the staff's past regulatory practices and explained how its "interim performance objectives" had been incorporated into the proposed rules. Because the proposed and final rules are not significantly different, our discussion is in terms of the final rules.

[66]U.S. Nuclear Regulatory Commission, "Final Generic Environmental Impact Statement on Uranium Milling," NUREG-0706 (September 1980).

[67]45 Fed. Reg. 65521 (October 3, 1980). The rules became effective on November 17, 1980.

[68]*Id.* at 65522.

[69]In contrast, our findings concerning radon release rates are, in some respects, different from the factual assumptions used in the GEIS (see *e.g.*, pp. 55-59, *supra*).

[70]45 Fed. Reg. at 65533.

impracticable or not environmentally sound, above grade disposal must provide "reasonably equivalent isolation" of the tailings. Regardless of whether above or below grade disposal is used, additional siting and design criteria must be met. They are: (a) location to decrease the possibility of water erosion and flooding; (b) topographic features providing good wind protection; (c) relatively flat embankment slopes (preferably, a horizontal to vertical ratio of at least 10 to 1, and no steeper than 5 to 1); (d) a full cover of rocks or self-sustaining vegetation to retard wind and water erosion; (e) absence of faults capable of threatening the integrity of the impoundments; and (f) features to enhance the thickness of cover over time (Criterion 4). Tailings must be sufficiently covered to maintain a calculated surface release rate of less than two picocuries per square meter per second (2 $pCi/m^2/s$) above natural background levels (Criterion 6). Soil or other materials used for cover must be chosen "to ensure that surface radon exhalation is not significantly above background because of the cover material itself." For example, near-surface cover materials (that is, those found within the top three meters of cover) may not include mine waste or rock containing elevated levels of radium; rather, "soil used for near surface cover must be essentially the same, as far as radioactivity is concerned, as that of surrounding surface soils" (Criterion 6).

Steps must be taken "to reduce seepage of toxic materials into groundwater to the maximum extent reasonably achievable." Any seepage that does occur must not degrade groundwater supplies. In order to accomplish this objective, consideration must be given to the use of low permeability clay or synthetic liners, leak detection systems, groundwater monitoring, testing programs, and tailings neutralization and dewatering systems (Criterion 5).

During active milling operations, airborne effluent releases must be kept "as low as is reasonably achievable," primarily through the use of monitoring and emissions controls. Uncovered tailings must be wetted or chemically stabilized (Criterion 8). Tailings or waste retention systems must be inspected frequently (Criterion 8A).

Finally, mill operators must establish financial surety arrangements for decontamination and decommissioning of mills and milling sites as well as for reclamation of any tailings or waste disposal areas (Criterion 9). Each mill operator must pay a minimum charge of $250,000 to cover the costs of long-term surveillance (Criterion 10). Ownership of tailings and their disposal sites shall be transferred to the United States or, at the State's option, the State in which the land is located (Criterion 11).

We shall now consider intervenors' questions about the adequacy and feasibility of tailings management in light of the statutory and regulatory scheme just outlined.

2. Stabilization Methods

Intervenors alleged in Deficiency 13 that the *Perkins* record contained "no information concerning what will be necessary to accomplish" the staff's regulatory objective, which was stated in that proceeding to be the stabilization of mill tailings piles to limit radon emissions to no greater than twice the naturally occurring background level.[71] Intervenors asserted that until the staff could demonstrate "precisely what must be done" to achieve that level of radon control, it would not be "possible to conclude that as a practical matter the Commission's objective is attainable."

Mr. Miller explained that "the problems of managing and stabilizing mill tailings piles involve conventional earth-moving and civil engineering operations," and that disposal in accordance with staff licensing requirements was "feasible from cost and engineering points of view." Miller, p. 32, fol. Tr. 150. Similarly, Dr. Goldman testified that "there is no technical difficulty in moving and placing large volumes of soil or similar materials," and that stabilization and reclamation techniques had already been developed in connection with open-pit coal mining operations. Goldman, p. 13, fol. Tr. 441.

In ALAB-562, we noted that the intervenors had expressed concern about the effect of lower ore grades on the size of the tailings pile and, correspondingly, the amount of radon released from the pile.[72] 10 NRC at 442. We also questioned whether the staff's stabilization criteria would be applied to take that effect into account. As we have seen, the statute and regulations governing uranium milling and tailings disposal prescribe in

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[71]Deficiency 13 is as follows:

> 13. Mr. Kerr for the Staff testified [that] the licensing restrictions for mills [impose] a requirement on mill operators that [tailings] be stabilized so the radon emissions are no greater than 2X background. However, the record contains no information concerning what will be necessary to accomplish the desired objective. [Until] evidence is obtained which [indicates] precisely what must be done to reduce tailings emissions to 2X [background], it is not possible to conclude that as a practical matter the Commission's objective is attainable. In addition, Mr. Kerr did not indicate where the background is to be measured. Is the background baseline a national average or an average in the vicinity of the mill?

Response of Ecology Action, fn. 22, *supra*, at 13.

[72]In Deficiency 17, intervenors alleged:

> 17. The uranium industry is already turning to lower and lower grades of ore. This means higher volumes of tailings than assumed [in] Perkins. Although the number of potential curies may remain the same, larger piles will be more expensive and difficult to isolate and stabilize. [Intervenors] are prepared to present testimony on this point.

Id. at 14.

detail how tailings management will be performed. The regulations now in effect provide an adequate response to intervenors' challenge. At the time of the hearing, however, the rules were not yet final and the staff necessarily presented testimony about its past regulatory practices and described examples of recently licensed tailings management programs. Miller, pp. 19-23. fol. Tr. 150. That testimony further supports the conclusion that the Commission's regulatory objectives are attainable.

Mr. Miller testified that differences in the volume of tailings that would result from changes in ore grade would not significantly affect the manageability of tailings disposal. He explained that there is more variability in the size of existing tailings piles than would result with average ore grade estimates of from 0.1 percent to 0.07 percent. The size of the tailings piles is determined less by ore grade than by other variables, such as the size of ore bodies being mined and milled or the number of mines supplying a particular mill. Miller, pp. 38-39, fol. Tr. 151. Thus, the effect of lower ore grades on tailings management would appear to be negligible.[73] Moreover, the new regulations require that sufficient cover be placed over mill tailings to yield a calculated surface exhalation rate of less than 2 picocuries per square meter per second above background radiation. That requirement is independent of any change in ore grade; hence, the use of lower grades of ore would not result in increased radon emissions from covered mill tailings piles. *Id.* at 38.

We conclude that the final regulations have adequately answered intervenors' previous concerns about stabilization methods and the effect of lower grades of ore on tailings management. Even in the absence of the new rules, however, the record is sufficiently complete to find that the regulatory criterion of limiting incremental radon releases to no greater than 1 curie per AFR per year is attainable as a practical matter.[74] We turn now to the degree of assurance for long-term stability of protected mill tailings, with particular attention to the likelihood and effect of tailings erosion, migration, and seepage.

3. Long-Term Stability of Protected Mill Tailings

The new regulations are clearly designed to provide long-term stability of covered tailings. In promulgating the final rules, the Commission explained that they were developed in light of the fact that tailings remain

[73]Similarly, Figure 3 of Dr. Goldman's testimony shows the relatively low degree of dependence of radon release rates on ore grades from 0.07 to 0.20 percent.

[74]For ease of comparison with other release rates discussed in this opinion, the new regulatory requirement of 2 $pCi/m^2/s$ can also be expressed as about 0.26 Ci/acre/yr. Using the staff's value of 3.6 acres per AFR (*i.e.*, 2.9 acres per AFR corrected for the density error Dr. Goldman pointed out, fn. 46, *supra*), this represents a release rate of slightly less than 1 curie per AFR per year.

hazardous for hundreds of thousands of years. It recognized, however, that attempting to provide absolute assurances that tailings will always remain completely isolated is both impractical and inappropriate, considering the high volume and low activity of tailings as well as the potential effects of other naturally occurring and technologically enhanced sources of radon. 45 Fed. Reg. at 65525.

As we have seen, the *Perkins* Board found that tailings stabilized according to the staff's then applicable criteria would emit only 1 curie per year per AFR. With regard to long-term stability, the Board simply declared that tailings piles meeting the staff's standards did not appear to be erodible "in a matter of a few hundred or a few thousand years." 8 NRC at 93-95.

Intervenors challenged the adequacy of that assessment of long-term stability. Although they advanced and litigated their deficiencies before promulgation of the Commission's final rules, their concerns remain the same whether considered in the absence of or in conjunction with the new regulations. In essence, intervenors assert that isolation of mill tailings from the environment must be guaranteed for the full period of toxicity and that there is no assurance that regulatory control can be maintained for the requisite period of time.[75]

Dr. Pohl testified for the intervenors in support of those assertions. Quoting the Environmental Protection Agency's proposed Criteria for Radioactive Wastes, he stated that "[t]he fundamental goal for controlling any type of radioactive waste should be complete isolation over its hazardous lifetime" and that institutional controls should be relied on for no longer than 100 years, after which engineered and natural barriers must be sufficient. Pohl, p. 5, fol. Tr. 24. Dr. Pohl claimed that the staff had

[75]Intervenors made these allegations about long-term stability of mill tailings impoundments:

> 14. In computing the long range emissions from mill tailings, the Staff assumes gradual deterioration of the [vegetative] cover. However, no consideration is given to the effect of spatial diffusion of the tailings piles which is likely to follow upon [erosion of] the cover. As the surface area of the [pile] increases, the radon released also increases. Evidence should be obtained indicating the release rate of piles as their surface area increases.
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> 21. Mill tailings will constitute a massive amount of material. [Intervenors] are prepared to submit testimony that with respect to lesser amounts of radioactive materials the experience of the federal government has been that radioactive materials migrate to a much greater extent than originally anticipated and that there is every reason to believe this problem will be worse with the larger volume represented by mill tailings.

Response of Ecology Action, fn. 22, *supra*, at 13-14, 15-16. Aspects of Deficiency 21 which have to do with water pathways are discussed in the next section (see pp.532-534, *infra*).

underestimated the release rate for protected mill tailings in *Perkins* because it had not demonstrated that "the required degree of protection can be provided for the required length of time." *Id.* at 2. Noting that the half life of one of the progenitors of radon, thorium-230, is 80,000 years, he explained that "no amount of earth or rock cover, or of vegetation, can be expected to withstand the natural erosive forces of the elements" for hundreds of thousands of years. *Ibid.* Because of the geological time scale involved, Dr. Pohl stated that it would be "very difficult" to predict what would happen to the mill tailings piles over long periods of time (Tr. 36) and that the proposed 2 pCi/m²/s standard for radon emissions from stabilized tailings piles would not adequately protect the environment (Tr. 107). For that reason, he asserted that to assess the potential environmental impact of radon emissions on future generations, it must be assumed that tailings piles will be completely dispersed by erosion. Tr. 36. Dr. Pohl offered no basis for predicting the means or probability of that dispersal; rather, he explained that he had not attempted to make assumptions about what would likely happen to mill tailings piles over long periods of time. Tr. 36-37.

Dr. Pohl calculated the release rate for a completely dispersed mill tailings pile. Using an 8 meter high, dry and uneroded 80 hectare mill tailings pile (based on information presented in the staff's draft GEIS on uranium milling), he calculated that 82 percent of the radon available for diffusion would be shielded from release due to the pile height. He then assumed that, as the pile spread through erosion, the 82 percent shielding gradually would be lost, eventually resulting in an increase of radon releases to the atmosphere by a factor of 5.5. Pohl, p. 2, fol Tr. 24. For one AFR's worth of mill tailings spread very thinly over the ground, Dr. Pohl estimated a radon release rate of about 1,000 curies per AFR per year. Tr. 57.

Dr. Pohl also criticized the *Perkins* Board for failing to give adequate consideration to certain significant localized impacts associated with mill tailings piles. He cited as an example the use in the Grand Junction, Colorado area of some 50,000 tons of abandoned tailings as foundation material for occupiable structures. The use of such material contaminated some 800 buildings, 325 of which had been cleaned up as of February 1979. He expected that proper stabilization of tailings piles would prevent problems such as those encountered at Grand Junction, but only for so long as people remained aware of the toxic nature of mill tailings. Pohl, pp. 3-5, fol. Tr. 24. It was his opinion that because tailings had been used for building materials in the past, they would likely be so used again. Tr. 39.

Testifying for the staff, Mr. Miller acknowledged that the very long-term future of mill tailings impoundments is indeterminable. Miller, p. 16, fol. Tr.

150. He stated that it is impossible to predict and quantify their performance over the period of more than 100 thousand years during which the tailings will remain hazardous. *Id.* at 15. Mr. Miller concluded, however, that if the regulatory requirements are followed, erosion can be virtually eliminated and mill tailings disposal sites can remain stable for many thousands of years. *Id.* at 13; Tr. 205.

Mr. Miller explained that the staff's regulatory approach to tailings management was based on a "systematic, comprehensive study"[76] of potential means of long-term failure. The technical siting and design requirements which the staff has used in the past to insure tailings pile integrity are now part of the regulations governing uranium milling and mill tailings disposal. Miller, p. 11-13, fol. Tr. 150. As previously indicated, those regulations prescribe specific siting and design features to minimize or eliminate failures of tailings impoundments. Because the likelihood of failure is influenced by a variety of factors such as topography, climate and seismicity, the staff conducts site-specific evaluations of erosion potential to insure long-term stability. *Id.* at 13.

To minimize the potential for failure due to flooding, the staff evaluates the effects of the "probable maximum flood," which is the largest possible flood that can reasonably be expected to occur at a given site, based primarily on climatologic records. *Id.* at 14; Tr. 231-33. In addition, the regulations specify that tailings may not be located near streams, upstream drainage areas subject to flooding, or topographic lows where rainfall might accumulate.

Mr. Miller also testified that tailings can be protected from erosion due to water sheet flow and wind by means of full surface covers of vegetation or "rip rap" (a cover of cobble and large rocks). He stated that rip rap is not erodible under gale force wind and, where properly installed, can virtually preclude sheet water erosion as well. Its use in tailings management programs is now being required to provide very reliable long-term protection. Although Mr. Miller also mentioned vegetative cover as a possible means of erosion control, he explained that it could not be counted on in semiarid regions and that the staff would most likely require the replacement of vegetation with a more inherently stable rock cover. Miller, pp. 14-15, fol. Tr. 150; Tr. 251-52.

Finally, Mr. Miller explained how the staff considers earthquake protection in its licensing reviews. Tailings may not be located near a potentially active fault that could cause a more severe earthquake than the

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[76]J. Nelson and T. Shepherd, "Evaluation of Long-Term Stability of Uranium Tailings Disposal Alternatives," Civil Engineering Department, Colorado State University, prepared for Argonne National Laboratory (April 1978). At the hearing, the study was frequently referred to as the "Colorado State University report."

tailings impoundment could be expected to withstand. Miller, p. 12, fol. Tr. 150.

Mr. Miller acknowledged that beyond a period of several thousand years, geologic and climatic changes will determine the stability of tailings isolation. He pointed out that there will almost certainly be some failures at some sites, but that they could easily be remedied as long as institutional controls continue to exist. He considered it entirely possible that the tailings cover will increase over time, because impoundments are being designed to promote that result. *Id.* at 15-16. Based on the Colorado State University report on long-term stability (fn. 76, *supra*), which he conceived and helped to prepare, Mr. Miller concluded that properly isolated tailings should remain stable for many thousands of years, as opposed to the staff's *Perkins* testimony that the cover would essentially deteriorate in about 500 years. Tr. 208-209, 215. He was unable to be more specific than that, and explained that it would be nonsensical to make any statements about long-term stability in terms of hundreds of thousands of years because of the impossibility of predicting climatic and geologic changes. Tr. 215-18.

To provide some perspective on the question of erosion, Mr. Miller explained that the staff had estimated in *Perkins* that if all tailings cover were stripped away, radon releases would be 110 curies per AFR per year. Miller, p. 16, fol. Tr. 150. He explained that for a single pile, the worst possible situation would be a failure resulting in dispersion of mill tailings over a wide area. Tr. 274, 279-80. The staff did not incorporate this possibility into its generic assessment of erosion potential, however, because it required postulation of an unreasonable series of events — *i.e.,* that *all* tailings piles would erode and be totally dispersed. In light of the tailings management practices being required of licensees, Mr. Miller considered this assumption unrealistic. In other words, Mr. Miller believed that there was sufficient conservatism in the staff's regulatory approach to account for a reasonable range of conditions and that it was therefore inappropriate to assume long-term radon releases greater than those that would result if all of the tailings piles completely lost their protective covering. Tr. 293-94.

Dr. Goldman testified for the applicants that even assuming[77] erosion of all stabilizing cover, mill tailings piles would disperse very slowly. He noted that internal friction would inherently limit the degree to which the piles would spread out, and that the slopes prescribed in the new regulations would provide ample protection against tailings migration due to slope instability. Goldman, pp. 14, 20, fol. Tr. 441.

Dr. Goldman also attempted to calculate the radon emissions that would result if all the stabilizing cover were eroded from a mill tailings pile. He derived his estimate from an EPA report which presented the results of a survey of radioactive contamination from 20 inactive tailings sites. From the total amount of radium estimated to have been dispersed from the piles, he calculated that 743 curies of radon per year would be released. Dr. Goldman then applied that average dispersal rate to an EPA estimate of the total radium content of all inactive mill tailings piles in the United States. From this he obtained a release rate of 1,130 curies of radon per year due to dispersion. Dr. Goldman calculated that if erosion rates remained constant, the tailings would be completely dispersed in about 2700 years; however, he expected erosion rates to decrease with time as more of the readily erodible material was removed. He concluded that this very slow rate of dispersion would provide ample opportunity for taking remedial action. *Id.* at 17-20.

All three expert witnesses who testified before us agreed that because of the geologic time scale dictated by the 80,000 year half-life of thorium-230, the very long-term stability of mill tailings impoundments cannot be assured.[78] The real question, then, is the effect to be given that long-term uncertainty. Because the integrity of mill tailings piles cannot be guaranteed for the full period of their toxicity, intervenors would have us "immediately suspend the operating licenses of all reactors subject to this proceeding in

[77]Dr. Goldman agreed with Mr. Miller's assessment that properly covered tailings should remain stable for thousands of years. Tr. 468. In this connection, he testified about the survival for thousands of years of earth structures, or mounds, built by Pre-Columbian Indians in North America. Several thousand mounds exist, largely in the eastern United States, and the oldest dates back about 3000 years. Goldman, p. 15, fol. Tr. 441; Tr. 446, 482. Dr. Goldman considered them roughly analogous to unstabilized mill tailings piles because no sprecial precautions (such as the placement of rock cover) were taken to prevent erosion. Tr. 445, 469, 483-84. He stated that although it is unknown what fraction of the mounds originally built still exist, those which remain have survived for many centuries without substantial erosion. Some mounds were constructed in the shape of animals and the details of their shapes (such as horns, ears, or toes) are still visible upon aerial inspection. According to Dr. Goldman, those details would have disappeared if substantial erosion had taken place. Tr. 482-83. In view of the survival of those ancient mounds, Dr. Goldman concluded that contemporary engineers should be able to provide at least equal protection against erosion and that mill tailings piles could be maintained in a stabilized condition with a minimum amount of administrative control. Goldman, pp. 16-17, fol. Tr. 441.

[78]Pohl, p. 2, fol. Tr. 24; Miller, pp. 15-16, fol. Tr. 150; Goldman, Tr. 498.

view of the prodigious, long-term releases of radon which are attributable to the nuclear fuel cycle."[79] In contrast, the staff would have us find that its "stringent siting and design requirements for tailings disposal assure long-term stability under natural weathering forces."[80] The applicants propose a similar finding, to the effect that the staff's regulatory approach "will provide reasonable assurance of long-term stability [for] uranium mill tailings piles."[81] Both the applicants and the staff base their conclusions on Mr. Miller's testimony that properly stabilized mill tailings should remain isolated for many thousands of years.

Mr. Miller stated it was not possible to specify or bound that time period with any greater precision. This was because of the uncertainties associated with the effects of climatic and geologic changes that might occur over long periods. Tr. 210, 215, 216a. He did explain, however, that the average denudation rate in the arid regions where tailings will be located is on the order of a foot every four thousand years. Tr. 209-210. Because gentle slopes and rock covering of above-ground tailings piles would provide armoring similar to "desert pavements" which have been stable for some 20,000 years, Mr. Miller testified that tailings stabilized in that fashion might possibly last for the first half-life of thorium-230. Miller, p. 15, fol. Tr. 150; Tr. 252. Although the intervenors would have us reject Mr. Miller's rough time estimate of "many thousands of years" as unreliable in view of his inability to be more precise,[82] we think that Mr. Miller adequately explained why further precision would be possible only on a site-specific basis. Moreover, we recognize that tailings will continue to be a potential source of radon for hundreds of thousands of years. On such a time scale, reliable predictions are impossible even for specific sites.

Long term stability of tailings disposal sites is guaranteed for at least as long as monitoring and institutional controls continue. Beyond that, tailings disposed of in accordance with the new regulations will be isolated by means of physical barriers which should not require active maintenance. The record demonstrates that mill tailings impoundments will, in all likelihood, remain stable for thousands of years. The staff's regulatory approach focuses on the specific problems likely to occur at each site and incorporates particular design features to account for them. Siting and design requirements are being used to maximize protection against floods, earthquakes, and erosion by wind or water and to promote deposition of sediment to enhance thickness of cover over time. Such measures make it

[79]Proposed findings of *Peach Bottom-Three Mile Island* intervenors at pp. 1, 10 (filed June 18 1980); see also proposed findings of Ecology Action at p. 1 (filed June 20, 1980).
[80]Staff's proposed findings and conclusions at pp. 35-36 (filed July 3, 1980).
[81]Applicant's proposed findings at pp. 8-9, 11-12 (filed April 28, 1980).
[82]Ecology Action's proposed findings, fn. 79, *supra*, at p. 1.

reasonably certain that tailings stabilized according to the new regulations will remain protected for very long periods of time.

We find it unnecessary to adopt Dr. Pohl's scenario of complete dispersal of mill tailings piles into a thin layer on top of the ground, yielding radon releases of about 1000 curies per year per AFR far into the future. Dr. Pohl himself had no basis for predicting such dispersal; he sought only to postulate a "worst case." Mr. Miller considered such complete dispersal not reasonably likely in view of the regulatory criteria designed to provide long-term stability. Dr. Goldman also testified that complete dispersal of mill tailings piles was quite unlikely. He added that even if it should occur, the tailings material would not remain on top of the ground emitting radon for thousands of years, but would either be carried by surface water to the bottom of the oceans or would eventually be covered by soil deposits. Dr. Pohl's assertion that we must assume complete dispersal of mill tailings piles is merely speculative and lacks factual support. We concur with Mr. Miller's opinion that to account for geologic and climatic uncertainty over the very long term, we need not go beyond the staff's *Perkins* assumption that radon releases would not exceed those associated with erosion of the stabilizing cover of all tailings piles.

We find that isolation of mill tailings pursuant to the Commission's rules is reasonably assured for the foreseeable future. Further, we find that it is neither useful nor feasible to quantify the effects of various uncertainties about the long term stability of mill tailings impoundments. Hence, we reject the intervenors' assertion that tailings isolation must be guaranteed for the full period of toxicity.

Finally, we note that there is a potential gap between the regulations as written and as implemented by individual uranium mill licensees. There is no evidence to suggest that there will not be adequate quality assurance for the design and implementation of mill tailings disposal (see generally Tr. 308-317). But the new rules do not provide for quality assurance programs beyond the requirements for inspection and testing. Needless to say, proper attention to this aspect of tailings management is essential to insure that the Commission's regulatory objectives are, in fact, achieved.

V. OTHER POSSIBLE RADON RELEASE MECHANISMS

There remain two questions set forth in ALAB-562 concerning radon release mechanisms which are of possible relevance to our evaluation of radon emissions. They are the transport of radium by water pathways and the recovery of uranium from the residues of phosphate fertilizer production. We discuss them in turn.

A. Water Pathways

In ALAB-562 we noted our general agreement with intervenors' Deficiencies 7 and 18 to the effect that neither the parties nor the *Perkins* Board had assessed the potential for human exposure to radon through water pathways.[83] In particular, we recognized that groundwater might possibly enter abandoned mines or mill tailings piles and transport radium to inhabited areas where it could then be ingested or, upon decay into radon, inhaled.

As we have seen (pp. 521-22, *supra*), the Commission's new mill tailings regulations — which became effective subsequent to the evidentiary hearing before us — are designed to prevent surface erosion and seepage of mill tailings into the groundwater at disposal sites. Accordingly, these rules provide an adequate response to intervenors' contention that humans may be exposed to radium, and hence radon, transported by water from mill tailings.[84] But the new rules do not deal with uranium mining, an activity

[83]Intervenors' Deficiencies 7 and 18 were as follows:

> 7. Perkins considers only the atmospheric pathways for radon emissions from mining. However, it is possible for there to be releases to streams or the ground water. Improperly sealed or unsealed mine test holes could fill with rain or ground water. [An] EPA report, *Water Quality Impact of Uranium Mining and Milling Activities in the Grants Mineral Belt New Mexico,* EPA 906/9-75-001 Sept. 1975, found radioactive contamination of drinking water in mining facilities and ground water contamination exceeding EPA limits for certain chemicals by 740%. This report demonstrates the existence of hydrologic pathways for radon contamination.
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> 18. The NRC is considering underground burial of mill tailings. Although this method of disposal seems preferable from the point of view of preventing [erosion] by wind and water of above surface piles, buried tailings are more likely to be leached by groundwater. In fact, one could imagine [that] a below grade [quantity] of mill tailings might represent a [preferred] location for collecting groundwater. Hence, people drilling for water wells may be attracted [to] the burial [sites], and thus be exposed to large radiation exposures through radium 226. This exposure pathway ought to receive careful attention before a decision is made to dispose of mill tailings in this way.

Response of Ecology Action, fn. 22, *supra*, at 11, 15.

[84]Even in the absence of the Commission's mill tailings regulations, however, we would conclude on the record before us that the possibility of surface or groundwater transport of radon and its progenitors from mill tailings has been adequately assessed. As mentioned above (p. 527-528, *supra*), Mr. Miller testified for the staff that tailings impoundments are being designed and located to minimize the likelihood of their disruption due to flooding or water erosion. Specifically, tailings will not be located near perennial or seasonal streams of any appreciable size, or in areas subject to flooding or concentration of surface runoff. Siting and design evaluations are based on the probable maximum flood, and rip-rap (or, in some cases, vegetation) must be used to control water sheet erosion and gullying. Miller, pp. 14-15, fol. Tr. 151.

Testifying for the intervenors, Dr. Pohl mentioned as a means of surface water transport a tailings dam failure which occurred at Gallup, New Mexico in the summer of 1979. Pohl, p. 4, fol. Tr. 24. Mr. Miller testified that the 1,100 tons of tailings which were spilled as a result of which the NRC has no authority to regulate. Thus, our inquiry must still focus on aspects of intervenors' deficiencies concerning water pathways from mines.

Dr. Goldman provided the only quantitative analysis of surface water transport from mines. Goldman, pp. 33-35, fol. Tr. 441. He testified that short term, high intensity rainfall could erode piles of waste rock or sub-ore stored on the surface near uranium mines. *Id.* at 34. To reach an upper estimate of the significance of such erosion, he assumed that all of the sub-ore piled at the model open pit mine presented in the Battelle report[85] was

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the failure Dr. Pohl described were deposited near the dam embankment and had since been retrieved, although the tailings solutions released were unrecoverable. Tr. 254-55. He added that, in recognition of the need for better dams during active milling, the staff is now incorporating provisions of its regulatory guides for tailings dam construction into its licensing and regulatory program. Tr. 258.

Dr. Goldman testified for the applicants that in areas where uranium mining and milling take place, annual evaporation exceeds annual precipitation and most surface streams are normally dry except during periods of rainfall. He stated that if, over a short span of time, precipitation exceeded evaporation, excess moisture in uncovered tailings piles would temporarily seep into the ground, but with no significant movement of radium into the groundwater. Goldman, p. 33, fol. Tr. 441.

Dr. Pohl asserted that below grade burial of mill tailings would enhance the potential for groundwater pollution. Mr. Miller testified, however, that clay or synthetic liners are placed on the bottom and sides of tailings impoundments to protect against seepage of tailings solutions before they dry out. Tr. 302. Thereafter, liners are not needed. Tr. 248-49, 302-303, 325. Further, the likelihood of a rise in the groundwater at a given location is taken into consideration in the staff's siting and design requirements for licensing a particular project. Tr. 325-27. The staff does not try to predict long-term changes or reversals of climate; instead, it requires the building of impoundments which, under reasonably stable climatic and geologic conditions, it believes will be maintenance-free for some thousands of years. Tr. 206-207. Assuming that some mill tailings will come into contact with groundwater, the distance that dissolved thorium and radium might travel is a complex, site-specific question. It depends on such factors as the characteristics of the aquifer, the composition of the soil, and the nature of the milling processes used. Miller, pp. 40-41, fol. Tr. 150; Goldman, p. 30, fol. Tr. 441; Tr. 516-19 (Goldman); Tr. 69 (Pohl). Dr. Pohl expressed concern about the rate of migration of radionuclides, citing the movement of radium and thorium from a thorium waste pile in Chicago. Pohl, p. 6, fol. Tr. 24. But Dr. Goldman testified that the materials. processes, and disposal methods used there bore no relationship to those used in commercial uranium fuel cycle activities. Tr. 507. He further stated that if tailings were placed below groundwater, any increase in the concentration of radium present would be confined to a relatively short distance from the tailings area. Tr. 513-16. Similarly, Mr. Miller testified that some seepage of tailings would occur, but the nuclides involved "tend to sorb or ion-exchange and not migrate to an appreciable extent." Miller, p. 40, fol. Tr. 150. He added that because the factors controlling the rate of movement and possibility for release are variable, it is "not very illuminating to postulate generic scenarios." *Ibid.*

In short, given the steps which must now be taken to protect tailings from erosion and to isolate them from groundwater, radon releases from mill tailings via water pathways should not significantly contribute to total radon emissions associated with uranium milling.

[85]NUREG/CR-0628, fn. 23, *supra.*

completely dispersed. He then calculated that the resulting increase in radon emissions attributable to dispersal of the sub-ore pile would be 130 curies per year per AFR. *Ibid.*

Dr. Goldman considered such complete erosion of sub-ore piles to be highly unlikely. *Id.* at 34. Moreover, he testified that unstabilized sub-ore or waste rock piles would erode in a site-specific manner and that preliminary investigations by EPA had indicated that such material would be transported relatively short distances. *Id.* at 33. Thus, Dr. Goldman's upper bound estimates do not appear to be representative of radon releases due to surface water transport from mines. They do suggest, however, that compared to radon emissions attributable to uranium mining, surface water releases from mines are not significant.

Concerning groundwater contamination from mines, Mr. Miller testified for the staff that most ore bodies are located in aquifers and that the situation following mining is comparable to that which exists before any uranium is recovered. Miller, p. 41, fol. Tr. 150. Dr. Goldman testified that the indirect release of radon following transport in groundwater has not been modeled generically. This, he explained, is because of (1) the highly localized, site-specific nature of groundwater movement patterns; (2) the local relationships between precipitation, evaporation, and runoff; and (3) the variable physical and chemical effects of specific compositions of local soils and rocks on the chemical precipitation of uranium, thorium, and radium. Goldman, pp. 29-30, fol. Tr. 441.

Dr. Goldman nevertheless attempted to estimate the effect of groundwater transport from underground mines. He examined mine drainage water samples for their radon and radium content and compared them with the ratio of radon to radium concentrations found in representative groundwater samples. He then calculated an upper bound of less than 4 curies per year for radon releases from groundwater in an abandoned underground mine. He did not consider that figure to be a significant addition to the uranium mining source term for radon. *Id.* at 31-32.

Based on the record before us, it appears that groundwater contamination from mines would not contribute significantly to radon releases associated with the uranium fuel cycle.

B. Phosphate Residues

In ALAB-562 we asked for quantification of radon releases attributable to the recovery of uranium from residues of phosphate fertilizer production so that they could be compared "with the amount of radon released from the direct mining and milling of an equivalent amount of uranium." 10 NRC at 443. We based our request on intervenors' Deficiency 26, which asserted that uranium was being recovered commercially from the slag byproduct of phosphate fertilizer production and that if radon releases were attributable to that process, they should be accounted for.[86] The staff's *Perkins* estimates of radon emissions were based only on the mining and milling of uranium ores, and the Licensing Board did not mention the possibility of radon emissions associated with the byproduct recovery of uranium from phosphate fertilizers.

On this point, the staff presented Homer Lowenberg,[87] who testified that uranium is currently being extracted from the intermediate phosphoric acid liquor stage during fertilizer production rather than from the slag or residue of the manufacturing process. Lowenberg, p. 3, fol. Tr. 126. He also explained that this process is a byproduct operation, and the phosphate mining and fertilizer manufacturing would continue whether or not uranium was recovered. In Mr. Lowenberg's opinion, any radon emitted during the mining or processing of phosphate ores, as well as the storage or use of either the fertilizer or the gypsum wastes produced, would be properly attributed to the fertilizer production process rather than the uranium fuel cycle.[88] *Id.* at 5. The extraction of uranium from the intermediate phosphoric acid mixture neither increases nor decreases the amount of radon emitted during the fertilizer production process. *Id.* at 3-4. Thus, any uranium produced from that source would have no radon releases beyond those already attributable to the production of phosphate fertilizer. *Id.* at 4.

Testifying for the applicants, Dr. Goldman reiterated that there is no radon released from the recovery of uranium "beyond that attendant upon the phosphate production itself." Goldman, p. 35, fol. Tr. 441. He furnished

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[86]Deficiency 26 states:

> 26. Morton Goldman, at page 2342 of the transcript, indicates some uranium is being recovered commercially from the slag which is a byproduct of the production of phosphate fertilizer. [Information] should be obtained whether radon is released from the recovery of uranium by this process. If this process results in radon emissions, such emissions should be quantified.

Response of Ecology Action, fn. 22, *supra*, at 17.

[87]Mr. Lowenberg has been with the NRC since 1975 and is now Asssistant Director for Operations and Technology for the Division of Fuel Cycle and Material Safety in the Office of Nuclear Material Safety and Safeguards. He holds a B.S. degree in mechanical and chemical engineering from the Stevens Institute of Technology. His experience includes 24 years with private industry in nuclear-related areas and 3 years with the former AEC. He recently served on the United States steering committee for the International Nuclear Fuel Cycle Evaluation (INFCE).

[88]Mr. Lowenberg pointed out that in the processing of phosphate ores to produce fertilizer, most of the radium remains with the gypsum waste materials produced, although about ten percent of the total radium content ends up in the fertilizer where it is a potential source of radon.

an estimate of the radon releases that would occur during the time required for the solvent extraction of uranium, even though such releases would result regardless of whether uranium was being extracted. He calculated that value to be less than one curie per AFR, which is insignificant compared to the amount of radon releases from uranium mining and milling.

We find it unnecessary to quantify the radon releases attributable to the recovery of uranium as a byproduct of fertilizer production. Because all such releases occur during the production of phosphates whether or not any uranium is extracted, we believe they are properly ascribed only to that activity. It would be inappropriate to treat any radon releases from the processing of phosphate ore as part of the uranium fuel cycle.

VI. SUMMARY AND CONCLUSIONS

The radon release values which we have adopted are summarized here and applied in a manner that will enable them to be factored into the cost-benefit analysis for the Peach Bottom, Hope Creek, and Three Mile Island reactors. After quantifying the levels of radon emissions to be allocated to each reactor, we then turn to the question of their environmental significance.

A. Application of Radon Release Findings

In accounting for the radon emissions to be attributed to individual reactors, we have used the same units which the expert witnesses for all parties presented in their testimony before us. Specifically, emissions during active mining and milling are expressed as a finite release in terms of curies per AFR (Ci/AFR). Long term, continuing releases after the mines and mills have shut down are expressed as a yearly release rate for each AFR produced during active mining or milling; that is, in terms of curies per AFR per year (Ci/AFR/yr.).

Table 1 shows the values we adopted for radon emissions attributable to mining. It presents separate figures for radon releases from underground and open pit mines, as well as a composite mine representing uranium production from the mining industry as a whole (*i.e.,* 60 percent from underground and 40 percent from open pit mines). Release rates for sealed underground and reclaimed open pit mines are included to illustrate the degree of reduction in radon emissions that would occur if the mines were properly sealed and reclaimed.

Table 2 summarizes the values which we adopted for radon emissions attributable to milling and is based on the staff's model mill. For convenience of organization, the active milling period is defined to include the five-year period after milling operations have ceased and the tailings are drying out. Long term, continuing release rates are given for both covered and uncovered tailings piles. As we explained earlier, our finding for covered tailings piles is applicable for the foreseeable future, whereas the release rate for uncovered tailings piles represents a conservative upper limit to account for any uncertainty about long term stability of covered tailings. See p. 518 *supra*.

Table 3 shows how our adopted values can be combined to obtain representative radon release rates for the uranium fuel cycle using different assumptions. In each instance, total emissions consist of: (1) a finite release of 6600 curies per AFR from active mining and milling; and (2) a continuing, long term release rate after the mines and mills have been shut down. For those long term radon emissions, three cases are presented. All three are based on the composite mine and the model mill. Case 1 assumes that the mines are sealed and reclaimed and the tailings are covered, resulting in a release rate of 21 curies per AFR per year. Case 2, which summarizes the values to be used in the cost-benefit analyses for the reactors involved in this proceeding, indicates a release rate of 91 curies per year per AFR. It is comprised of unsealed and unreclaimed mines, and tailings stabilized in accordance with Commission regulations. Case 3 includes unsealed and unreclaimed mines together with uncovered tailings, yielding a release rate of 230 curies per AFR per year.

Those release rates can easily be applied to the model 1000 MWe reactor upon which they are based. By definition, the model reactor burns one AFR per year of operation[89] for a total of 30 AFRs needed during its 30 year lifetime. Accordingly, we simply multiply 30 AFRs by the finite and continuing release rates given in Table 3 to establish the quantity of radon emissions attributable to operation of a typical nuclear plant.

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[89]See fn. 18, *supra*, and accompanying text.

This yields radon releases of approximately (1) 198,000 curies during the active mining and milling of the uranium needed to fuel the plant for 30 years;[90] and (2) 2730 curies per year as the long term, continuing release from unsealed mines and protected mill tailings (Case 2). The maximum long term release rate would be 6900 curies per year (Case 3). These values are shown in Table 4.

Because the amount of fuel required depends on the reactor's operating characteristics, we cannot know for certain how many AFRs a particular plant will use during its lifetime. Moreover, the parties have not suggested a total fuel requirement for each of the three facilities under consideration here. We can, however, obtain a rough but conservative (*i.e.*, high) estimate of the number of AFRs a given reactor might use by assuming that, like the model, it will operate at 80 percent capacity for 30 years. Its annual and total fuel requirements would then be proportionately higher or lower than that of the model, depending on its electrical rating. For example, Units 2 and 3 of the Peach Bottom facility are each rated at 1065 MWe for a combined rating of 2130 MWe. This represents an estimated fuel consumption rate of 2.13 times that of the model 1000 MWe reactor. Thus, the two Peach Bottom reactors would require 2.13 AFRs per year or approximately 64 AFRs over their 30-year operating life.

To determine the representative portion of total fuel cycle releases that must be allocated to the Peach Bottom reactors, we multiply the finite and continuing radon release rates by 64 AFRs. This yields radon emissions of approximately (1) 422,400 curies released during the active mining and milling of the uranium needed to fuel the plants for 30 years; and (2) 5820 curies per year as the continuing, long term release attributable to unsealed and unreclaimed mines and covered mill tailings (Case 2). The maximum long term release rate would be 14,720 curies per year (Case 3). These values are summarized in Table 4, together with the results of similar calculations for the Three Mile Island and Hope Creek reactors.

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[90]This finite release attributable to the mining and milling of 30 AFRs is not a "one-time" release; it is emitted over a period of years rather than all at once. Radon emissions from the mining and milling of one AFR can be attributed to the operation of the model reactor on an annual basis. Thus, for purposes of comparison, the finite release of 198,000 curies corresponds to an average release of 6600 curies per year during the 30 year lifetime of the plant.

TABLE 1*

RADON RELEASES FROM MINES

A. UNDERGROUND MINES	
Release Rate During Active Mining	8000 Ci/AFR
Release Rate Following Mine Shut Down	
- Sealed	10 Ci/AFR/yr
- Unsealed	90 Ci/AFR/yr
B. OPEN-PIT MINES	
Release Rate During Active Mining	1000 Ci/AFR
Release Rate Following Mine Shut Down	
- Reclaimed	40 Ci/AFR/yr
- Unreclaimed	100 Ci/AFR/yr
C. COMPOSITE MINE	
Release Rate During Active Mining	5200 Ci/AFR
Release Rate Following Mine Shut Down	
- Sealed and Reclaimed	20 Ci/AFR/yr
- Unsealed and Unreclaimed	90 Ci/AFR/yr

* All release rate values in this and subsequent tables are rounded to the nearest unit of ten. The only exception is the figure of 1 Ci/AFR/yr given for covered mill tailings piles in Tables 2 and 3. Although that number is sufficiently small to be disregarded, we have included it for the sake of completness.

TABLE 2

RADON RELEASES FROM MILLS

A. RELEASES DURING ACTIVE MILLING

Ore Processing	30 Ci/AFR
Tailings Piles During Active Milling	940 Ci/AFR
Tailings Piles During Five-Year Dry-Out	430 Ci/AFR
Total	1400 Ci/AFR

B. LONG TERM RELEASES FROM TAILINGS PILES

Dry, Uncovered Piles	140 Ci/AFR/yr
Covered Piles	1 Ci/AFR/yr

TABLE 3

RADON RELEASE RATES ATTRIBUTABLE TO MODEL URANIUM FUEL CYCLE ACTIVITIES

A. RELEASE RATE DURING ACTIVE MINING AND MILLING (Ci/AFR)

Composite Mine	5200
Model Mill	1400
Total	6600

B. LONG TERM RELEASE RATES AFTER MINING AND MILLING HAVE CEASED (Ci/AFR/yr)

Case 1: Sealed and Reclaimed Mines, Covered Tailings	
Composite Mine	20
Model Mill	1
Total	21
Case 2: Unsealed and Unreclaimed Mines, Covered Tailings	
Composite Mine	90
Model Mine	1
Total	91
Case 3: Unsealed and Unreclaimed Mines, Uncovered Tailings	
Composite Mine	90
Model Mill	140
Total	230

TABLE 4

PORTION OF RADON RELEASES FROM THE URANIUM FUEL CYCLE ATTRIBUTABLE TO OPERATION OF SPECIFIC NUCLEAR FACILITIES

Model Light Water Reactor	
Rating:	1000 MWe
Total Fuel Requirement:	30 AFRs
Release During Active Mining and Milling	198,000 Ci
Continuing Releases, Case 2	2,730 Ci/yr
Continuing Releases, Case 3	6,900 Ci/yr
Peach Bottom, Units 2 and 3	
Rating:	1065 MWe per reactor
Total Fuel Requirement:	64 AFRs
Release During Active Mining and Milling	422,400 Ci
Continuing Releases, Case 2	5,820 Ci/yr
Continuing Releases, Case 3	14,720 Ci/yr
Three Mile Island, Unit 2	
Rating:	906 MWe
Total Fuel Requirement:	27 AFRs
Release During Active Mining and Milling	178,200 Ci
Continuing Releases, Case 2	2,460 Ci/yr
Continuing Releases, Case 3	6,210 Ci/yr
Hope Creek, Units 1 and 2	
Rating:	1067 MWe per reactor
Total Fuel Requirement:	64 AFRs
Release During Active Mining and Milling	422,400 Ci
Continuing Releases, Case 2	5,820 Ci/yr
Continuing Releases, Case 3	14,720 Ci/yr

B. Prematurity of Health Effects Issue

As we indicated in Part I (p. 493 *supra*), now that the radon emissions attributable to the mining and milling of uranium fuel have been quantified, we must confront the acceptability of the *Perkins* Board's *de minimis* approach for assessing the environmental significance of possible health effects from those releases.[91] We also have before us the staff's motion — supported by applicants — for leave to file proposed findings on the health effects question. As explained below, however, the health effects question is not yet ripe for decision. The staff's motion is therefore denied.

The intervenors were not parties to the *Perkins* proceeding and cannot be bound involuntarily by the record adduced in that case. In response to our prior orders on the radon question, intervenors objected to portions of the *Perkins* record on health effects and sought to supplement the record on this subject.[92] Accordingly, fundamental fairness dictates that intervenors now be provided an opportunity to challenge the factual underpinnings of the Licensing Board's *de minimis* rationale. Only then may we properly face the applicability of the *de minimis* approach to the health effects question in the proceedings before us.

In our prior decisions we attempted to fashion feasible and fair procedures for resolving the generic radon issue without holding separate, repetitive trials in a large number of licensing proceedings.[93] The procedures we established involved novel and somewhat complicated steps, which necessarily represented a compromise among the competing views of

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[91]The Licensing Board in *Perkins* articulated its *de minimis* finding as follows:

> Based on the record available to this Board, we find that the best mechanism available to characterize the significance of the radon releases associated with the mining and milling of the nuclear fuel for the Perkins facility is to compare such releases with those associated with natural background. The increase in background associated with Perkins is so small compared with background and so small in comparison with the fluctuations in background, as to be completely undetectable. Under such a circumstance, the impact cannot be significant.

8 NRC at 100.

[92]See Response of Ecology Action of Oswego to ALAB-480, pp. 2-5 (July 26, 1978) (challenging, *inter alia, Perkins* record estimates of radon health effects and failure to consider the greater health impacts on populations living close to uranium mills); Response of *Three Mile Island-Peach Bottom* intervenors to ALAB-480, pp. 3, 7 (July 27, 1978) (asserting need for supplementation of *Perkins* record on the health effects of low level, low-dose rate radiation). See also Response of *Three Mile Island-Peach Bottom* intervenors to ALAB-509, pp. 9, 20 (February 19, 1979) (in effect, challenging the *Perkins* Board's finding on natural background radiation from radon).

[93]See ALAB-480, 7 NRC 796 (1978); ALAB-509, 8 NRC 679 (1978); ALAB-512, 8 NRC 690 (1978); ALAB-540, 9 NRC 428 (1979); ALAB-546, 9 NRC 636 (1979); ALAB-562, 10 NRC 437 (1979); ALAB-566, 10 NRC 527 (1979).

the various parties. Insofar as providing a sound basis for determining the magnitude of radon emissions attributable to the mining and milling of uranium, these procedures proved adequate. At the same time, however, it appears that our prior decisions were somewhat ambiguous with respect to how we would factor the consequences of such radon releases into the cost-benefit analyses of the individual license proceedings before us. No useful purpose would be served by an extended study of those decisions to ascertain the basis of the parties' differing interpretations of them. Suffice it to note that the primary focus of our earlier decisions was upon the magnitude of radon emissions from the mining and milling of uranium.[94] We must now turn to the question of the health effects of those emissions. Toward that end, intervenors must be given an opportunity to challenge

[94]The parties' differing interpretations of our prior decisions apparently stem from the oft-repeated language contained in ALAB-509 concerning the Licensing Board's *de minimis* theory of health effects and our direction that the parties brief the legal sufficiency of that approach as employed by the lower Board. See 8 NRC at 684. Nothing we said there was intended to deny the intervenors an opportunity to challenge those parts of the *Perkins* record dealing with health effects to which they objected in response to ALAB-480. Indeed, at the time we issued ALAB-509; intervenors' objections to the *Perkins* record on health effects were still outstanding — a fact which we recognized in ALAB-509 when we stated that most of intervenors' objections filed in response to ALAB-480 "went to the adequacy of that record on the question of health effects." 8 NRC at 683. The intended focus of ALAB-509 and our subsequent decisions was on the magnitude of radon releases form the mining and milling of uranium rather than the health effects question. For example, in ALAB-562, we stated in granting summary disposition of intervenors' Deficiency 22 that "at this stage we are still trying to ascertain the magnitude of the releases of radon involved in the relevant aspects of the fuel cycle; only after that is done will health effects come into play." 10 NRC at 444-45. In that same opinion, we repeated: "[W]e remind the parties that health effects will not be taken up at this hearing." 10 NRC at 448 fn. 36. Finally, upon the motion of one of the applicants at the hearing on radon releases, we struck certain testimony of intervenors' witness dealing with health effects on the ground that it was beyond the scope of that proceeding. Tr. 8-10, 13-24.

We cannot fully endorse the view of our dissenting colleagues that the issues of natural background radon releases and radon concentrations are now entirely foreclosed. While intervenors did not advance deficiencies on those subjects in response to ALAB-509, they objected to our taking up the *de minimis* theory of health effects before the issue of radon releases had been settled and they clearly expressed the view that ALAB-509 required them to assume, *arguendo*, the natural and fuel cycle related radon issues found in *Perkins*. Response of *Peach Bottom-Three Mile Island* intervenors to ALAB-509, fn. 92, *supra*, at 9; Response of Ecology Action of Oswego to ALAB-509, fn. 22, *supra*, at 18. Not unreasonably, they read our deferral of the health effects question in ALAB-509 as calling for a deferral of consideration of some of their objections to the *Perkins* record as well; namely, the *Perkins* quantification of health effects and localized impacts in the vicinity of the mines and mills, among others. Response of Ecology Action of Oswego to ALAB-509, fn. 22, *supra*, at 4. Although localized radon concentrations may be estimated without proceeding to evaluate their health effects, the subjects are closely related as a practical matter. And, in decisions subsequent to ALAB-509, we failed to indicate that intervenors had misread our directive.

certain facts in the *Perkins* record on health effects to which they previously objected.[95] As soon as practicable, we will issue a memorandum detailing the procedures to be followed in that regard.[96]

It is so ORDERED.

FOR THE APPEAL BOARDS

C. Jean Bishop
Secretary to the
Appeal Boards

The opinion of Dr. Buck and Dr. Johnson, dissenting with respect to Part VI B (pp. 543-545, *supra*), follows.*

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[95] The Commission directed us to reopen the record in pending cases "to receive new evidence on radon releases and on health effects resulting from radon releases." 43 Fed. Reg. at 15616. With all deference to our dissenting colleagues, we believe it is necessary first to establish a proper record on the health effects, if any, from radon releases associated with the mining and milling of uranium fuel before discounting those impacts as remote and speculative. It may ultimately be that any health effects from radon releases are so small as to be environmentally insignificant. But to reach that conclusion before intervenors have had an opportunity to present and prove their case on that subject prejudges the issue. Such a result is contrary to law and inconsistent with our prior decisions in this proceeding. See ALAB-480, 7 NRC at 805.

[96] We do not suggest that a hearing on the health effects question is inevitable. It may be that the issue can be finally resolved upon motions for summary disposition.

*Drs. Buck and Johnson constitute a majority in two of the three contested cases which are the subject of this decision. As we pointed out in ALAB-509, however, the radon matter is a generic one and any significant developments in this proceeding will have to be taken into account in our review of the uncontested cases still pending before us. 8 NRC at 683, fns. 8 and 9. In four of the five uncontested cases for which there is at present a three-member Appeal Board, the conclusion of Messrs. Rosenthal, Salzman, and Moore in Part VI B of this opinion constitutes the majority view. Accordingly, it will govern the larger number of radon cases and is thus being treated as the majority opinion.

Opinion of Dr. Buck and Dr. Johnson dissenting with respect to Part VI B:

In the preceding pages it has been determined from the record of the *Perkins* proceeding and our own evidentiary hearing that a reasonable upper limit on the radon emissions to the atmosphere attributable to a typical 1000 MWe nuclear power plant's fuel cycle at any time during or after the plant's useful life would be about 7000 Ci per year.[1] Further, for the case in which the mine is properly sealed and the mill tailings piles remain adequately covered, the long-term (after plant life) yearly emissions would be about a tenth of this upper limit value.[2] As the *Perkins* Board points out, the only reasonable way to evaluate this environmental release is in comparison to natural radon releases. While we find the magnitude of the maximum radon emissions attributable to nuclear plant operation to be somewhat higher than the values found by the *Perkins* Licensing Board, we nevertheless find no reason to disagree with its conclusion that this amount is negligibly small compared to natural emissions.

Although our colleagues have elected not to consider the matter of natural radon background, the *Perkins* record discloses that atmospheric radon emissions in the United States due to natural sources is some 100,000,000 Ci per year — more than 10,000 times greater than the upper limit value for a typical nuclear power plant.[3] This value was not contested by any party.[4]

The *Perkins* Licensing Board went further. It found that radon emissions from the soil gave rise to average concentrations of radon in the air in the United States of about 0.1/pCi/1 (pico curie per liter).[5] We note that the radon from the fuel cycle operations (which leaks into the atmosphere by the same mechanisms as does natural radon) would add a proportionately small increment to this concentration (*i.e.,* less than 0.00001 pCi/1).[6]

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[1]Table 4, p. 542, *supra.* If one makes the reasonable assumption that, on the average, uranium mining and milling is carried on apace with the utilization of uranium fuel over the 30-year life of a nuclear power plant — the average yearly release due to active mining and milling operations is 6600 Ci/yr.

[2]Table 3, p. 540, *supra.* Compare Case 1 with Case 3 in Part B of that Table.

[3]LBP-78-25, 8 NRC 87, 94 (1978).

[4]We cannot, as does the majority, read the response to ALAB-509 of the *Three Mile Island-Peach Bottom* intervenors (at pp. 9 and 20) as even an "in effect" challenge to the *Perkins* Board findings on natural radon releases (fn. 92, *supra*).

[5]LBP-78-25, 8 NRC 87, 95 (1978).

[6]Our discussion relates solely to the widespread or nationwide average concentration of radon. To be sure, local concentrations, such as those in the immediate vicinity of uranium fuel cycle facilities, would be higher than the large area averages, just as radon concentrations near natural uranium ore bodies are higher than the average concentration across the country. In their footnote 94 our colleagues appear to ignore the testimony on the range of local concentrations of natural radon that was presented in the *Perkins* record. See affidavits by staff

The *Perkins* Board then used an estimated radon radiation dose to provide a further basis for comparison. Natural outdoor radon concentrations (*i.e.*, 0.1 pCi/l) give rise to a bronchial epithelium dose of about 50 mrem/yr.[7] However, as a result of radon emitted from building materials, indoor radon concentrations are much higher than outdoor concentrations, giving rise to bronchial epithelium doses ranging from 210 to 23,000 mrem/yr, with an estimated average value of 1650 mrem/yr.[8]

Thus, indoor concentrations of radon average about 30 times larger than those out of doors, and fluctuate over a range of from 4 to 400 times that value.[9] The incremental addition to the outdoor radon concentration due to a single typical nuclear power plant, on the other hand, is less than one part in 10,000 of the outdoor concentration.

In circumstances such as this, in which the addition to a natural environmental substance (*i.e.*, radon) caused by human activities is extremely small compared with the existing natural concentration (it is small even compared to fluctuations in that concentration), we believe that any assignment of environmental impact to the incremental addition could only be characterized as remote and speculative.[10] We conclude that this impact may properly be ignored in the assessment of the overall environmental impact of a nuclear power plant.[11]

We are additionally of the opinion that all parties were properly put on notice that our disposition of this matter might well be the same as the Licensing Board's, assuming we were to find that the magnitude of radon releases due to the fuel cycle was substantially the same as that Board found. In particular, we specifically informed the parties that we might

witnesses Gotchy (at pp. 11-15) and Magno (at pp. 7-9) which appear following *Perkins* Tr. 2369. These affidavits were supplemented by affidavits of applicants' witnesses Hamilton (at pp. 1-3), Goldman (at pp. 8-12) and Lewis (at pp. 1-3) appearing following *Perkins* Tr. 2266. None of the data or conclusions of these affidavits was challenged by any intervenor even though we gave them the opportunity to do so.

[7] 8 NRC at 95. The bronchial epithelium dose is a direct result of radon inhalation. The radioactive daughters of radon can be deposited on the surfaces of the bronchial (respiratory) system. Since much of the radiation energy deposited by these isotopes is in the form of short range alpha particles, the epithelium (surface) region of the respiratory system is a particularly sensitive measure of exposure to concentrations of radon and its daughters.

[8] *Id.* at 96.

[9] Values quoted in LBP-78-25 for radon radiation exposure include some averaging of the time a person stays outdoors and indoors. For full-time indoor occupancy, the exposures, and hence the comparative indoor concentrations, would be even larger.

[10] We note in this connection that monocellular life began and developed eventually into human life in an atmosphere containing more natural radon than now surrounds us.

[11] We do not deny that it is possible to multiply the small increment of radiation dose due to fuel cycle radon by a factor to achieve a "health effects" consequence or impact. The point is that this same factor must also multiply the much larger natural radon dose values, and the relative levels of significance remain the same. Further litigation to define the factor, even if this could be achieved, will not change the outcome.

forego any further discussion of radon health effects if we were to make such a finding on radon emissions.

In pertinent part, our order, ALAB-509, stated:

> In this connection, two areas seem to call for attention now. First, we need to clarify the extent to which particular parties are dissatisfied with *Perkins* insofar as it deals with *rates of radon release or levels of radon concentration from either natural sources or nuclear fuel cycle activities* (as distinguished from the health effects of any resulting exposure). Second, if *Perkins* is accurate on emission rates and concentration levels, it seems appropriate to examine at the threshold the Licensing Board's *de minimus* theory, *i.e.*, its conclusion that the nationwide health effects attributable to radon released in fueling nuclear power plants must be deemed to be insignificant because those emissions are extremely low in relation not only to natural radon background but also to fluctuations which occur in the background.

8 NRC at 682 (footnote omitted, emphasis added).

We then called upon the parties to set forth which portions of the *Perkins* record data on radon release rate and concentrations they felt were deficient. Regarding the subject of health effects, we went on the say:

> As indicated by the preceding section, we are not now in a position to determine whether *Perkins* accurately reflects the levels of exposure to radon. If, however, at some future time we were to find the *Perkins* emission and concentration figures correct (or reasonably close to being so), we would have to come to grips with the Licensing Board's *de minimus* theory.
>
> The *Perkins* board took the approach that, whatever else might be said about the health effects of radon,
>
> > Based on the record available to this Board, we find that the best mechanism available to characterize the significance of the radon releases associated with the mining and milling of the nuclear fuel for the Perkins facility is to compare such releases with those associated with natural background. The increase in background associated with Perkins is so small compared with background and so small in comparison with the fluctuations in background, as to be completely undetectable. Under such circumstance, the impact cannot be significant.
>
> If we were to subscribe to that view, there would appear to be no reason to consider the question of health effects further. Consequently,

> we believe it appropriate to consider this aspect of the Board's decision at the outset.

8 NRC at 684 (footnote omitted).

We then called for briefs on the question whether the Licensing Board's approach was acceptable. Our reading of the briefs that were subsequently submitted does not cause us to change our opinion that the Licensing Board's dismissal of the radon impact as insignificant was correct.

In light of the foregoing explicit indication of how we might ultimately resolve the radon matter, we cannot accept the idea that the parties should be given yet an additional opportunity to contest the basis upon which this resolution could be made. Nor do we understand how the colleagues with whom we joined in issuing ALAB-509 can now blithely proclaim that "it is necessary first to establish a proper record on health effects" (fn. 95, *supra*).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:
Richard S. Salzman, Chairman
Dr. John H. Buck
Thomas S. Moore

In the Matter of **Docket Nos. 50-387**
50-388

PENNSYLVANIA POWER & LIGHT COMPANY and ALLEGHENY ELECTRIC COOPERATIVE, INC.
(Susquehanna Steam Electric Station, Units 1 and 2) **May 15, 1981**

The Appeal Board denies a motion seeking directed certification of a part of the Licensing Board's ruling in LBP-81-8 denying summary disposition of a portion of a contention raised in the proceeding.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Denial of partial summary disposition is an interlocutory order from which an appeal is proscribed by the Rules of Practice. 10 CFR § 2.730(f). *Louisiana Power and Light Company* (Waterford Steam Electric Generating Station, Unit 3), ALAB-220, 8 AEC 93, 94 (1974); *Pacific Gas and Electric Company* (Stanislaus Nuclear Project, Unit No. 1), ALAB-400, 5 NRC 1175, 1177 (1977).

RULES OF PRACTICE: DISCRETIONARY INTERLOCUTORY REVIEW

The exercise of jurisdiction under 10 CFR § 2.718(i) (certification authority) is reserved for those important licensing board rulings which, absent immediate appellate review, threaten a party with serious irreparable harm or pervasively affect the basic structure of the proceeding. *Public Service Company of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977).

APPEARANCES

Ms. Jessica H. Laverty for the Nuclear Regulatory Commission staff.

Messrs. Jay E. Silberg and **Matias F. Travieso-Diaz**, Washington, D.C., for the Pennsylvania Power & Light Company *et al., applicants.*

MEMORANDUM AND ORDER

1. On March 16, 1981, the Licensing Board granted in part applicants' motions, each supported by the NRC staff, for summary disposition on Contentions 2 and 16. The Board denied the motion with respect to that portion of Contention 2 which deals with chlorine discharges from the Susquehanna nuclear facility. LBP-81-8, 13 NRC 335 (1981). We now have before us the staff's April 14 motion, supported by the applicants, seeking directed certification of a part of the ruling denying summary disposition. Invocation of our discretionary authority under 10 CFR § 2.718(i) to review issues before the end of the hearing is necessary because denial of partial summary disposition is an interlocutory order from which an appeal is proscribed by the Rules of Practice. 10 CFR § 2.730(f); *Louisiana Power and Light Company* (Waterford Steam Electric Generating Station, Unit 3), ALAB-220, 8 AEC 93, 94 (1974); *Pacific Gas and Electric Company* (Stanislaus Nuclear Project, Unit No. 1), ALAB-400, 5 NRC 1175, 1177 (1977).

2. The exercise of jurisdiction under Section 2.718(i) is reserved for those important licensing board rulings which, absent immediate appellate review, threaten a party with serious irreparable harm or pervasively affect the basic structure of the proceeding. *Public Service Company of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977). See *Puget Sound Power and Light Company* (Skagit

Nuclear Power Project, Units 1 and 2), ALAB-572, 10 NRC 693, 695 n. 5 (1979), and cases there cited. In this instance that standard is not met.

The staff argues that certification is appropriate because the Licensing Board's ruling unwarrantedly expanded the scope of the chlorine issue and, as a consequence, both the staff and applicants must be prepared to litigate two issues which they claim are not properly part of the intervenor's original contentions.[1] According to the staff, trial of these issues will force them into preparing wasteful, expensive and time-consuming predictions about remote and speculative matters; their papers assert that this unnecessary expense and delay amounts to immediate and irreparable harm that cannot be alleviated by subsequent appeal. In addition, the staff argues that in expanding the issues the Board below made extra-record "findings of fact" based solely on the unsworn and unsubstantiated assertions of intervenors. In the staff's view, the Board's action is in the teeth of the Commission's regulations and the Administrative Procedure Act and therefore pervasively affects the basic structure of the proceeding.

3. In the context of the denial of a motion for partial summary disposition, the staff's arguments do little more than state the apparent. Obviously the Licensing Board's ruling will result in the trial of issues with the concomitant investment of time and money such litigation entails. Equally obvious is the fact that once the hearing is held the time and money expended in the trial of an issue cannot be recouped by any appellate action. But the same is true any time summary disposition of an issue is denied and a litigant must go to hearing. The fact that the ruling below may have erroneously expanded the issues to be tried or done so on the basis of unsworn allegations does little to distinguish this case from any other where it is alleged that summary disposition was erroneously withheld.[2] Indeed, had the Licensing Board raised the challenged issue on its own motion, we think it clear that directed certification would not be appropriate. We therefore conclude that the staff's asserted injuries fall short of the standard for discretionary interlocutory review.

In reality, adoption of the staff's rationale would alter the standard for discretionary interlocutory review; certainly where a denial of summary disposition is involved it would be reduced to a simple determination whether the Licensing Board erred. As we stated in *Houston Lighting and*

[1]As we understand its motion, the staff does not contest the denial of summary disposition with respect to part of issue 2(b) and all of issue (c) identified in Part D (the "order" portion) of the Licensing Board's March 16, 1981 ruling. LBP-81-8, *supra*, 13 NRC at 348. The applicants, on the other hand, appear to desire certification directed to all three issues, even though they only filed a response supporting the staff's motion rather than their own motion.

[2]For the same reason, applicants' argument that the ruling below will deter it from filing further summary disposition motions in this proceeding (or deter parties in other proceedings from filing such motions) is unpersuasive.

Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-635, 13 NRC 309 (1981): "It is scarcely necessary to expound at any length upon why a drastic alteration of existing practice to accomodate that thesis would be intolerable—as well as in derogation of the Commission's explicit policy disfavoring interlocutory review."[3]

The petition is *denied.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the
Appeal Board

[3]The staff and the applicants, of course, remain free to pursue other avenues of possible relief that remain open to them before the Licensing Board. For example, they may ask for reconsideration of the ruling and press the fuller exposition of their positions made to us on the Board below. We have no reason to believe the Board below (or any party) wishes to conduct an unnecessary hearing—if the staff's and applicants' characterization of the situation is correct. Indeed, for that reason, the Licensing Board may wish to reconsider the issue itself on the basis of the staff's motion before us. In addition, if the staff and applicants are convinced (as their papers suggest) that the issues on which summary disposition was denied raise matters so remote and speculative as to merit no consideration under the National Environmental Policy Act, they may simply "stand pat" and seek vindication on appeal in the event the Licensing Board disagrees.

Additional views of Mr. Salzman, concurring:

My colleagues' reasons for declining to grant certification are sound ones. I am concerned, however, that the papers before us reflect misconceptions about the operation of the summary disposition rule and the Licensing Board's prerogative to explore potential safety issues not placed in controversy by the parties.

On the first point, the summary disposition rule (10 CFR § 2.749) is the Commission counterpart of Rule 56 of the Federal Rules of Civil Procedure governing summary judgments; essentially the same standards govern both.[1] Consequently, even though a summary disposition motion may rest on affidavits technically sufficient to justify a decision in movant's favor, the absence of opposing affidavits does not mean that the relief sought must be forthcoming automatically. The courts have explained that:

> Although Rule 56 of the Federal Rules of Civil Procedure states that summary judgment "shall" be rendered when the stated conditions are met, the rule is not mandatory in operation: "a motion for summary judgment is always addressed to the discretion of the court." Satisfying the basic requirements of the rule does not guarantee that the motion will be granted: "Even in cases where the movant has technically discharged his burden, the trial court in the exercise of a sound discretion may decline to grant summary judgment."[2]

In other words, "the exercise of sound ... discretion may dictate that the motion should be *denied*, and the case fully developed."[3]

This principle is possibly more important in Commission than in judicial practice, which brings up the second point. The NRC functions as the arbiter of important safety and environmental questions. That role "does not permit it to act as an umpire blandly calling balls and strikes for adversaries appearing before it .... "[4] For this reason, a licensing board may if need be explore issues not placed in controversy by the parties but which come to its attention during the course of the proceeding (or are suggested

[1]*Cleveland Electric Illuminating Co.* (Perry Plant, Units 1 and 2), ALAB-443, 6 NRC 741, 753 (1977); *Tennessee Valley Authority* (Browns Ferry Plant, Units 1, 2 and 3), LBP-73-29, 6 AEC 682, 688 (1973).

[2]*In re Franklin National Bank Sec. Litigation*, 478 F. Supp. 210, 223 (E.D.N.Y. 1979) (citations omitted).

[3]*McLain* v. *Meier*, 612 F.2d 349, 356 (8th Cir. 1979) (emphasis in original). *Accord*, *Roberts* v. *Browning*, 610 F.2d 528 (8th Cir. 1979); *Browns Ferry*, *supra*, LBP-73-29, 6 AEC at 688.

[4]*Perry*, *supra*, ALAB-443, 6 NRC at 752, quoting *Scenic Hudson Preservation Conference* v. *FPC*, 354 F.2d 608, 620 (2nd Cir. 1965).

informally in unsworn limited appearance statements[5]). As the Commission stressed in *Indian Point:*

> A Licensing Board, typically comprised of two technical experts and a lawyer, is this agency's primary fact-finding tribunal in the hearing process. These expert tribunals are entrusted with critical tasks in the licensing process. ... To tie a Board's hands, when it sees an issue that needs to be explored, would be utterly inconsistent with its stature and responsibility.[6]

In passing on applicants' summary disposition motions, the Board below evinced apprehensions about chlorine discharges from the Susquehanna plants resulting in unacceptable water pollution levels and has called for further exploration of the question. The Board's concerns may turn out to be misplaced or later shown to be insubstantial. But it elevates form over substance to suggest that the Board is precluded from considering a safety issue that it apparently deems significant because of the way it came to light. An inadequate response to a summary disposition motion jeopardizes the respondent's rights to explore an issue, *not the Licensing Board's.*

The Board's misgivings are perhaps inartfully framed as "findings." But these are preliminary and obviously designed to alert the parties to the principal areas of its concern. There are means short of a full trial by which the Board's fears can be alleviated by parties (assuming, of course, that they are correct about the true situation). Applicants suggest one; there are others.[7] If one is selected and employed to educate the Board, it in turn will no doubt rule appropriately. There is no reason to presume that the Licensing Board wishes to go through the formalities or incur the expense of a hearing unnecessarily.

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[5]See 10 CFR Part 2, App. A, § V(b)(4).

[6]*Consolidated Edison Co. of New York* (Indian Point, Unit 3), CLI-74-28, 8 AEC 7, 8 (1974). This decision is now codified in the regulations. 10 CFR § 2.760a.

[7]See, *e.g.*, *Consumers Power Co.* (Midland Plant), ALAB-235, 8 AEC 645, 646 (1974).

Cite as 13 NRC 557 (1981) **LBP-81-12**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
Elizabeth S. Bowers, Chairman
Frederick J. Shon
Dr. Richard F. Cole

In the Matter of **Docket No. 50-312 SP**

SACRAMENTO MUNICIPAL UTILITY DISTRICT (Rancho Seco Nuclear Generating Station) **May 15, 1981**

The Licensing Board issues an initial decision approving the adequacy of required short-term actions and long-term modifications for continued reactor operation which were imposed by the Commission as a result of the accident that occurred at Three Mile Island, Unit 2 on March 28, 1979.

TECHNICAL ISSUES DISCUSSED:

Integrated control system;
Feedwater transients;
Once through steam generator sensitivity;
Anticipatory reactor trips;
Pressurizer and quench tank sizing
Natural circulation;
Void formation;
Small-break loss-of-coolant accidents;
Auxiliary feedwater system reliability;
Safety system challenges;
Operator and management competence;
Instrumentation
Control room configuration;
Hydrogen control;
Venting back into containment;

Controlled filtered venting.

APPEARANCES

Messrs. Thomas A. Baxter and **Matias F. Travieso-Diaz**, Washington, D.C., for the Licensee.

Messrs. Christopher T. Ellison, Sacramento, California and **Lawrence Coe Lanpher**, Washington, D.C., for California Energy Commission.

Messrs. Stephen H. Lewis and **Richard L. Black**, Bethesda, Maryland, for the Nuclear Regulatory Commission Staff.

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NOTE: Transcript Corrections will be issued separately.

INITIAL DECISION
(Permitting Continued Reactor Operation)

I. INTRODUCTION AND BACKGROUND

1. The Rancho Seco Nuclear Generating Station, owned and operated by the Sacramento Municipal Utility District (Licensee or SMUD) was granted a Facility Operating License No. DPR-54 in 1974. This facility which utilizes a Babcock and Wilcox (B&W) pressurized water reactor (PWR) is located in Sacramento County, California. Following the March 28, 1979 accident at the B&W facility at Three Mile Island, Unit 2, the Commission issued an Order on May 7, 1979 (44 Fed. Reg. 27779) pertaining to the operation of Rancho Seco. The Order was based on the premise that B&W PWR's are unusually sensitive to certain off-normal transient conditions originating in the secondary side. B&W reactors have been viewed as placing more reliance than other reactors on the performance characteristics of the auxiliary feedwater system, the integrated control system, and emergency core cooling system to recover from anticipated transients. If this position is correct, there is a greater burden on B&W plant operators in the event of off-normal system behavior during anticipated transients. (In anticipation of the above Order, SMUD by letter of April 27, 1979 agreed to the NRC requirements and shut down Rancho Seco on April 28, 1979.) CEC Ex. 25.

2. The Order provided that the facility would remain in a shutdown condition until five short-term actions were accomplished and to subsequently accomplish, as promptly as practicable, four long-term modifications.

The five short-term actions were:

(a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979.
(b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.
(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.
(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator.

The four long-term modifications are:

(a) The licensee will provide to the NRC Staff a proposed schedule for implementation of identified design modifications which specifically relate to items 1 through 9 of Enclosure 1 to the licensee's letter of April 27, 1979, and would significantly improve safety.
(b) The licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC Staff as soon as practicable. The licensee stated that this analysis is now underway with high priority by B&W.
(c) The reactor trip following loss of main feedwater and/or trip of the turbine to be installed promptly pursuant to this Order will thereafter be upgraded so that the components are safety grade. The licensee will submit this design to the NRC Staff for review.
(d) The licensee will continue operator training and have a minimum of two licensed operators per shift with TMI-2 simulator training at B&W by June 1, 1979. Thereafter, at least one licensed operator with TMI-2 simulator training at B&W will be assigned to the control room. All training of licensed personnel will be completed by June 28, 1979.

3. The Order also provided that, within 20 days of issuance, SMUD, or any person whose interest may be affected by the Order may request a hearing based on it. The Order provided that such a request would not stay the immediate effectiveness of the Order.

4. A request for a hearing was filed by two of the five elected Directors of SMUD, Mr. Gary Hursh and Mr. Richard D. Castro. A request was also filed by Friends of the Earth, Environmental Council of Sacramento, and Original SMUD Rate Payers Association (collectively FOE).

5. On June 21, 1979, the Commission ordered the establishment of an Atomic Safety and Licensing Board to consider the requisite personal interest of petitioners and, if appropriate, to conduct a hearing. The Order also confirmed that the resumed operation of Rancho Seco would not be stayed by the pendency of these proceedings. *Sacramento Municipal Utility District* (Rancho Seco Nuclear Generating Station), CLI-79-7, 9 NRC 680 (1979), *motion to stay denied,* Friends of the Earth, Inc. v. U.S., 600 F. 2nd 753 (9th Circ. 1979).

6. The Order further provided that the subjects to be considered at the hearing should include:

1) Whether the actions required by subparagraphs (a) through (e) of Section IV of the Order are necessary and sufficient to provide reasonable assurance that the facility will respond safely to feedwater transients, pending completion of the long-term modifications set forth in Section II. A contention challenging the correctness of the NRC Staff's conclusion that the actions described in subparagraphs (a) through (e) have been completed satisfactorily will be considered to be within the scope of the hearing. However, the filing of such a contention shall not of itself stay operation of the plant.

2) Whether the licensee should be required to accomplish, as promptly as practicable, the long-term modifications set forth in Section II of the Order.

3) Whether these long-term modifications are sufficient to provide continued reasonable assurance that the facility will respond safely to feedwater transients.

Resumed operation of the Rancho Seco facility on terms consistent with the Order of May 7, 1979 is not stayed by the pendency of these proceedings. Contrary to the contention of the Friends of the Earth in their filing of June 8, 1979, the transcripts of the Commission proceedings of April 25 and 27 reflect no Commission intent that hearings necessarily precede restart of the facility. Nor is such a requirement compelled by law or by the factual circumstances before us. Mere speculation that the hearing might develop facts indicating the need for further enforcement action does not suffice to warrant a prohibition on restart of the facility. In the event that a need for further enforcement action becomes apparent, either in the course of the hearing or at any other time, appropriate action can be taken at that time.

NRC Staff has now determined that the actions set forth in subparagraphs (a) through (e) have been completed satisfactorily, and it shall provide the Commission with an informational briefing as to the basis for its conclusions prior to permitting restart of the facility. That briefing will be open to the public. In receiving this briefing, the Commission will in no manner prejudge the merits of the adjudicatory hearing authorized by this Order. Any adjudicatory determination by the Commission that may arise from that hearing will be based solely on the record developed in that proceeding. (footnote omitted)[1]

7. On June 27, 1979, the NRC Director of Nuclear Reactor Regulation authorized the operation of Rancho Seco on the basis that SMUD has satisfactorily completed the five short-term items. (NRC Staff's Evaluation of Licensee's Compliance with the NRC Order dated May 7, 1979, fol. Tr. 362). The NRC also required SMUD to undertake other equipment, procedure and personnel changes related to the TMI accident primarily contained in Inspection and Enforcement (I&E) Bulletins 79-95A, 79-05B, and 79-05C, and two reports of the "TMI-2 Lessons Learned Task Force" (NUREG-0578 and NUREG-0585). NRC Ex. 4 at 3-1 to 3-8 and App. A.

8. An Atomic Safety and Licensing Board was established by Order of June 22, 1979. On July 3, 1979, the Board issued an Order giving petitioners an opportunity to amend their petitions to more fully explain how their interest might be affected by the Commission's May 7, 1979, Order. Petitioners were also invited to file their contentions in the proceeding. On July 17, 1979, the California Energy Commission (CEC) filed notice of its intent in participating on behalf of the State of California as an interested state under 10 CFR 2.715(c). On August 1, 1979, at the first prehearing conference, the Board admitted FOE and Messrs. Hursh and Castro as Intervenors. The Board also admitted CEC under 2.715(c). These admissions were stated in the August 3, 1979, Board Order subsequent to the prehearing conference. The Board directed the parties to confer to attempt to arrive at possible stipulations as to contentions and also to brief the question of the Board's jurisdiction. The Board also stated that while the burden of proof did not shift from SMUD, the burden of going forward rested on the proponent of a contention.[2]

9. On October 5, 1979, the Board issued an Order ruling on scope and contentions.[3] In considering scope, the Order stated that the proceeding would include all matters and issues which hinge upon response to

[1]On July 11, 1979, in an open public meeting, the Commission amended the June 21 Order to provide that the Board was not precluded from inquiring into SMUD's management competence and control. The Commission furnished the transcript of the meeting to the Board.

[2]CEC originally sought to be relieved of any burden of going forward on its contentions because it was participating as an interested state and, thus, did not wish to take a position with respect to the issues. However, in our Order Ruling on CEC's Motion of October 24, 1979 Relative to Burden and Going Forward, dated December 17, 1979, we ruled that CEC's issues were essentially "contentions" and that CEC had the burden of going forward with those issues that were not previously adopted as "Board Issues." By its submission of testimony, we believe CEC has met its burden of going forward on its issues.

[3]A prehearing scheduled in November had to be cancelled due to the illness of the Chairman. The Board was reconstituted.

feedwater transients. The Board stated the scope would include the propagation of a response throughout the Rancho Seco system, where "system" includes the physical facilities as well as the organization and personnel which operate them. The Order further stated:

> As to "various transient events" as the phrase is used at page four of the Commission's May 7 Order, we believe that, taken in the context of page five of that same Order, the scope of this proceeding can be expanded no further than " ... feedwater and/or trip of the turbine .... " We will, therefore, not allow matters such as loss of off-site power to be raised and considered among the contentions here.

The Board also ruled that the subject of emergency planning was beyond the scope of the proceeding because it was about to become the subject of generic Commission rulemaking.[3a]

10. The Board's rulings on specific issues and contentions of CEC and Intervenors were supplemented in our Order Relative to Proposed New Schedule (for filings), December 4, 1979, and in a Memorandum of Clarification, December 27, 1979. The Board's Order of January 7, 1980 set forth Additional Board Questions.

11. A Second Prehearing Conference was held on February 6, 1980 to consider motions for summary disposition and any other outstanding motions. SMUD moved for summary disposition on all of the contentions of Hursh and Castro and on one issue of CEC. In an opening statement at the conference, Hursh and Castro withdrew from the proceeding in a diatribe alleging the proceeding was a "hoax and a sham". Tr. 72. Both the Staff and CEC encouraged the Board to adopt as Board questions those Hursh and Castro contentions on which summary judgment would not have been granted. Tr. 76, 78. The Board denied summary disposition on 9 of the Hursh and Castro contentions and made them Board questions (rephrased in its Order Subsequent to the Prehearing Conference, dated February 14, 1980).

12. The evidentiary hearing commenced on February 26, 1980 with the first two days, plus an evening session, set aside for limited appearance statements. A representative from FOE informed the Board that FOE by letter of February 19, 1980, was withdrawing from the proceeding on the basis that the proceeding would not serve a useful purpose, and in part, because Intervenors had the burden of "going forward" with their

[3a]In response to a motion by the CEC, the Board referred this ruling to the Atomic Safety and Licensing Appeal Board. LBP-79-33, 10 NRC 821 (1979). The Appeal Board accepted the referral, but CEC subsequently moved to terminate the Appeal Board's consideration of the referred question. The Appeal Board granted the motion. ALAB-576, 11 NRC 16 (1980).

contentions. They also said that issues such as emergency plans and loss of offsite power should have been determined to be within the scope of the proceeding. Tr. 210-223. Both SMUD and Staff presented previously prepared testimony on FOE contentions and the Board will rule on each of those contentions admitted on October 5, 1979. The hearing proceeded with SMUD and NRC Staff appearing as parties and CEC as a representative of an interested state. Hearing sessions were held February 26-28, 1980; March 3-6, 1980; April 8-11, 1980; April 14-17, 1980; May 6-10, 1980; and May 12-14, 1980. (Appendices A and B identify the testimony of all parties and the exhibits are identified and if received, so noted.)

13. This proceeding is unique in the fact that although we did not have intervenors admitted under 10 CFR 2.714 participating in the hearing, CEC on behalf of the State of California assumed a role more active than that customary for a state. Although CEC alleged that it took no position on the issues (Tr. 349), its witnesses freely expressed their personal opinions, often adverse to SMUD's facility and operation. CEC also conducted a vigorous, lengthy, in-depth cross-examination of SMUD and Staff witnesses, on occasion lasting several days for a witness or panel of witnesses. *De jure* it may or may not have been a "contested" hearing but *de facto* it was contested indeed. At the conclusion of the hearing, the Board expressed its appreciation to CEC for the valuable aid it rendered in developing a full record. Tr. 4285.

14. This is a special proceeding. Due to the unique nature of the proceeding, the Board requested the parties to brief the scope of our jurisdiction and we have considered the scope in a continuing reviewing process. The Commission determined those matters we should consider and we were directed to evaluate the petitions and, if appropriate to conduct an evidentiary hearing within the jurisdiction the Commission has delegated. *Portland General Electric Company* (Trojan Nuclear Plant), ALAB-534, 9 NRC 287, 289 n. 6 (1979); *Union Electric Company* (Callaway Plant, Units 1 and 2), ALAB-527, 9 NRC 126, 144 (1979); *Public Service of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-316, 5 NRC 167, 170 (1976).

15. Both the Staff and CEC concluded that we were called upon to consider the adequacy of the immediate effective order issued by the Commission albeit with different final conclusions. SMUD takes the position that the adequacy of the Commission Order is not before us insofar as we might find additional long-term requirements necessary, but if we should so find the appropriate procedure would be to conclude the initial decision with a recommendation that the Commission issue a show cause order. The Commission Order of June 21, 1979, directed the Board to determine if the short-term actions are "necessary and sufficient" and also

should SMUD be required to accomplish the long-term modifications and if so are the long-term modifications "sufficient to provide continued reasonable assurance that the facility will respond safety to feedwater transient." (On July 11, 1979, the Commission authorized the Board to inquire into management competence and control.) The June 21 Order also stated "In the event that a need for further enforcement action becomes apparent, either in the course of the hearing or at any other time, appropriate action can be taken at that time." Based on the clear language of the Order, we have determined that our charge is to consider the adequacy of the short and long-term actions and if we determine they were not adequate to recommend to the Commission that it issue a show cause order.

16. The findings of fact set forth the issues considered by the Board in the hearing. They addressed fundamental aspects of the B&W nuclear steam supply system, related balance of plant design features at Rancho Seco, SMUD's operating procedures, SMUD's management and plant operators, Rancho Seco's control room configuration and diagnostic instrumentation as they relate to feedwater transients, and certain plant modifications suggested by the issues and testimony. We have not, however, conducted an additional investigation into the accident at TMI-2, nor assessed the necessity or adequacy of the many requirements, other than those in the Commission's Orders pertaining to this proceeding, which have been imposed upon SMUD, other B&W plants, other PWRs and other operating plants generally. The Commission did not direct this Board to determine SMUD's compliance with the May 7 Order. The Commission has specifically delegated this responsibility to the Director, Office of Nuclear Reactor Regulation. 44 Fed. Reg 27779, 27780 (1979). While we have inquired into many facets of the TMI-2 accident and the post-TMI-2 requirements, and the incidents occurring at the B&W facilities at Crystal River and Arkansas, this decision will be limited to that jurisdiction conferred by the Commission. That is: (1) whether the actions and modifications required by the May 7 Order provide reasonable assurance that the Rancho Seco facility will respond safely to feedwater transients, and (2) whether SMUD's management and plant operators are sufficiently competent to operate the plant in a safe manner.

17. Any proposed findings of fact and conclusions of law submitted by the parties hereto which are not incorporated directly or inferentially into this Initial Decision are herewith rejected as being unsupportable in law or in fact, or as being unnecessary to the rendering of this Initial Decision.

II. FINDINGS OF FACT

A. Integrated Control System

18. **Board Question H-C 16:**
Is the failure mode and effects analysis for the Rancho Seco integrated control system complete and adequate?

One of the long-term actions directed by the Commission in its Order of May 7, 1979, was that "[t]he licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC Staff as soon as practicable." 44 Fed. Reg. at 27779 (1979). Such an analysis was performed by B&W for Licensee as part of B&W's study of the reliability of the integrated control system ("ICS"). The results of B&W's reliability study are contained in B&W Report BAW 1564, "Integrated Control System Reliability Analysis." CEC Ex. 3.

19. In order to assess the completeness and adequacy of B&W's analysis, it is important first to understand the Rancho Seco ICS and the Staff's concerns regarding it. The ICS is an automatic control system whose basic function is to continuously match the unit's power generation to its load demand. The ICS does this by coordinating the rate of steam generation and the steam flow to the turbine. NRC Staff Testimony of Dale F. Thatcher Relative to the Integrated Control System (Board Question 16), following Tr. 1163 ("Thatcher ICS Testimony"), at 2.

20. During normal operations, the ICS provides proper coordination of the reactor, steam generator, feedwater control, and turbine. Proper coordination consists of producing the best load response to unit load demand within the limitations and capabilities of the plant equipment. *Id.* at 3.

21. The ICS includes four subsystems: unit load demand control, integrated master control, steam generator control, and reactor control. *Id.* at 2. Each of these subsystems (except for the unit load demand control) regulates and interacts with a number of other plant control systems, such as the control rod drive system and the feedwater pump and valve controls. *Id.* at 3. The ICS can maintain a constant average reactor coolant temperature at power levels between 15% and 100% of load and can maintain constant steam pressure at all loads. *Id.* at 3. During load changes and system upsets the ICS applies signals to control major parameters (feedwater flow, steam pressure, reactor power and reactor coolant temperature) in such a manner as to achieve optimum overall plant response without challenging the safety systems. Testimony of B. A. Karrasch and R.C. Jones, fol. Tr. 535 ("Karrasch-Jones") at 7-9. It has been

demonstrated that the ICS can reduce power from 100% to 15% and maintain that level should the turbine trip without calling upon the reactor's protective systems (Karrasch-Jones testimony at 10), although presently an anticipatory reactor trip on turbine trip has been added so that the ICS can no longer perform this function. *Id.* The ICS was thus designed to keep the reactor on line during off-normal conditions and enhance plant availability. *Id.* at 7; Tr. 1076. If, because of protective system actions, the reactor does shut down, the ICS will control steam pressure and maintain a preset steam generator level by controlling steam and feedwater, so long as either main or auxiliary feedwater is available. Tr. 1105, 1118, 1119.

CEC has emphasized, both in its cross-examination and in its Proposed Findings, the notion that it is the sensitivity of the B&W steam supply system to secondary side conditions which makes the ICS necessary and which, therefore, makes reliability of the ICS a very important matter. CEC Proposed Findings at 31-32; Tr. 1103-1105. Both Staff and Licensee emphasize the similarity of the ICS to the systems used at other power plants, including fossil-fueled plants. Staff's Proposed Findings at 11; Licensee's Proposed Findings at 24; Karrasch-Jones Testimony at 7. It appears that, in the days shortly after the TMI-2 accident, the Staff was concerned that the ICS could cause or contribute to an incident. Thatcher ICS Testimony at 5; CEC Ex. 26 at 1-5, 2-9. In particular, the Staff then believed that an ICS malfunction could prevent auxiliary feedwater (AFW) from being supplied during a loss-of-main-feedwater transient or could cause such a transient. *Id.*; Tr. 1270-72.

23. The first concern was addressed on a short-term basis in the Commission Order of May 7, 1979, by requiring Licensee to "[d]evelop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control." 44 Fed. Reg. at 27779 (1979). The adequacy of Licensee's compliance with this aspect of the May 7, 1979 Order was established by the Staff by visiting the site and conducting examinations of the operators to verify the adequacy of their training. This evaluation included a walk-through of some of the procedural aspects of manually controlling AFW independent of the ICS and a review of plant diagrams to verify that the valves that would be utilized for AFW flow control were indeed independent of the ICS. Thatcher ICS Testimony at 4, 5; Tr. 1386, 3730, 3731; Staff Evaluation at 13.

24. A permanent solution to the first concern has been provided by Licensee's commitment to install during the 1981 refueling outage a safety-grade AFW control system independent of the ICS. This modification will completely remove the operation of the AFW system from the ICS. Thatcher ICS Testimony at 5; Tr. 1273.

25. It was the second concern relating to the ICS that led the Staff to ask that a failure mode and effects analysis ("FMEA") of the ICS be performed. Since the Staff was interested in the potential role of the ICS as the instigator of a transient, it sought to have an analysis made of the reliability of the ICS and the effects of failures of that system on the plant's operation. Tr. 648, 937-39; Tr. 1270-73. A FMEA is a systematic procedure for identifying the modes of failure of a system and for identifying their consequences. It seeks to determine if any *single failure* in a system can prevent the system's function. It is considered to be the first general step of a reliability analysis. Thatcher ICS Testimony at 6. Accordingly, an ICS FMEA was one of the long-term actions directed by the Commission in its Order of May 7, 1979. As a long-term action it was not a condition of restart.

26. B&W performed the FMEA as part of its reliability analysis of the ICS. It determined the expected effects upon the B&W steam system from single failures of ICS inputs, outputs and internal modules. The Rancho Seco plant was chosen specifically as a representative design for all the B&W units for the analysis. The analysis was complemented with an evaluation of field data from all B&W operating plants and a computer simulation to confirm the effects of various ICS failures on associated equipment. Karrasch-Jones testimony at 11; Staff Ex. 5 at 3. The analysis was made a part of our record as CEC Exhibit 3, "Integrated Control System Reliability Analysis," BAW-1564, August 1979, as was a review by Oak Ridge National Laboratory of the analysis (Board Exhibit 1). Also a part of the record is Staff Exhibit 5, the Staff review of both reports.

27. Fundamentally, B&W's analysis of the reliability of the ICS thus consisted of three parts: the FMEA, a computer simulation used to study the effects of failures in more detail, (both of these specific to Rancho Seco), and a review of operating experience from all B&W operating plants. Board Ex. 1 at 5.

28. The overall conclusion of the FMEA was that the reactor core remains protected throughout any of the ICS failures studied. For those postulated ICS failures which could cause reactor trip, the safety systems would operate independently of the ICS malfunction and they were assumed to operate properly. The overall conclusion from the operating experience evaluation was that ICS hardware performance has not led to a significant number of reactor trips. It was, in fact, concluded that the ICS has prevented more reactor trips than it has caused and, accordingly, its net effect has been a reduction in the number of challenges to the Reactor Protection System. It was further concluded that the FMEA shows that no ICS failure can prevent proper safety system functioning and that the

operating experience demonstrates that the ICS is a reliable system with regard to preventing plant upsets. Karrasch-Jones Testimony at 11-12.

29. The ORNL Review concluded that although the ICS and related control systems contain areas which can be potentially improved, the ICS itself has proven to have a low failure rate and it does not appear to precipitate a significant number of plant upsets. Specifically, the examination of the failure statistics revealed that only a small number of ICS malfunctions resulted in a reactor trip (approximately 6 of 162). In its review, the ORNL concluded that the ICS is a "significant asset to plant safety and availability." Board Ex. 1 at 11.

30. While agreeing with B&W's findings and conclusions and with the recommendations made by B&W for further improvements in areas relating to the ICS, the ORNL Review pointed out a number of perceived deficiencies in B&W's approach to the FMEA portion of the reliability analysis. Tr. 1706-07, 1774. Board Ex. 1 *passim.* The main criticism leveled at the FMEA by ORNL was that the scope of the FMEA was too limited, leading to results having only limited value. Board Ex. 1 at 4. The scope limitations identified by ORNL were: (1) not considering the interactions between plant safety and non-safety systems such as ICS; (2) not including analysis of failures of plant systems external to the ICS; (3) not considering multiple system failures; and (4) utilization of functional versus component diagrams as the building blocks in the analysis. Board Ex. 1 at 3, 4 and 6 through 8.

31. It was, indeed, critical language from Board Exhibit 1 that formed the basis for this Board's inclusion of BQHC 16 in this hearing. In particular, such statements as:

> "...the B&W analysis is more notable for what it does not include than for what it does include."

and "...Because of this limited scope, the results are of limited value."

(Board Ex. 1 at 3 and 4)

would surely give one pause if taken out of context. We note, however, the following points about each of the four numbered limitations of scope set forth above:

> Point (1): Interactions between safety and non-safety systems such as ICS were not considered. That is true, but such analysis was not specifically required by the NRC's May 7, 1979, order. A study of such actions is underway for all plants as a part of the Staff's "Integrated Reliability Evaluation Program" (IREP) which has as one of its

objectives to identify the risk significance of systems interactions originating in the ICS of B&W plants. Thatcher ICS Testimony at 8.

Point (2): Failures in systems external to the ICS were not included. This is beyond the scope of the May 7 Order. Actually, the B&W analysis did include some such failures in that it included failures in the inputs to and outputs from the ICS. Tr. 681-83, 1083-86.

Point (3): Multiple failures were not considered. They were not, nor is it usual to include multiple failures in a FMEA. Tr. 1083; Thatcher ICS Testimony at 6-7. Such an analysis is usually used to determine whether a single failure can prevent operation of a safety system. *Id.* The ICS has not been required to meet the single failure criterion and was not previously analyzed; such analysis can, however, be used to identify failure modes which lead to undesirable consequences. *Id.* at 7. As we noted above, no such consequences were found.

Point (4): Functional block diagrams were used rather than component diagrams to analyze the ICS. By this we mean that only the general functions of the ICS were used and failures of each functional block were considered, rather than identifying each specific piece of equipment and considering its failure. Board Ex. 1 at 6, 10. It is possible that presently undisclosed interactions between functions might be revealed by examining specific component failures. Board Ex. 1 at 6. It is also possible that (if the failure rates of specific components were known) one might estimate the probabilities of various modes of failure by that method. Tr. 1086. However, by taking the approach which they took the B&W analysts clearly met the requirements put upon them. Further, it is not clear to the Board that a component-based analysis and estimated failure rates would give a clearer picture of reliability than the "actual history" approach which B&W supplied in addition to the FMEA. We think, in fact, that the reverse is true.

32. We note that the first conclusion of the B&W analysis was that:

> 1. The [Non-Nuclear Instrumentation] power sources (external to ICS cabinets) have been vulnerable to single failures and human errors that have led to reactor trips and plant overcooling. (CEC Ex. 3 at 2-2)

and we note further that it was failure of the Non-Nuclear Instrumentation power supplies that initiated the incident at B&W-designed Crystal River-3 on February 26, 1980 while we were in session. Tr. 1737 The power supply reliability has been increased at Rancho Seco. Tr. 3703; 3717-18.

33. In other areas identified by the study, Licensee is considering changes to increase the reliability of the reactor coolant flow input signal to

the ICS (Tr. 3703-04), and has developed procedures to improve the "tuning" of the ICS to the balance of the plant, having trained operators further in ICS control. Tr. 3704-05.

34. Thus the ICS itself is even better now than it was when the B&W analysis was performed. As to what that analyses showed, even Board Ex. 1, which was, as we noted, in some respects critical, says:

> The manufacturer contends, and we agree, that (1) the system prevents or mitigates more upsets than it causes and (2) the system is generally superior to manual or fragmented control schemes. Board Ex. 1 at 15.

35. In sum we find that the FMEA was undertaken in response to certain Staff concerns, that the results of the analysis (combined with certain actions taken to alter plant and procedures) should allay those concerns, and that the FMEA was adequate and complete for its purpose. We note that it raised other issues whose resolution would be expected to yield an even more reliable and safer plant (para. 33 *supra*), and that those issues are being acted upon. Although the need to perform a broader study of the B&W control system and its role in the initiation and the mitigation of transients has been identified and it will be carried out in the IREP, we see no reason to believe that the Rancho Seco plant would present a hazard to public health and safety during the ongoing investigations and upgrading.

B. Feedwater Transients

36. **FOE Contention III(a):**

> The NRC orders in issue do not reasonably assure adequate safety because the orders fail to evaluate or comment upon the acceptability of 27 feedwater transients over the past year in nine Babcock & Wilcox (B&W) reactors, a frequency which is 50 percent greater than the corresponding rate for other pressurized reactors.

This contention by FOE apparently was inspired by an NRC Staff study (NUREG-0560), initiated shortly after the Three Mile Island accident, to assess the effect of feedwater transients on B&W reactors. While reviewing the significant feedwater transients that had occurred at B&W plants, the Staff also reviewed the operating experience at all PWR plants from March 1978 to March 1979. The events reviewed in this study were simply the cases where forced plant shutdown resulted from a feedwater system malfunction. The study was described by Staff witnesses as "cursory in nature," designed to see if "a vast difference" in feedwater related malfunctions existed for the various vendors. The results showed that nine operating B&W plants had experienced 27 feedwater related transients over the time period studied. That was not felt to be an appreciably higher frequency than for the other vendors. The Staff also expressed the thought that the greater number of feedwater transients may have been due to the generally younger age of B&W plants. In any event, a somewhat greater frequency of feedwater related transients was not by itself considered by the Staff to be a safety concern. NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Acceptability of Feedwater Transients Referenced in NUREG-0560 (FOE Contention IIIa), following Tr. 1163 ("Rubin-Novak Feedwater Testimony"), at 3. In a more recent examination of feedwater transients at all operating plants since the Three Mile Island accident, the NRC Staff found that a substantial portion of the reactor trips that occur in all PWRs are associated with feedwater transients and that B&W was second among the three PWR vendors (B&W, Combustion Engineering and Westinghouse) in the number of feedwater transients per plant. Tr. 3754.

37. CEC's witness testified that B&W plants historically have been more prone to feedwater transients than other PWRs, but he relied on the same study (NUREG-0560) discussed above in Paragraph 36. Prepared Direct Testimony of Clifford N. Webb Concerning Design Sensitivities of the Babcock and Wilcox Nuclear Steam Supply, following Tr. 1801 ("Webb Testimony"), at 5, n. 5. Data presented by Licensee for the year 1978 show that the frequency of feedwater transients causing reactor trip at B&W reactors was less than the corresponding rate for other PWRs. Karrasch-Jones Testimony at 13, 14.

38. Licensee's witnesses reviewed the data contained in the NRC "Gray Book" (NUREG-0020, Operating Units Status Report) for the calendar year 1978. Those data revealed that the frequency of feedwater transients causing reactor trips at B&W reactors during 1978 was less than the corresponding rate for other PWRs. Thus the Licensee contends that the above-stated contention is not correct. Karrasch-Jones Testimony, at 13.

39. The Board is aware of the fact, as pointed out in CEC's Proposed Findings at 56-57, that there is a distinction to be drawn between transients which *cause a reactor trip* and those which *force a plant shutdown*. Because of that fact, and because of the changes in high pressure trip set point relative to pilot-operated relief valve (PORV) setting and the addition of an anticipatory trip, recent experience with B&W reactors may not be directly comparable with previous experience, and previous experience with B&W reactors may not be directly comparable with that at other plants, which already had anticipatory trips. CEC Proposed Findings at 57, CEC Ex. 26 at 2-3. Nevertheless, the data of Karrasch and Jones compared reactor trip frequencies and the differences do not seem great to us in either direction,

B&W reactors having shown 50% greater frequency than others before the change (Webb Testimony at 5) and ranking second thereafter. Karrasch-Jones Testimony at 14.

40. A Staff Witness (Capra) told us there are no specific criteria for transient frequency (Tr. 1267), and we see no real need for such criteria. CEC tried to convince us that there is some validity to the contention at bar by citing a Staff Witness (Capra) to the effect that:

> Personally I don't think it...is a good idea to me to have transients of that nature, such as Crystal River Three or TMI. Tr. 1268.

In context, we believe the witness meant what he said, "transients of that *nature*" not "transients of that *frequency*." We certainly agree that incidents like those at TMI-2 or Crystal River 3 are not "a good idea." But we do not believe that the evidence shows that more consideration should be given to the frequency of transients mentioned in FOE Contention III(a).

41. We, therefore, find the contention without merit.

C. Once Through Steam Generator Sensitivity

42. **Additional Board Question 3:**

> It appears from a Board Notification issued by R. H. Vollmer on December 5, 1979, that the basic design of the Once Through Steam Generator (OTSG) may so closely couple primary system behavior to secondary system disturbances that gross disturbance of the primary system is inevitable for feedwater transients. Further, it seems there are situations in which an operator may not be able to tell exactly what is wrong or what response is appropriate (e.g., overcooling *vis-a-vis* a small-break LOCA).
>
> a. What changes in the system and procedures have been made to ameliorate this situation?
>
> b. What are the implications for safety of operating Rancho Seco before any uncertainties are resolved?

B&W plants use a once through steam generator (OTSG) design rather than U-tube steam generators which are used in other PWRs. Each B&W steam generator has approximately 15,000 vertical straight tubes, with the primary coolant from the reactor entering the top at 603-608°F and exiting the bottom at about 550°F. Primary coolant flows down inside the steam generator tubes while the secondary coolant flows up from the bottom on the shell side of the OTSG. As the secondary coolant moves upward it gathers heat and turns to steam and then the steam becomes superheated before exiting to the steam piping system leading to the turbines. NRC Staff Testimony of M. P. Rubin and T. M. Novak Regarding the Sensitivity of the Once Through Steam Generator Design fol. Tr. 1163 ("Rubin-Novak Steam Generator Testimony") at 3.

43. Primary-to-secondary heat transfer is strongly dependent on the rate of feedwater introduction to the generator, for the feed rate establishes the steam generator's secondary coolant level, and the tube bundle length that is exposed to liquid secondary coolant depends upon that level. This variation of heat transfer length with inventory allows a constant primary coolant temperature to be maintained over the load range from 15 to 100% power, and results in a rapid primary system response to feedwater flow changes. The close coupling also allows the steam generator secondary side to borrow energy from the primary coolant to promptly increase load and to store energy in the primary coolant to rapidly decrease load. This results in a nuclear steam system design which can automatically respond to feedwater flow changes and maintain the reactor within the limits of the Reactor Protection System. Karrasch-Jones Testimony, at 17. The design also has certain operational advantages in that it appears to offer longer turbine life, better tube integrity and increased thermal efficiency compared to the U-tube steam generators used by other manufacturers. Staff Ex. 4 at 5-18.

44. The OTSG has, however, certain increased sensitivities to transient conditions. For example, it will boil dry (thus losing its cooling capacity) more quickly than U-tube designs. Upon a complete loss of feedwater flow, the B&W OTSG will boil dry in approximately 4 to 5 minutes whereas U-tube design generators can maintain some cooling capacity for about 15 to 30 minutes. (Before installation of the anticipatory trip discussed below, the OTSG would boil dry in approximately one minute in the event of a loss of all feedwater.) Webb Testimony at 6; Tr. 588.

45. Further, disturbances in the secondary system tend to propagate very quickly to the primary. The OTSG heat transfer is determined by the liquid level in the secondary side: liquid in the feedwater side of the OTSG removes heat from the primary side much more rapidly than steam. Thus the heat transfer rate depends on whether there is liquid or steam present on the shell side of the heat transfer tubes. If feedwater flow decreases and causes the OTSG liquid level to drop, the heat transfer in the OTSG drops and, if it drops too far, can cause the primary system to be undercooled. Conversely, if the liquid level in the OTSG becomes excessive, the primary system can become overcooled. Thus, the B&W design is sensitive because small changes in feedwater flow cause relatively large changes in the OTSG liquid level and these changes in the OTSG liquid

level cause changes in the rate of heat transfer from the primary system. This tends to couple the primary system very closely to secondary side conditions. The response of the primary system pressure and the pressurizer level to feedwater flow rate is rapid, and therefore B&W designed plants are considered more sensitive and susceptible to feedwater transients than the other types of PWR reactors. Rubin-Novak Steam Generator Testimony at 4.

46. The nature of the B&W OTSG sensitivities has been well documented (see e.g. NRC Memorandum of Oct. 25, 1979, Exhibit 1; Webb Testimony *supra*), and it was the general recognition of these sensitivities that prompted the Board to ask Additional Board Question 3. The Staff had already reviewed the B&W system and in its Order of May 7, 1979, the Commission especially noted the following design features as contributing to the system's sensitivity to secondary-side transients:

1. The design of the steam generators to operate with relatively small liquid volume on the secondary side.
2. The lack of direct initiation of reactor trip upon occurrence of off-normal conditions in the feedwater system.
3. Reliance of an Integrated Control System (ICS) to automatically regulate feedwater flow.
4. Actuation before reactor trip of a power operated relief valve (PORV) on the primary system pressurizer.
5. A low steam generator elevation relative to the reactor vessel which provides a small driving head for natural circulation.

Because of these features, the Staff determined that B&W reactors placed more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the ICS, and the emergency core cooling system (ECCS) to recover from anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR plants. This, in turn, was thought to place a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients. Rubin-Novak Steam Generator Testimony, at 4-5; Karrasch-Jones Testimony, at 21-22.

47. As a result of the lessons learned from TMI, the Staff required and the Licensee implemented the following actions to minimize the Rancho Seco plant's sensitivity to loss of feedwater transients (Karrasch-Jones Testimony at 22-24):

(a) The high reactor coolant pressure trip setpoint was lowered and the setpoint for the pressurizer PORV increased. This will minimize challenges to the PORV and the possibility of the valve sticking open, causing a small-break LOCA, and aggravating a transient situation.

(b) The reliability of the auxiliary feedwater to provide secondary cooling has been enhanced, including providing direct indication of auxiliary feedwater flow in the control room.

(c) Procedures have been developed and implemented for initiating and controlling auxiliary feedwater independent of the Integrated Control System. This further assures that a malfunction in the normal feedwater control system does not impair the ability to deliver auxiliary feedwater.

(d) Hard-wired reactor trips on loss of main feedwater and turbine trip have been installed. These result in the prompt decrease in core heat generation in the event of a loss of feedwater or turbine trip. This provides an additional margin for avoiding a reactor coolant pressure increase which might challenge the pressurizer power operated relief valve and further minimizes the possibility of the PORV sticking open, causing a small-break LOCA, and aggravating a transient situation.

(e) Additional small-break LOCA analyses have been performed and operator guidelines developed and implemented. This provides further assurance that adequate core cooling will be maintained in the event of a small-break LOCA.

(f) A failure modes and effects analysis has been completed which confirms the reliability of the Integrated Control System.

(g) Additional operator training has been provided with respect to auxiliary feedwater operation, natural circulation management, small-break LOCA mitigation, and multiple failure transient response.

48. The Staff has found that the above actions taken at Rancho Seco provide the necessary assurance that the facility will respond safely to a loss of feedwater transient. Rubin-Novak Steam Generator Testimony at 7.

49. With respect to overcooling (excess feedwater) transients (it was, of course, just such a possibility that is mentioned as a specific source of possible operator confusion in Additional Board Question 3, above) the Staff does not believe any problem exists. The inability to quickly differentiate between a small break LOCA and an overcooling transient is, Staff witnesses allege, tolerable because the immediately required manual actions are the same for both events. Rubin-Novak Steam Generator

Testimony at 9; Karrasch-Jones Testimony at 48. In any event, analyses performed both by Licensee and Staff show that overcooling of the primary system cannot, by void formation or otherwise, result in inadequate core cooling. Rubin-Novak Steam Generator Testimony at 9; Karrasch-Jones Testimony at 43-45.[4] Nevertheless, the Staff has identified several studies which are currently underway to further evaluate the B&W sensitivity for both loss of feedwater and excess feedwater transients. These studies may recommend various system modifications to further reduce sensitivity and transient frequency. The Staff believes that, while the current level of safety in B&W facilities is acceptable, changes may be possible which will provide greater levels of protection and enhance the defense-in-depth concept. Therefore, while design changes on Rancho Seco to reduce sensitivity will be considered, the Staff has concluded that there are no significant safety problems with deferring these changes until review is completed. Rubin-Novak Steam Generator Testimony at 9. Indeed, one Staff witness pointed out that there may be adverse safety implications in making changes too quickly, saying:

> ...[I]t is quite possible that one, two, or more of these recommendations may have some detrimental effects. Tr. 3735.

50. We recognize the position adopted by CEC, *viz*, that the B&W OTSG does couple primary and secondary disturbances closely and that the changes made so far are not "basic design changes which eliminate this close coupling." CEC Proposed Findings at 29. However, CEC's witness, although calling the measures mandated by the Commission's Order "inadequate," was not able to suggest any design measures which could be depended upon to increase safety. Indeed, he stated: "I have no easy solutions to this problem." Webb Testimony at 13-14.

51. Certainly the changes thus far made consist largely of procedural, set-point, and instrumentation changes. They do not alter such factors as the low secondary side water inventory or the very variable heat transfer capacity of the OTSG. They all seem to us to be sensible, conservative changes, however, and all are directed toward maintaining defense-in-depth toward an accident like that at TMI-2. Indeed, as one of CEC's own witnesses put it:

> [W]e are triple locking this particular [barn] door because a horse really did escape through it. Tr. 522.

[4]We recognize that CEC's witness Webb disagrees with the behavior predicted when overcooling produces voiding. Tr. 1914. We treat this matter at greater length in Paragraphs 80 through 84 *infra*.

Overall we accept the Staff's conclusions that the changes ordered by the Commission represent an appropriate present response to the system's secondary-to-primary sensitivity, but that further study is warranted. We see no remaining uncertainties in the system's behavior which would present serious adverse implications for safety.

D. Anticipatory Reactor Trips

52. **Board Question H-C 9**

Has the reliability of the recently installed control grade reactor trip on loss of feedwater/turbine trip been adequately demonstrated?

Additional Board Question 1:

At a meeting with owners of B&W reactors held on August 23, it was noted that, in the interim then elapsed since the TMI-2 accident, control-grade hard-wired anticipatory reactor trips (ART) had been called on to respond four times and had failed once:

a. Is this typical of performance by control grade trips?

b. What are the safety implications for operation of Rancho Seco before such trips are upgraded?

One of the concerns that developed in the aftermath of the TMI-2 accident was that the B&W reactor protection system design did not produce a reactor trip directly on loss of both main feedwater pumps or turbine trip. The B&W system was originally designed to be capable of rapidly reducing power following a turbine trip instead of completely tripping the reactor. The reactor would trip only after the reactor coolant reached the high pressure setpoint (2355 psig). As a result of this design, the reactor trip at TMI-2 occurred only after the high pressure set point was reached in the primary system, at a time about eight seconds after the loss of main feedwater and turbine trip occurred. Since the PORV was set to open at 2255 psig, it opened before the reactor shut down. The concern was that, with the situation as it was, the PORV could open every time a loss of feedwater or a turbine trip occurred, raising the potential that the PORV might fail to reclose as it did at TMI-2. NRC Staff Testimony of Dale F. Thatcher Relative to Direct Initiation of Reactor Trip Upon the Occurrence of Off-Normal Conditions in the Feedwater System, following Tr. 1163 ("Thatcher on Reactor Trip") at 2,3.

53. In order to limit the reactor coolant pressure rise and to reduce the likelihood of actuating the PORV, and thereby reduce the potential for failure to reclose, the Commission required certain modifications to B&W

facilities. One such requirement was to lower the existing high pressure reactor trip setpoint (from 2355 to 2300 psig) and to raise the PORV automatic opening setpoint (from 2255 to 2450 psig). This action would therefore minimize the likelihood of automatic opening of the PORV by having reactor trip occur earlier (at a lower pressure) and thus limiting the subsequent reactor coolant system pressure rise. *Id.*

54. To provide additional margin to the PORV setpoint, a hard-wired control-grade reactor trip was required at Rancho Seco prior to the resumption of operation on July 5, 1979. That trip was to be actuated on loss of main feedwater and/or turbine trip. It is called an "anticipatory trip" because it trips the reactor in anticipation of a high pressure trip. Licensee's Testimony of R. A. Dieterich fol. Tr. 1988 ("Dieterich Testimony") at 14. These reactor trip signals are independent of the existing reactor protection system and can provide an earlier trip for events involving the loss of both feedwater pumps or turbine trip, and therefore, can aid in limiting the reactor coolant pressure rise and subsequent operation of the PORV. Thatcher on Reactor Trip at 3-4.

55. An additional, and perhaps more important, reason for implementing hard-wired reactor trips upon loss of main feedwater and turbine trip was to minimize the transient response of the plant to secondary system upsets, a matter discussed at some length in the preceding section of this decision. Tr. 1079, 1641. The presence of the anticipatory trip on loss of main feedwater results in a prompt decrease in core heat generation (8 to 10 seconds earlier than the high pressure trip would provide) so that the steam generator inventory is not depleted as rapidly as it would be if a trip had been delayed until the high pressure setpoint was reached. Tr. 928, 929. This prompt decrease in core heat generation adds 3 to 4 minutes to the potential steam generator dry-out time and, in turn, results in at least 3 to 4 additional minutes in which to reestablish a heat sink in the system. Tr. 588-589, 1443-46, 1753-54.

56. The trip had been specified as "control-grade" rather than "safety-grade." Thus it was not required to meet certain safety system requirements and it was expected that some failures might occur. Thatcher on Reactor Trip at 6. As we noted in Additional Board Question 1, *supra*, in the period before we actually began these hearings it appeared that experience with similar reactor trips on other B&W reactors suggested a high probability of failure. It was because of that we asked Additional Board Question 1 and retained Board Question HC-9 after Intervenors Hursh and Castro withdrew. It was our intent to put into perspective the reliability of these trips *vis-a-vis* the extent to which they are being relied upon for safety. After due consideration we note the following:

The Rancho Seco hard-wired control-grade reactor trip utilized existing plant equipment to a large extent and the circuitry has been designed to the highest industry standards to provide high reliability of operation. This circuitry is comparable in quality and reliability to other control-grade circuitry installed at Rancho Seco, such as the turbine generator controls, which has proved extremely reliable in over five years of operation. The trip functions were pre-startup tested, verified by the Staff, and have operated successfully in three turbine trips at Rancho Seco. In addition, they have been tested monthly without any failures. Dieterich Testimony at 15; Thatcher on Reactor Trip at 4.

57. The control-grade trip is independent of the existing safety grade reactor protection system and, therefore, its failure would not result in the loss of the reactor trip function because the reactor protection system would cause reactor trip as before on high reactor coolant pressure. Accordingly, control-grade circuitry is considered acceptable in the short term because the trip does not perform a direct safety function but merely operates in anticipation of the possibility of the unit reaching a safety limit. It provides an additional margin of safety because in the event of a loss of feedwater or turbine trip it will result in a reactor trip and a prompt decrease in core heat generation. Karrasch-Jones Testimony at 27; Dieterich Testimony at 14, 15; Thatcher on Reactor Trip at 5. The Board concludes that the control-grade circuitry on these trips is acceptable in the short-term not only because the trips have performed as expected during testing and operation at Rancho Seco, but also because the trips do not perform a direct safety function. Accordingly, we do not believe that there are any adverse safety implications for operation of Rancho Seco with these hard-wired control-grade reactor trips before such trips are upgraded to safety-grade. The Licensee is committed to installing safety grade trips in the next few months. Dieterich Testimony at 15.

58. The high failure rate (one out of four failures when challenged) experienced prior to August 23, 1979 resulted in the B&W licensees improving their schedules for the installation of the safety-grade trips. Further experience, as of February 8, 1980 shows that the anticipatory trips have operated successfully eleven out of twelve times at B&W designed plants during loss of feedwater and turbine trip transients. Dieterich Testimony at 16; Thatcher on Reactor Trip at 8-9. Accordingly, we believe the earlier failure rate is not indicative of poor system reliability and, in any event as noted above, trip failure will not *per se* result in violation of any safety limit.

E. **Pressurizer and Quench Tank Sizing**

59. **Board Question H-C 21:**

> Do the fundamental transient assumptions utilized in sizing Rancho Seco's pressurizer and quench tank truly represent extrema, or are there other expected transients (or even transients already experienced elsewhere) which call for the greater capacity in these pieces of equipment?

The pressurizer is a cylindrical vessel with the same design pressure as the reactor coolant system. It is considered to be an integral part of that system. Karrasch-Jones Testimony at 28; NRC Staff Testimony of Philip R. Matthews [on] Adequacy of the Pressurizer and Pressurizer Relief Tank Size (Board Question 21), following Tr. 1163, ("Matthews Pressurizer Testimony"), at 2, n. 1. The purpose of the pressurizer is to provide a gas volume to accommodate pressure and density changes in the reactor coolant system during normal operating conditions as well as during anticipated transients. Karrasch-Jones Testimony at 28; Matthews Pressurizer Testimony at 2. During normal operating conditions, the pressurizer is partially filled with water in saturation with steam. A decrease in reactor coolant system temperature and pressure causes some of the water in the pressurizer to flash to steam, thus helping to maintain reactor coolant system pressure. Heaters in the pressurizer then come on to maintain temperature and pressure. Conversely, on an increase in reactor coolant system temperature and pressure, water is sprayed into the steam space of the pressurizer to condense steam and reduce pressure. Spray and heaters are controlled by the system's pressure controller. Matthews Pressurizer Testimony at 3.

60. The fundamental criterion to which the pressurizer is designed is the one applicable to the entire reactor coolant system, General Design Criterion 15 of 10 CFR 50 Appendix A, which states:

> The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant system pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Matthews Pressurizer Testimony at 5.

The criterion for pressurizer volume is that it should neither empty nor overfill during expected transients. The pressurizer volume was determined by adding the minimum volume of reactor coolant to be maintained following a reactor trip, the maximum volume change to be expected following such a trip from full power, the maximum volume change to be expected during normal operating conditions, and the maximum expected increase in volume due to a turbine trip. This sum, increased by an appropriately conservative engineering factor, gave the total design volume of 1500 cubic feet for the Rancho Seco pressurizer. Karrasch at Tr. 784; Karrasch-Jones Testimony at 29, 30. Thus the intent is to make the pressurizer large enough to prevent the water in it from going below a fixed level if the reactor trips at full power and to prevent the pressurizer from overfilling if the turbine trips at full power.

61. There are, however, a number of transients or accident conditions that can theoretically result in emptying the pressurizer or causing it to go water-solid. For instance, emptying of the pressurizer is possible for short periods of time during a depressurization or over-cooling transient such as the overcooling transients described in Section F, *infra*. Overfilling, or going water-solid, occurred during the non-nuclear instrumentation failure at Crystal River 3 on February 26, 1980. Tr. 1685. Emptying of the pressurizer can occur in an overheating transient in which the "feed and bleed" mode of core cooling has been exercised for an extended period of time and then there is a start of AFW delivery. Tr. 1128. An anticipated transient without reactor trip could cause pressures beyond the pressurizer's design criteria. Tr. 1680. Finally, the design basis for sizing the pressurizer does not seek to accommodate continuous fluid inventory losses that may occur due to a break in the system. Tr. 1127, 1128; Tr. 1681. Thus, conditions such as existed during the early phases of the TMI-2 accident (*i.e.*, loss of coolant through a stuck-open PORV) were not accommodated in sizing the pressurizer. Tr. 1681, 1682.

62. Although some of the conditions mentioned above could, in theory, lead to the pressurizer emptying, analysis of operating data from all B&W PWRs shows that in every instance of a reactor trip, including those involving overcooling events, the pressurizer liquid volume was maintained, *i.e.*, the pressurizer did not empty. Tr. 771-773; Tr. 775-777; Licensee's Supplemental Testimony of Bruce A. Karrasch and Robert C. Jones in Response to Board Question H-C 22 ("Karrasch and Jones Supplemental") at 2.

63. We note, however, that it was only after-the-fact calculation that showed that the pressurizers involved did not empty during actually experienced transients. Level indication has, at times, been lost downscale, (Tr. 774; Staff Ex. 4 at 5-13), and the Staff has recommended that steps be taken to assure that pressurizer level indication not be lost. Staff Ex. 4 at 5-13. One Staff witness noted that that objective might be achievable without changing the size of the pressurizer (Tr. 1462-63), and we are of the opinion

that the Staff recommendation should be complied with. While loss of pressurizer level indication during a transient may not in itself be a threat to safety, the sense of "flying-blind" which it could impose on the operators is a thing to be avoided.

64. The pressurizer relief tank (also "pressurizer quench tank" or "PRT") is a vessel located within the containment and which condenses, cools and collects steam discharged from the pressurizer overpressure protection valves (the PORV and the two code safety valves). Matthews Pressurizer Testimony at 4. The PRT is not part of the reactor coolant system pressure boundary, but is an operational convenience whose purpose is to accommodate the fluid discharges produced in the few instances in which the PORV or code safety valves may lift. Tr. 943; Karrasch-Jones Testimony at 30, 31.

65. The PRT is protected against overpressure by a rupture disc sized for the total combined relief capacity of the PORV and the two code safety valves. If steam discharge pressurizes the PRT beyond the disc setpoint the rupture disc will rupture to avoid the possible failure of the entire tank. Tr. 1691, 1692; Matthews Pressurizer Testimony at 4.

66. The PRT size was determined in accordance with the sizing criteria of the Staff's Standard Review Plan NUREG 75-087, Section 5.4.11. Matthews Pressurizer Testimony at 7. The basis for PRT sizing was to accommodate the total steam discharge and discharge rate from the maximum pressure increase that the code safety valves will be subjected to during a control rod withdrawal accident. Karrasch-Jones Testimony at 31; Matthews Pressurizer Testimony at 6. This accident bounds all other design basis accidents, including loss of feedwater transients, because it results in the largest discharge to the PRT through the PORV and the code safety valves. Karrasch-Jones Testimony at 31.

67. The PRT has a 500 psig design pressure, but the rupture disc is set to rupture at a lower pressure. Tr. 1690. The PRT at Rancho Seco has a volume of 1100 cubic feet and operates with about half of that volume filled with water and the other half filled with nitrogen gas. Tr. 1691. Thus, the PRT is usually not full of liquid in cases where the rupture disc ruptures, for the entry of fluids from the pressurizer compresses the nitrogen cushion and makes the disc fail while there is still a gas phase within the tank. Tr. 1469, 1470. Normally steam discharged into the tank is cooled and condensed by the water contained therein. Tr. 1691. If the rupture disc pressure is not exceeded, the water is then ultimately cooled by a cooling water system. Tr. 1694; Matthews Pressurizer Testimony at 4. If the discharge or discharge rate is such as to rupture the disc, some primary water will be blown onto the containment building floor (Tr. 954, Tr. 1464), but the Staff does not

regard this as a significant threat to health and safety as long as the material released remains confined to the containment building. Tr. 954, 1686, 1770.

68. As to what operating experience has taught us about PRT behavior, there were on the order of 149 reactor trips with documented PORV openings at B&W PWRs prior to the TMI-2 accident and one incident thereafter. Tr. 1689; Staff Ex. 4 at 4-15. Three of these incidents (including TMI-2 and Crystal River 3) resulted in rupture of the PRT rupture disc. Tr. 1687.

69. Thus such a rupture is a comparative rarity, occurring only under extreme transient conditions. On the basis of the evidence presented, we find that there have been transients (and could, no doubt, be others in the future) wherein it would have been desirable to have a larger pressurizer or a larger PRT or both. There seems to us no ready way to define completely bounding extrema. One could, for example, envision a stuck PORV which stayed open long enough to fill the PRT, or an overcooling transient severe enough to empty the pressurizer (see Section F, *infra*) even if these devices were increased substantially in size. With the present parameters of operation, however, such events would be rare and would seem to pose no immediate safety hazard. We are, however, concerned about the possible adverse effect on operator response of loss of pressurizer level indication. We therefore direct the licensee and Staff to proceed directly with plans for extended pressurizer level indication.

F. Natural Circulation, Void Formation, and Small-Break, Loss-of-Coolant Accidents

70. **Board Question CEC 1-2:**
Can poor understanding of natural convection in the Rancho Seco system result in a situation that will lead to inadequate cooling despite the modifications and actions of Subparagraphs a-e?

Board Question CEC 1-4:
Will the failure of safety and/or relief valves in the Rancho Seco primary system result in an unsafe condition despite the modifications and actions of Subparagraphs a-e?

Board Question CEC 1-7:
Do the operator training actions responding to Subparagraph (d) of Subparagraphs a-e for Rancho Seco fail to give sufficient attention to providing appropriate analytical bases for operator actions?

Board Question CEC 1-10:
Is the physical configuration of the Rancho Seco primary system such as to permit unsafe accumulation of steam or other gases despite the modifications and actions of Subparagraphs a-e?

Additional Board Question 2:
We note (letter D. Ross to J. J. Mattimoe, December 14, 1979) that there is still some dispute as to the fundamental logic for Reactor Cooling Pump (RCP) trip in a small break LOCA.

a. What current instructions to reactor operators govern tripping of the pumps in small break LOCA's and upon what theory of system behavior are those instructions based?
b. What are the implications for safety of operating Rancho Seco until the exact behavior of the system in a small-break LOCA is well understood?

Board Question H-C 24:
What features of the Rancho Seco system serve to prevent or control bubble formation in the primary system following a loss-of-feedwater transient?

The Board Questions listed above all fundamentally mesh with one another in any holistic view of the plant-operator system and the response of that system to off-normal conditions. We therefore treat them all in this single section. An understanding of the ways in which the phenomena involved interact requires an understanding of the processes involved, and we present here a brief description of the fundamentals.

71. Natural circulation, or natural convection, is a process by which coolant can be circulated in the primary system without the aid of pumps. In the Rancho Seco design heat is removed from the primary fluid at a point higher in the system than that at which it is generated. (It is, of course, generated in the core and removed in the steam generators.) Removing core decay heat from the primary coolant with the steam generators (and increasing its density) at a higher elevation than the elevation at which heat is added in the core (decreasing its density) produces a force (from the density change) which maintains a continuous flow in the primary loop. Karrasch-Jones Testimony at 33, 34 and 36 (Figure 4); NRC Staff Testimony of Paul E. Norian on Natural Circulation (Board Question CEC 1-2), following Tr. 1163 ("Norian Circulation Testimony"), at 2, 3. Clearly such a process can operate in a system which is "single phase," that is, one

in which there are no voids, although its magnitude and effectiveness might be questioned.

72. Analyses have been performed, utilizing conservative assumptions over a wide range of plant conditions, to determine that natural circulation is adequate to maintain core cooling when all of the reactor coolant pumps are inoperative. Natural circulation has also been tested at other operating B&W plants. The testing confirmed that natural circulation can be initiated and maintained over a wide range of plant conditions and demonstrated that the design analyses conservatively predict the natural circulation capabilities of the plants. Karrasch-Jones Testimony at 32; Norian Circulation Testimony at 3. The analysis and the testing show a temperature difference of between 20° and 40°F for the core and in the steam generators, which results in a natural circulation flow rate between 2% and 4% of the normal flow rate with all four reactor coolant pumps in operation. This is adequate for decay heat removal, which requires only 0.6-0.7% of full flow. Karrasch-Jones Testimony at 34. Further, unplanned occurrences of natural circulation have been experienced at B&W operating plants and in all of these events, where the reactor coolant pumps were inoperative, natural circulation core cooling maintained the plant in a safe condition. *Id.* at 32, 33, 35. Perhaps the most prominent example of verification of natural circulation in a B&W plant has been at Three Mile Island where, since April 27, 1979, natural circulation with one steam generator has been removing the core decay heat. *Id.* at 35, 37; Norian Circulation Testimony at 3.

73. When voiding occurs, either because of steam bubble formation or because of the introduction of non-condensible gases into the system, the behavior is more complex. If the fluid contains only limited voids, the liquid with the entrained voids will continue to circulate around the system. As the primary system voids increase, the steam tends to separate from the liquid. This eventually results in the core being covered with a boiling liquid pool. The steam generated is transported to the steam generator and condensed. The condensed liquid travels back to the core via the cold leg piping to replenish the liquid that is being converted to steam. The resultant flow is considered another mode of natural circulation; called "reflux boiling." Norian Circulation Testimony at 3. Sufficient condensation will be maintained and the core adequately cooled during all postulated small-break accidents which require natural circulation for decay heat removal. Tr. 804, 813; Tr. 797. However, the entire spectrum of two-phase modes of natural circulation has not been examined. The Staff has recommended, in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox Designed 177-FA Operating Plants," dated January, 1980, (Staff Exhibit 2) that these modes be

experimentally verified by December 3, 1980, at the LOFT/Semiscale facilities.[5]

74. Operator action is not required at Rancho Seco to establish natural circulation cooling following the anticipated transient loss of main feedwater and loss of forced reactor coolant flow. The plant design requires only that auxiliary feedwater cooling be established, which is an automatic plant function. The operator can monitor the following parameters to determine that natural circulation has been established: reactor coolant temperature, reactor coolant pressure, steam generator level and pressurizer level. These parameters indicate reactor coolant subcooling and steam generator heat removal. Karrasch-Jones Testimony at 37; Rodriguez Testimony at 52, 53. The natural circulation occurring under these circumstances is, of course, single phase (or essentially single phase) and subcooled. If, however, a simultaneous small-break loss-of-coolant accident (LOCA) occurs, voids can be formed and the system can enter a flow regime which (as was described above) is less well-understood. The small-break LOCA could be a pipe break or a safety or relief valve which has stuck open. For small breaks (0.02 sq. ft. or less), the energy discharged through the break is not sufficient to remove the core heat decay and, therefore, natural circulation would be used to depressurize the primary system so that the core could be cooled in the residual heat removal (RHR) mode. For break sizes greater than approximately a 2-inch diameter pipe, the energy loss through the break is sufficient to remove the core decay heat so that the primary system will depressurize and core cooling with the RHR system can be established. Accordingly, with very small breaks, sufficient heat removal is the prime concern, whereas in large breaks, maintaining adequate water inventory is of primary importance. Testimony of Dr. Harold W. Lewis Concerning Natural Circulation Cooling following Tr. 477 ("Lewis Testimony") at 6.

75. Board Question CEC 1-2 was based upon a concern regarding the ability of the operators and the B&W nuclear steam system at Rancho Seco to provide adequate natural circulation for core cooling. As indicated earlier, natural circulation without significant voiding has been demonstrated through analyses, testing at operating plants, and actual experience to provide adequate core cooling. With a subcooled primary system (which would be expected with no leak), the main prerequisite to achieve adequate core cooling is that a sufficient inventory on the secondary side of the steam generator be available to remove decay heat. As long as the steam generator removes enough heat from the primary system, the density difference drives

[5]These tests are currently scheduled to be conducted at the Semiscale facility in mid-1981. We would emphasize that our decision in no way hinges upon completion of the tests.

the primary circulation adequately in the manner noted *supra.* Norian on Natural Circulation at 6; Lewis at 9; Karrasch-Jones at 33-34.

76. If, however, a primary system break is large enough to exceed makeup capacity of the HPI system voiding will occur. The voids will most likely occur in the hottest part of the system; the hot leg, and will eventually put the reactor into the reflux boiling circulation mode. As indicated earlier this mode of circulation is adequate to cool the core if sufficient inventory is maintained to keep the core covered. However, if the core is not covered steam may form in the primary system in the region of the core. At TMI just such a situation occurred. With the core uncovered, overheating and core damage resulted.

77. As may be obvious from the preceding discussion regarding natural circulation, this mode of cooling requires a high degree of understanding and judgment by the reactor operators. Although natural circulation conditions are automatically established following an anticipated loss of main feedwater and loss of forced reactor coolant flow (providing no excess of voiding occurs) the operators must determine that in fact natural circulation has been established and that the conditions of minimum voiding are satisfied. To do this, the operator must monitor four basic parameters: (1) reactor coolant temperature, (2) reactor coolant pressure, (3) steam generator level and (4) pressurizer level. These parameters indicate reactor subcooling and steam generator heat removal and hence provide the information required to confirm natural circulation cooling. Karrasch-Jones at 37, 38.

78. Although operator errors can occur, we believe that the Rancho Seco operators have a good enough understanding of the analytical basis and the associated procedures necessary to establish and monitor the natural circulation mode of core cooling. Additional operating procedures and training on the establishment of natural circulation cooling have been provided to the Rancho Seco operators since TMI-2. The operators have been audited by the Staff and although a deficiency was originally noted a final audit after additional training revealed no deficiencies. NRC Staff testimony of Bruce A. Wilson on Operator Training and Competence following Tr. 3788 ("Wilson Testimony") at 4-7.

79. The question, of course, is whether, in the event that voids form and especially in the event that the large quantities of voiding are present, a smooth transition to a reflux boiling mode with the core covered will occur. CEC would have us find (Proposed Findings para. 109 at p. 70) that when the primary system is in a voided condition after a small break LOCA, natural circulation is no longer a reliable means of providing cooling. We investigate this position further in the following paragraphs.

80. The evidence indicates that bubble or void formation can develop in the primary system in three types of events: (1) overcooling transients, (2) overheating transients, and (3) loss-of-coolant accidents. We will analyze these events in detail *seriatim* below. Overcooling of the primary system can occur due to excessive addition of feedwater to the steam generators following a reactor trip or due to demanding too much steam from the steam generators. Staff Exhibit 4 at 5-22. These events cause a reduction in the reactor coolant temperature which in turn causes a contraction of the system inventory and a decrease in reactor coolant pressure. Karrasch-Jones at 43-44. As the volume of coolant in the system decreases, a portion of the pressurizer steam space may be transferred to the reactor coolant system. This mechanism results in limited steam voids entering the system through the pressurizer line. NRC Staff Testimony of Paul Norian on Bubble Formation, fol. Tr. 1163 ("Norian on Bubble Formation") at 3. It is also possible to produce steam voids during overcooling events because of the flashing of the reactor cooling system water. As the pressure rapidly decreases in the system the liquid in the hotter portions of the system can flash to steam because of the hot metal in the upper portion of the vessel. Tr. 1669-1670; CEC Exhibit 15 at A and B-4; Norian on Bubble Formation at 2; Tr. 1437.

81. B&W has evaluated the potential for void formation during an overcooling event. This analysis is set forth in CEC Exhibit 15 and is referenced in the testimony of Karrasch-Jones at 44 and 45. On the basis of plant transient data B&W determined that the most severe overcooling transient is a main feedwater overfeed of the steam generators following a reactor trip with turbine trip. CEC Exhibit 15 at A and B-3. The analysis further shows that, while the pressurizer may empty for a short period, the steam from the pressurizer is condensed quite rapidly as it mixes with the colder fluid in the reactor coolant piping and no voids are carried far in the loops. Consequently, if the reactor coolant pumps are inoperative natural circulation will not be interrupted. In addition, the results of the analysis indicate that high pressure injection is adequate to offset the reactor coolant volume contraction and to restore pressurizer level. Karrasch-Jones at 44 and 45; Tr. 1021-1023.

82. Brookhaven National Laboratory also conducted overfeed transient analysis. Its results differed significantly from those of B&W's analysis in that it concluded that voids would be formed and persist in the primary system. Tr. 728-732. Brookhaven's overcooling transient scenario was significantly different from that of B&W. Brookhaven assumed that both main feedwater and auxiliary feedwater provide flow to the steam generators after the reactor and turbine trip and that a steam bypass valve would malfunction also and remain open. Tr. 733. This sequence of events

is not an anticipated transient; it would involve multiple failures which are not considered credible either by B&W or the Staff. Tr. 734, 1418. It should be noted that even though the Brookhaven analysis predicts that voids will be formed it does not indicate that the core will become uncovered. Voids generated in the upper head of the reactor vessel would be swept into the hot leg and afterward condensed because the reactor coolant at that point is subcooled. Thus, natural circulation would not be inhibited. Tr. 729, 730, 1094, 1302, 1325.

83. In this regard we note that CEC's Witness Webb expressed some doubt that natural circulation would be continuous and effective under these voiding conditions. Using a figure which he obtained from Brookhaven's analysis for the total volume of void which might be formed in the hot leg and an estimate for the total volume which might cause binding which he obtained from the testimony of Staff Witness Norian, he reached the conclusion that natural circulation could not necessarily be relied upon under these circumstances. Tr. 1913, 1914. We are not inclined to give Mr. Webb's testimony on this point great weight since it is based on data he did not derive himself and upon an analysis he has not studied in the depth Applicant's and Staff's witnesses have studied it.

84. We note that Staff conducted its own analysis for the case where the turbine stop valves were assumed not to close during an overfeed. That analysis indicated that natural circulation would not be inhibited. Tr. 1325. Accordingly, on the basis of the analyses by B&W, Brookhaven National Laboratory and the NRC Staff, we conclude that a feedwater transient causing excessive feedwater addition to the steam generators, although undesirable, will not result in voids that impede natural circulation. Thus adequate core cooling will be ensured. Even for severe over-cooling accidents such as steam line breaks where some void formation is indeed expected to occur, adequate natural circulation will be maintained and the core will not become uncovered. Tr. 1093 through 1095, 1324, 1325. Consequently the Board finds that there is reasonable assurance that Rancho Seco will respond safely to such overcooling events.

85. Overheating of the primary system can occur if multiple equipment failures reduce the secondary side cooling capability below that required to remove core decay heat following a reactor trip. During such an event the primary coolant temperature can increase. This will result in saturated or boiling conditions and the system pressure will rise to the PORV and/or safety valve set points. Steam created within the hot leg piping and core outlet regions would then be dispersed through the primary system if corrective action is not initiated. Such corrective action could include either the restoration of feedwater and/or the actuation of high pressure coolant injection. Karrasch-Jones at 45, 46.

86. Analyses have been performed for such a loss of feedwater with various time delays on the delivery of auxiliary feedwater. CEC Exhibit 20. The reliability of the auxiliary feedwater system will be discussed in detail later. We note here however that if the auxiliary feedwater is provided to the steam generators or if HPI is delivered to the reactor coolant system within twenty minutes, adequate cooling will be maintained. Karrasch-Jones at 46; CEC Exhibit 20, at 2; Tr. 1442.

87. CEC Witness Lewis also essentially confirmed this conclusion with his own rough calculations which show that in case of a reactor trip upon loss of all feedwater and with no other heat losses from the primary system the operators have between 10 and 20 minutes to initiate an alternative cooling mode. Lewis at 4 and 5. Accordingly, the Board concludes that with a reliable auxiliary feedwater system such as exists at Rancho Seco a loss of all feedwater is very unlikely and, even if such an event were to occur, the operators would have sufficient time as indicated above to take corrective action to prevent void formation, thereby assuring adequate cooling.

88. The situation which results from loss-of-coolant accidents is far more complex with respect to voiding than either that resulting from overcooling or from overheating transients. It was just such a small break loss-of-coolant accident which occurred at TMI-2. LOCA analyses conducted prior to TMI-2 were devoted mainly to the study of large breaks and were intended to show that adequate core cooling was maintained with the initiation of emergency core cooling. Karrasch-Jones at 46-47; Tr. 1035-1037, 1149, 1754. Slow system depressurization resulting from small break LOCA's in the reactor coolant system did not receive detailed analytical study. Typically, the smallest break size analyzed was one that would produce system depressurization without uncovering the core in accordance with the single failure criterion and other requirements imposed by Appendix K to 10 CFR 50. While the analyses in general were sufficient to show compliance with the requirements of 10 CFR Part 50.46 and Appendix K they did not provide the necessary information needed for operator action following a small break. Staff Exhibit 2 at 1-1.

89. Subsequent to the Three Mile Island accident and in response to the Commission's Order of May 7, 1979, Licensee performed additional analyses of small break LOCA's. Analyses performed subsequent to TMI-2 for small break LOCA's indicated an interesting phenomenon: If the reactor coolant pumps operate continuously throughout such a LOCA or, if they are tripped promptly upon receipt of a low reactor coolant pressure safety signal adequate core cooling is provided for all break sizes. However, for a certain range of break sizes, if the coolant pumps operate for a short while after the occurrence of the LOCA and then fail to operate the core may become uncovered. This is because the continued operation of the

pumps maintains the fluid in a mixed configuration. This results in adequate core cooling but it also causes more fluid to be discharged out the break than would otherwise occur for these break sizes. As a result of this increased loss, the fluid evolves to a very high void fraction. If the pumps are tripped after this high void fraction is reached, the water and steam will separate so that the resulting water level may be below the top of the core, and the cladding temperature will then begin to increase. The ECCS will not then provide an adequate reflooding to the core to assure that cladding temperatures are maintained within the criteria of 10 CFR 50.46. Karrasch-Jones at 53 to 54; Norian on RCP Trip at 3 to 4; Lewis at 8.

90. The phenomenon is not peculiar to B&W plants but can occur in any PWR. Tr. 1073, 1074. Since the analyses did indicate that the plants can be maintained in a safe condition during a small break LOCA without the reactor coolant pumps operating during the transient, the Staff required through I&E Bulletin 79-05-C that the pumps be tripped immediately on indication of a loss-of-coolant accident (that is, receipt of a low reactor coolant pressure safety injection signal). NRC Exhibit 4, Appendix A; Tr. 1073, 1074, 1937.

91. CEC's Witness Dr. Lewis indicated that he was not in total agreement with the requirement to trip the pumps. He stated that such a procedure may exacerbate a condition which is not a small break LOCA but which depressurizes the system and therefore simulates a small break. Lewis at 8; Tr. 487, 501. In these non-LOCA transients, Dr. Lewis felt that it was much better to continue flow through the RCP's. Accordingly, it was his opinion that NRC should place a premium on operator training and upon adequate instrumentation to differentiate between LOCA and non-LOCA transients. *Id.*

92. Another CEC Witness, Mr. Webb, agreed that tripping the RCP's would not place the plant in an unsafe condition but he, too, stressed that conditions other than LOCA's could, under the present rules, require tripping of the reactor coolant pumps and that some of these other conditions might be such that such tripping would not enhance safety. Tr. 1937, 1938. Even one of the Applicant's witnesses suggested that the trip is unjustified unless sub-cooling is lost. Tr. 3434, 3435. Thus it appears to the Board, that although the reactor coolant pump trip on low reactor pressure is at present prudent, there may be even better techniques available if we search for them.

93. In that regard, we are glad to see that the NRC is continuing to review the reactor coolant pump trip procedure because the Staff views it as not an ideal solution. Staff Exhibit 4 at 5-31. The Staff believes that improved understanding of small break LOCA's is necessary and supports the current research programs at the semi-scale and LOFT facilities which

will explore the sensitivity of operation with and without reactor coolant pump use. Norian on RCP Trip at 6. The Board endorses these continuing efforts. In the meantime, we conclude that the current procedures to trip the RCP's assure the safe operation of Rancho Seco.

94. We have treated above of voids in the primary system caused by small break LOCA's and by over-cooling transients. The voids in the cases above were steam voids. It is also possible under certain circumstances to produce voids containing non-condensible gases in the primary system. If significant amounts of non-condensible gases accumulate, these gases could interfere with decay heat removal in two ways: (1) by reducing heat transfer in the steam generators and (2) by collecting in system high points and interrupting natural circulation flow. Staff Exhibit 2 at 4 and 5.

95. Both B&W and the Staff have examined the effect of gas accumulation in the primary system during a postulated LOCA. These analyses show that the quantity of non-condensible gas produced will neither prevent natural circulation, significantly degrade the steam generator condensation heat transfer, nor invalidate the single fluid analysis model. Staff also concluded that the B&W small break analyses model which omitted non-condensible gas sources need not consider the effect of such gases on the calculated results because the analyses predicted that for all breaks which rely on the steam generators for decay heat removal core uncovery will not occur. Staff Exhibit 2 at 4-71; Karrasch-Jones at 47; Norian on Bubble Formation at 3 and 4. Staff in fact concluded that the maximum amount of noncondensible gas calculated to be available is approximately a factor of five less than the amount needed to inhibit natural circulation. Staff Exhibit 2 at 4-70. They further calculated that the effect of noncondensible gas on steam generator heat transfer is negligible. Staff Exhibit 2 at 4-71. We also note that in the proposed rule published by the Commission on October 2, 1980 (45 FR 65466) provision is made for the installation of remotely operable high point vents in all light water reactors by January 1, 1982. These vents would surely mitigate the accumulation of any noncondensible gases.

96. Board Question CEC 1-4 expressed a concern about safety problems which might arise from failure of safety and/or relief valves. Because of the modifications made at Rancho Seco revising the relative settings of the PORV and high pressure reactor trip set points and because of the added anticipatory reactor trips signals, all of which have been treated above, it is not likely that the PORV or the safety valves will be challenged for a loss of main feedwater and/or a turbine trip transient. Before these modifications were made, it was expected that the PORV would be challenged for all such transients during power operation. NRC Staff Testimony of Paul Norian on Adequacy of Safety and Relief Valves, fol. Tr. 1163 (Norian Valve

Testimony) at 4; Staff Exhibit 2 at 3-7. Failure of the PORV will be safely mitigated by high pressure injection. In the event that pressurizer safety valves also open to relieve excess reactor coolant system pressure and one or both of these valves stuck open the resulting break would be bounded by existing LOCA analyses which show that the core will remain covered and adequately cooled. Karrasch-Jones Testimony at 54 to 56; Norian Valve Testimony at 5. Consequently the Board finds that the failure of the safety and/or relief valves at the Rancho Seco primary system will not result in an unsafe condition.

97. With regard to Board Question CEC 1-7, the B&W analyses of small break LOCA's show that some immediate and follow up operator action is required during a loss-of-coolant accident. Immediate operator action is defined as that action committed to memory by the operators which must be carried out as soon as the problem is diagnosed. Follow up actions require operators to consult and follow the steps in written and approved procedures which must always be readily available in the control room for the operator's use. NRC Staff Evaluation of Licensee's Compliance with NRC Order dated May 7, 1979 fol. Tr. 362 (Staff Evaluation) at 19. On the basis of its analyses B&W developed operating guidelines as required by the Commission's Order of May 7, 1979 to define operator actions during a small break LOCA and to provide a description of the plant's behavior during a small break LOCA and the effect of the defined operator actions. *Id.;* Karrasch-Jones at 56. Revisions recommended by the NRC Staff were incorporated in these guidelines. Staff Evaluation at 20. Licensee has applied these guidelines to Rancho Seco and developed and implemented procedure changes to provide for appropriate reactor operator action. The procedures define the required operator action in a spectrum of break sizes for a loss-of-coolant accident in conjunction with various equipment availability in failures. The NRC Staff reviewed the procedures to determine conformance with the B&W guidelines; comments generated in the course of the review were incorporated in further revisions; and the procedures were approved by the Staff prior to the resumption of operation of Rancho Seco on July 5, 1979. Staff Evaluation at 20 to 23; Rodriguez Testimony at 26; Wilson Testimony at 2.

98. The required short-term actions for the overcooling, overheating, and small break LOCA events require monitoring and response to the same parameters: reactor coolant temperature and reactor coolant pressure. These two basic parameters provide the operator the information needed to recognize conditions of potential void formation, *i.e.,* insufficient subcooling of the primary system. In addition, these events require the same immediate action by the operators, that is, to assure that high pressure injection is initiated to restore coolant inventory and that it is not

terminated unless certain requirements are met. Rancho Seco procedures have also been modified to set forth the proper operator action to be taken in the event of a small break accident with the loss of forced circulation. Karrasch-Jones at 47 to 48; Reuben and Novak at 9; Lewis at 6 to 7; Rodriguez Testimony at 27 and 28 following Transcript 2948.

99. In this regard our additional Board Question 3 also expressed a concern that in the event of a feedwater transient an operator may not be able to distinguish between an overcooling event and a small break LOCA or to determine what response is appropriate. As we have indicated above, the correct short term actions for both of these events are the same, that is, restore reactor coolant inventory through high pressure injection. After these initial actions the operator can promptly distinguish between an overcooling event and a small break LOCA by observing the magnitude of the reactor coolant temperature decrease and the secondary system parameter. For an overcooling transient, the reactor coolant temperature decreases well below the normal value and abnormal secondary side conditions are expected such as higher anticipated steam generator level. If the reactor coolant temperature remains close to the normal value and the secondary side parameters are within normal limits a small break LOCA would be suspected and followup actions would be taken as appropriate. Karrasch-Jones at 48.

100. In answer to the specific inquiry made in our question CEC-1-7, no party has identified any specific operator training action responding to small break LOCAs which fails to give sufficient attention to providing appropriate analytical bases for operator actions. See, for example, Tr. 1853 and 1854. Our examination of the record of the depositions in this proceeding of the three licensed Rancho Seco operators gave us no reason to disagree with the Staff's testimony that the Rancho Seco licensed operators adequately understand the analytical basis of the actions they may be required to take pursuant to subparagraph D of the Commission's Order of May 7, 1979. Wilson on Operator Training at 7. CEC would have us find upon examining this section of the hearing that natural circulation in a voided primary system is so undependable that it cannot be relied upon to maintain cooling, that reflux boiling would also be unreliable and that when any type of voiding occurs in the primary one would expect ultimately to end up in a "feed and bleed" situation. CEC Proposed Findings at 73 through 75. CEC would also have us find that the "feed and bleed" method of cooling is in itself a weak reed to lean upon because of the possibility of jamming of pressure or safety relief valves, which CEC points out are not designed to take mixed water and steam flow. CEC Proposed Findings at 76, 77.

101. Our reasoning, based on the above testimony, would lead us to believe that the impairment of natural circulation cooling by voiding is much less severe than CEC would suggest. We feel that "feed and bleed" cooling itself is so remote a possibility as to represent a last ditch stand, an unlikely contingency at best. We conclude that the analyses performed have been adequate to demonstrate that core cooling will be sufficient, and that, coupled with the development and deployment of the appropriate operator guidelines and training, which has been done, assures that Rancho Seco can safely respond to and mitigate events including small break LOCAs, overcooling events and overheating events which may involve the accumulation of steam or other gases in the primary system.

G. Auxiliary Feedwater System Reliability

102. **Board Question CEC 1-6:**

> Will the modifications of subparagraphs a-e of Section IV of the Commission's Order of May 7 still leave the Rancho Seco emergency feedwater system in a condition of low reliability?

The Auxiliary Feedwater System is an emergency system designed to supply feedwater to the steam generators in order to remove heat from the reactor cooling system in the event of a loss of main feedwater. The AFW system must be able to operate over a time period sufficient to either hold the plant at hot standby for several hours or cool down the reactor coolant system to the temperature and pressure conditions which permit the low pressure decay heat removal system to operate. NRC Staff Testimony of Richard Matthews on Reliability and Timeliness of the Emergency Feedwater System, fol. Tr. 1163 ("Matthews on AFW") at 2; Dieterich at 6.

103. AFW systems for B&W reactors are not designed by the nuclear steam supply system vendor but by the architect engineer that furnishes the balance of plant design. Dieterich at 7; Tr. 576. Thus there are wide differences in AFW designs among B&W reactors although all the AFW systems do have to meet certain criteria specified by B&W relating to feedwater flow rate.

104. The Rancho Seco AFW system consists of two independent but interconnected subsystems or trains, each capable of supplying auxiliary feedwater to either or to both steam generators under automatic or manual initiation and control. Safety-dictated functional requirements of the AFW system are properly met if at least one train supplies water at the proper flow rate to at least one of the two steam generators following a demand for system operation. Matthews on AFW at 2.

105. The primary water source for the AFW system is the condensate storage tank. That tank has the capacity to supply auxiliary feedwater for a period of 24 hours. Two alternate water sources for the system are the Folsom South Canal and an onsite reservoir. Matthews on AFW at 3; Tr. 1491, 2057, 2324.

106. Each AFW train contains a pump capable of delivering auxiliary feedwater flow against the maximum steam generator pressure to piping supplying both steam generators. One AFW pump is driven by an electric motor; the other is a combination turbine/motor driven pump with both the turbine and electric motor on a common shaft. Each pump receives water from the condensate storage tank via separate pipes. The pumps are interconnected at their discharge by a cross connection containing motor operated valves. This cross connection permits either pump to feed either steam generator or both. Matthews on AFW at 2 and 3.

107. Each pump discharges into two parallel lines. In one line there is an air operated flow control valve whose operation is currently controlled automatically by the integrated control system. In the other line there is a motor operated flow control valve which operates independently of the integrated control system. Beyond the valves the parallel lines rejoin and go to the steam generator thus allowing automatic or manual control of auxiliary feedwater flow to the steam generators. Matthews on AFW at 3; Tr. 1491 through 93.

108. Steam supply for the turbine driven AFW pump is provided from the main steam lines downstream from each steam generator. Electric power for the AFW system components in each train is supplied from separate buses which are backed up by separate diesel generators. Upon loss of both offsite and diesel power, AFW flow can be supplied by the turbine driven pump. The AFW system is automatically initiated upon (1) loss of the reactor coolant pumps or (2) main feedwater pump low pressure. These signals start both the turbine driven and the motor driven AFW pump and open the air operated flow control valves whose position is controlled through the ICS system by steam generator water level signals. A safety features actuation signal (SFAS) from the reactor coolant system will also automatically start the turbine driven AFW pumps. But this signal will also open two motor driven valves which are in parallel with the air operated flow control valves. The AFW system can also be manually initiated by the reactor operator from the control room. Matthews on AFW at 4; Tr. 1512-13, 1534; Dieterich Testimony at 7.

109. With the exception of the fact that its automatic initiation system is control-grade the Rancho Seco AFW system meets all NRC requirements for safety-grade systems. Tr. 1314, 1493 to 94. After the TMI-2 incident, the reliability of AFW systems at Rancho Seco and at other B&W PWRs was

the subject of close scrutiny by the Staff. This scrutiny was in recognition of the fact that reliable AFW operation is an integral part of the plant's safe response to a loss-of-feedwater transient. The accident pointed up the fact that B&W PWRs respond faster to transients initiated on the secondary side of the system (for example, loss of main feedwater or turbine trip) and that timely and reliable AFW initiation and delivery are thus extremely important to these systems. Tr. 1485-86. The Rancho Seco AFW system meets the Staff's acceptance criteria as contained in Section 10.4.9 of the NRC Standard Review Plan except for certain high energy pipe break criteria. CEC Exhibit 21 at 9. Nevertheless, the Staff determined after its TMI-2 review that there was not adequate assurance that B&W plants could continue to operate without undue risk to the health and safety of the public and accordingly, pursuant to the Commission's May 7 Order, the B&W Licensees agreed to shut down their facilities until a number of short term actions were completed. Among other things these short term actions included changes to upgrade the timeliness and reliability of the AFW system. Matthews on AFW at 4 through 8; Tr. 2077-78. In our Board Question CEC 1-6, we asked whether the actions required by the Commission's May 7 Order might still leave the Rancho Seco AFW in a condition of low reliability.

110. Rancho Seco Licensee agreed to complete the required short term actions prior to restart and also to upgrade the timeliness and reliability of its AFW system by carrying out nine modifications and actions identified in its letter to the Staff dated April 27, 1979. CEC Exhibit 25. The actions committed to by the Licensee were primarily intended to assure AFW reliability by: (a) establishing procedures to train operators in the recognition and appropriate response to abnormal condition or postulated failures; (b) making sure that the AFW system was available and operable upon demand by using improved system testing and valve lineup procedures; (c) improving instrumentation in the control room to verify AFW flow to the steam generators; and (d) confirming system design reliability by periodic surveillance functional testing. Matthews on AFW at 9 through 15: Tr. 1486. Specifically, the Licensee agreed to upgrade its AFW system by completing the following actions which are identified in its letter to the Staff. CEC Exhibit 25, Enclosure 1:

1. Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor driven auxiliary feedwater pump(s) from vital AC buses upon loss of offsite power.

2. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode,

procedures will be implemented and training conducted to provide an operator at the necessary valves in phone communications with the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events.

3. Procedures will be developed and implemented and training conducted to provide for control of steam generator level by use of safety grade AFW bypass valves in the event that ICS steam generator level control fails.

4. Verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis will be conducted. Pump capacity with mini flow in service will also be verified.

5. Modifications will be made to provide verification in the control room of AFW flow to each steam generator.

6. Review and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the AFW pumps.

7. Design review and modification, as necessary, will be conducted to provide control room annunciation for all auto start conditions of the AFW system.

8. Procedures will be developed and implemented and training conducted to provide guidance for timely operator verification of any automatic initiation of AFW.

9. Verification will be made that the air operated level control valves (a) Fail to the 50% open position upon loss of electrical power to the electrical to pressure converter, and (b) Fail to the 100% open position upon loss of service air. The AFW bypass valves are safety grade.

111. The Staff reviewed the above actions and procedures and found them to be in compliance with the Commission's May 7 Order. The Staff found that these actions and procedures have improved the reliability of the Rancho Seco AFW system sufficiently to warrant continued plant operation. Staff Evaluation at 3-12; Matthews on AFW at 9-15 and 19. The Staff has concluded that:

> The AFW system and procedural changes accomplished thus far have improved overall system reliability by improving system availability

for operation upon demand; improving procedures and training to enable operators to take actions if necessary to maintain system functional capability under abnormal plant conditions; providing to the operator improved information of (sic) the system operating condition; and verifying the design capability of major system components. It is considered that these changes have improved AFW system reliability sufficiently to assure safe plant shutdown following loss of main feedwater. Matthews on AFW at 19.

112. After issuance of the Commission's May 7 Order the Staff requested that B&W PWR Licensees conduct a generic study of their AFW systems utilizing the same methodology, assumptions and data base that were previously employed in a similar AFW study by Westinghouse and Combustion Engineering PWRs. The study for the Rancho Seco AFW system was submitted to the Staff for review in December 1979 and is in the record of this proceeding as CEC Exhibit 20. This analysis was done on a more systematic basis than previous AFW system reviews and uses event tree and fault tree logic to determine any significant potential contributors to off-normal conditions. The loss of main feedwater scenarios utilized were (1) loss of main feedwater alone, (2) loss of main feedwater accompanied by loss of all offsite power, and (3) loss of main feedwater accompanied by loss of all AC power. Tr. 1582. All trees were developed to determine which human errors or equipment failures both within and outside the AFW system can preclude delivery of auxiliary feedwater to the steam generators. Matthews on AFW at 15 to 16; Tr. 1565-66, 1582-84.

113. In the analysis, the reliability of the Rancho Seco AFW system was calculated in terms of the probability that an operator will be able to take corrective action to restore AFW flow within a given period of time after the initiating event, assuming the AFW system has failed to operate. Tr. 1591. The time intervals chosen were 5, 15 and 30 minutes. These intervals were selected because NRC supplied operator reliability data for these times were available. CEC Exhibit 20 at 2; Tr. 1728, 1719. Mission success was defined in the study as attainment of flow from at least one pump to at least one steam generator. CEC Exhibit 20 at 2. Achievement of mission success within five minutes is roughly equivalent to the mission success criterion utilized by the Staff in its study of Westinghouse and Combustion Engineering plants AFW system reliability, *i.e.*, avoidance of steam generator dryout. Steam generator dryout time for B&W plants is approximately five minutes, assuming actuation of an anticipatory reactor trip on the loss of feedwater. Tr. 1593-94, 1754; CEC Exhibit 20 at 2.[6]

[6]A refinement on previous analyses has indicated that steam generator dryout assuming anticipatory reactor trip is actually closer to four than to five minutes. (Tr. 2089 through 2112).

Achievement of mission success within 15 or 30 minutes would also be important to the overall safety of the plant because adequate core cooling can be maintained for periods in excess of 20 minutes without AFW flow, providing that at least one high pressure injection pump is operating. CEC Exhibit 20 at 2; CEC Exhibit 21. Enclosure 1 at 1; Tr. 492 to 494, 519 to 522, 1484, 1586-87.

114. The Staff reviewed the Rancho Seco AFW reliability analysis and its review is contained in a letter to the Licensee dated February 26, 1980 which is a part of the record of this proceeding as CEC Exhibit 21. The Staff agrees with B&W that the Rancho Seco AFW system reliability is quite comparable to that of Westinghouse AFW systems in the event of any of the three loss of main feedwater conditions utilized in the study. However, such a statement can be made only when the criterion for mission success is that auxiliary feedwater flow be established within 5, 15 or 30 minutes of the initiating event. *Id.*, Matthews on AFW at 16; Tr. 1606, 1618. We note that the Staff believes that the mission success criterion should be revised to include the requirement to deliver the AFW flow to the steam generator before the steam generator boils dry. CEC Exhibit 21, Enclosure 1 at 1; Tr. 1597. Accordingly, the Staff has requested further analysis based on this revised criterion.

115. The Licensee disagrees that the "boil dry" criterion is the proper one to use. Licensee argues that the ultimate measure of the AFW system reliability is the ability to remove decay heat from the core to prevent core damage. Tr. 2088-89, 2093, 2107. Under any circumstances data for operator reliability at times under five minutes are not available. Tr. 2090-92. The Staff apparently feels that the "boil dry" criterion should be used without regard for the behavior of other systems that are available to protect the reactor. Tr. 1597.

116. We do not feel it is necessary to resolve this controversy in order to determine whether the short term actions required by the Commission's May 7 Order provide reasonable assurance that Rancho Seco will respond safely to feedwater transients. Our mandate is simply to determine whether the changes in design and procedures noted above have improved auxiliary feedwater system reliability sufficiently to assure safe plant shutdown following the loss of main feedwater. We do not think that the Staff's revision of the mission success criteria for AFW system reliability is crucial to this decision. Note the fact that the Staff is of the opinion that the

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This slight reduction in dryout time should have little impact on the results of the analyses. Tr. 2090-2092, 2107, Tr. 1659. If no anticipatory reactor trip takes place, the steam generator dryout time is approximately 1.5 minutes. Tr. 1594, 1753 to 54.

revision of this criterion would probably not change the relative comparability of B&W to either Westinghouse or Combustion Engineering systems. Certain of these AFW systems have low reliability because they have some specific type of vulnerability (e.g., manual initiation, one train systems, or control grade components) that would potentially inhibit the delivery of feedwater to the steam generator. Tr. 1657 to 1665.

117. The Board agrees with both the Staff and the Licensee that although steam generator dryout is an undesirable event because it results in challenging the plant's safety systems it is not an event of great safety concern. Tr. 1595, 2010-11, 2088-89. The steam generator boils dry, the primary system loses its heat sink, and the primary pressure and temperature increase until the PORV and safety valves open to relieve pressure. Tr. 1610. If either AFW is delivered to the steam generators or high pressure injection is delivered to the steam system within twenty minutes, adequate core cooling will be maintained. Karrasch-Jones at 46; CEC Exhibit 20 at 2; Tr. 1442. Thus the fact that a B&W steam generator will boil dry more quickly than a Westinghouse or Combustion Engineering steam generator in the event of loss of main feedwater is of little significance when determining whether the plant can be safely shut down because later restoration of feedwater and/or actuation of HPI assures adequate core cooling under any circumstances.

118. CEC would have us find that the items set forth above did not materially upgrade the timeliness or reliability of the Rancho Seco AFW system. CEC Proposed Findings at 46, para. 69. Six of the nine items CEC would have us dismiss as merely codifying procedures which were already known and which operators were already capable of performing. We feel that the very existence of codification under such circumstances represents an upgrade and a greater assurance that the procedures will be performed. Three of the items Nos. 5, 7 and 8, relate to instrumentation for AFW flow verification and enunciation. CEC would have us dismiss these on the ground that operators already had indirect methods to verify the proper functioning of the AFW system. In our opinion, the change from an indirect to a direct verification of auxiliary feedwater flow is, for example, an important upgrade.

119. The operating history of the Rancho Seco plant before the changes shows that auxiliary feedwater has been delivered in a timely manner on every occasion which it has been called upon to function. Tr. 1522, 3255. In fact, every instance of auxiliary feedwater flow has been provided early enough to avoid steam generator dryout. Tr. 2119. Even CEC's Witness, Dr. Lewis, agrees that the failure of AFW has been shown to be quite low. Lewis at 4. In our opinion the changes thus far mandated have made this reliability even greater. We conclude that the reliability of this system does

provide reasonable assurance that the plant can be safely shut down in the event of a loss of main feedwater. Despite its proven and improved reliability through the short term actions the Licensee has committed itself to make the following additional long term modifications:

(a) Provide a safety grade AFW automatic initiation and control system design that is independent of the ICS.

(b) Provide for the automatic loading of the motor driven AFW pump onto the diesel generator buses upon loss of all offsite power.

(c) Revise the AFW system piping and provide a remotely operated valve operated from the control room instead of the local manually operated full flow recirculation valve (FWS 055).

(d) Incorporate into the Technical Specifications a requirement to operationally verify AFW flow capability from the condensate storage tank to the steam generators following extended cold shutdown.

(e) Upgrade the existing condensate storage tank level indication and low level alarm to safety grade requirements.

(f) Upgrade the existing control room indication of AFW flow to each steam generator to safety grade.

(g) Establish procedures on how to obtain water for the AFW system from sources other than the condensate storage tank. CEC Exhibit 21, Enclosure 1 at 3 to 7; Matthews on AFW, at 17 through 19.

These changes will reduce continued dependence on operator action and thus reduce the likelihood of operator error in the long term. Matthews on AFW at 19.

120. The Board concludes that the timeliness and reliability of the AFW feedwater system at Rancho Seco is presently adequate to assure safe operation of the facility and will be further enhanced by completion of the long term modifications.

H. Safety System Challenges

121. **Issue CEC 1-1:**

Despite the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order, will reliance upon the High Pressure Injection System to mitigate pressure and volume control sensitivities in the Rancho Seco primary system result in increased challenges to safety systems beyond the original design and licensing basis of the facility?

Issue CEC 1-12:

Despite or because of the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order of May 7, will Rancho Seco experience an increase in reactor trips resulting from feedwater transients that will increase challenges to safety systems beyond the original design and licensing basis of the facility?

122. The modifications and actions taken as a result of the Commission's Order of May 7, 1979 — the addition of anticipatory reactor trips on loss of feedwater and turbine trip — combined with the changes in which the high pressure reactor trip and the PORV relief pressure settings were changed to make reactor trip occur before the PORV opens — are expected to increase the number of reactor trips at Rancho Seco. Karrasch-Jones Testimony at 39; NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Design Basis for Rancho Seco Safety Systems (CEC Contentions 1-1 and 1-12), following Tr. 1163 ("Rubin-Novak Safety Systems Testimony") at 3. The addition of the anticipatory reactor trip on loss of feedwater is not expected to increase the number of reactor trips, since such an event normally has resulted in a reactor trip on high reactor coolant system pressure. The addition of the anticipatory trip on turbine trip, however, is expected to increase the number of reactor trips, since the system previously was capable of avoiding a reactor trip during such an event. Karrasch-Jones Testimony at 10, 40.

123. From the Licensee's tabulation of data compiled from NRC publications, it appears that prior to the accident at Three Mile Island the yearly average of reactor trips at B&W plants was below the averages for Combustion Engineering and Westinghouse PWRs. Licensee now expects the reactor trip frequency for B&W units to increase and approximate the industry average. Karrasch-Jones Testimony at 40-41.

124. There is some disagreement between witnesses presented by the Staff and those presented by the Licensee as to whether increasing reactor trip frequency will, in itself, also increase the frequency of HPI actuation. Rubin-Novak on Safety Systems at 3; Karrasch-Jones at 41. Both sets of witnesses assured us, however, that this expected increase in the number of reactor trips will not result in the design and licensing basis of safety systems being exceeded. Karrasch-Jones Testimony at 39; Rubin-Novak Safety Systems Testimony at 3. During the course of designing Rancho Seco, certain criteria were established for the allowable number of plant transients which would result in thermal cycles and stress on the reactor

coolant pressure boundary. These criteria are detailed in design information supplied to the Rancho Seco operating staff in reports which describe the number of transients of each category allowed for in the basic plant design and include such plant responses as reactor trip and high pressure safety injection. Rubin-Novak Safety Systems Testimony at 4; Tr. 1449, 1450.

125. We note with some concern that one of the safety system cycling criteria, that applicable to the number of HPI initiation cycles permitted on each injection nozzle during the life of the plant, is already being approached: There have been some 30 thermal cycles on at least one injection nozzle in the plant's first six years of operation, whereas only 40 cycles were projected for the 40-year lifetime. Tr. 995, 997, 1159, 2013-18, 3358. It appears, however, that the 40 cycle projection was not based upon a known approach to a hazardous condition, that the limit may be overly conservative, and that there are several ways to cope with the matter should it become evident that a real safety limit is being approached. Tr. 2014-20.

126. We do not believe that the slight increase in reactor trip frequency noted since the change in PORV and pressure trip setpoints represents a significant increase in challenges to the reactor trip system. Certainly we see no reason to consider returning the setpoints to their original values as advocated by CEC. CEC Proposed Findings at 64. Indeed, testimony cited by CEC as suggesting that increased challenges to the safety system make the current arrangement an ill-advised approach is largely testimony directed at some other point. For example, CEC's Witness Dr. Lewis was discussing the "feed and bleed" operation and its challenge to the safety valves at the point cited (CEC Proposed Findings at 60, citing Lewis Testimony at 12, Tr. 449), a matter very different from a slight increase in reactor trip rate.

127. In sum, we see no reason why the modifications (a) through (e) of Section IV of the Commission's Order will either fail to correct or exacerbate any hazard due to challenges to the safety systems of the plant, nor do we see any reason to believe such hazards are significantly greater than those evaluated when the plant was licensed.

I. Operator and Management Competence

128. Important issues in this proceeding are operator and management competence. The question is "Does the competence provide reasonable assurance that the Rancho Seco plant will respond safely to feedwater transients?" One of the short-term actions and one of the long-term modifications required by the Commission's Order of May 7, 1979, directed Licensee to undertake additional training of its plant operators in light of the experience gained from the accident at Three Mile Island. Thus, the findings in this section will focus on (a) the training, both pre- and post-

TMI, that operators, personnel, and management receive at Rancho Seco, (b) personnel understanding of nuclear technology and the operation and fundamental aspects of the facility, (c) whether personnel are properly apprised of new information that pertains to the facility's operation, (d) whether emergency procedures are adequate and effective, (e) whether procedures to test operators are adequate, (f) whether unlicensed operators are sufficiently trained and competent and finally, (g) whether management and technical personnel are competent to manage and understand the operation of Rancho Seco.

129. The issues raised relative to this question are the following:

Issue CEC 3-1:	Whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?
Issue CEC 3-2:	Whether personnel are properly apprised of new information pertinent to the facility's safe operation and ability to respond to transients, particularly information on operating experience of other reactors?
Issue CEC 3-3:	Whether NRC and SMUD adequately ensure that emergency instructions are understood by and are available to plant personnel in a manner that allows quick and effective implementation during an emergency?
Board Question H-C 32:	What procedures have been used to test and evaluate the competence of Rancho Seco's operation personnel and management?
Board Question H-C 34:	What actions and/or programs are employed at Rancho Seco to assure that operating personnel, both licensed and unlicensed, adequately respond to feedwater transients?
FOE Contention III(d):	The NRC orders in issue do not reasonably assure adequate safety because no procedures have been taken to assure facility management competence.
FOE Contention III(e):	The NRC orders in issue do not reasonably assure adequate safety because no procedures exist or have been taken for the determination of the adequacy of operator competence.

1. Training

130. One of the primary concerns in the operation of a nuclear reactor is the training of the operating, technical, maintenance and management personnel since this training is vital for the protection of the health and safety of the public. SMUD in its various programs and the NRC through regulations, audits and testing have devised comprehensive and effective training programs and procedures. Rancho Seco has a diverse training program consisting of the following: "cold license" training; "hot license" training; a requalification program; B&W simulator training, and special post-TMI training.

131. The "cold license" training program was provided from 1970 to 1974 to the personnel initially licensed to operate Rancho Seco when it received a facility operating license in 1974. Rodriguez at 7. More than one-half of the presently licensed operating personnel received all or most of the cold license training. NRC Staff Testimony of Bruce A. Wilson on Operator Training and Competences fol. Tr. 3788 ("Wilson on Operator Training") at 3. The program included: 13 weeks of observation at an operating nuclear power plant; a 520-hour course in basic reactor physics and engineering; a 6-week PWR technology course and a 10-week simulator course presented by B&W at its headquarters; a final review training course (including a simulator refresher course); and participation in plant start-up activities. *Id.;* Rodriguez Testimony at Appendix I.

132. The "hot license" training program has been used to prepare operator candidates for licensing since the facility operating license was issued in 1974. Individuals eligible for this training program are selected for participation on the basis of a math and science written examination, an interview, and an evaluation of previous work performance.[7] Rodriguez Testimony at 7. The first part of the hot license training program consists of 600 hours of academic training and includes a mathematics course, a physics course and a related technologies course. The next phase of the program involves in-plant operations training at Rancho Seco and includes systems and operations training in the control room, the application of procedures to systems under operating conditions, and fuel handling

[7]The Licensee has attempted to hire into the Nuclear Operations Department, for unlicensed operator positions and licensing candidates, individuals with a two-year college degree in the electrical-mechanical area or equivalent experience. Tr. 3393, 3484. The NRC Staff is in the process of implementing revised criteria for license examination eligibility, including minimum experience. CEC Ex. 49, Enclosure 1 at 1, 2. *See also,* Tr. 3075.

training. The third part of the prospective operator's preparation is simulator training. (This training will be discussed separately in this decision.) Finally, the candidate undergoes a pre-license review course, including a comprehensive oral and written examination administered by SMUD. The NRC's license examination is then given to the candidate only if the Licensee certifies that the candidate is prepared. Rodriguez at 7-10 and Appendix II. Requirements for approval of the operator license application are set forth in the Commission's regulations at 10 C.F.R. Section 55.11. The scope and content of the NRC's written examinations and operating tests are set forth at 10 C.F.R. Sections 55.20 through 55.23.[8]

133. The Commission has imposed, as a condition of facility operating licenses, the requirement that the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A to 10 C.F.R. Part 55. 10 C.F.R. Section 50.54(i-1). Each operator and senior operator license expires two years after the date of issuance. 10 C.F.R. Section 55.32. Requirements for the renewal of operator licenses are set forth at 10 C.F.R. Section 55.33, and include successful completion of the requalification program. The requalification training program for licensed personnel at Rancho Seco is conducted continuously and on a two-year cycle. The program includes regularly scheduled lectures,[9] assigned individual study, on-the-job training including reactor control manipulation,[10] an annual one-week simulator course, an annual oral exam administered by Rancho Seco management,[11] and an annual written examination of comparable scope to the NRC licensing exam. Rodriguez Testimony at 11-15; Wilson on Operator Training at 4. The

[8]As a result of the accident experienced at TMI-2 the NRC Staff has specified criteria which increase the scope of the NRC license examinations. *See* CEC Ex. 49, Enclosure 1 at 4. In addition, Licensee is considering revisions to its hot license training program, in response to NRC Staff guidance, to increase the level of detail for training in heat transfer, fluid flow, thermodynamics and mitigating core damage. Tr. 3075 (Rodriguez); CEC Ex. 49, Enclosures 2 and 3.

[9]During the course of the two-year cycle an average of 60 different hours of lectures are scheduled and repeated to accommodate all licensed operating personnel. Rodriguez at 11; Tr. 3078-3079, 3087. Individuals who score sufficiently high in a particular subject area on the written requalification examination are not required to attend lectures in that subject area. Tr. 3079, 3080.

[10]Each licensed operator is required to manipulate the controls a minimum of ten times during the term of the license. Each licensed senior operator must manipulate the controls or direct the activities of operators during control evolutions a minimum of ten times during the term of the license. In meeting these requirements, credit is given for control manipulations at the B&W simulator. Rodriguez at 13, 14; Appendix A to 10 C.F.R. Part 55, paragraph 3.

[11]Members of the NRC's Performance Appraisal Branch testified that Licensee had not fully implemented the training program for licensed operators in that for several operators this oral exam was not administered within the time frame specified by the Rancho Seco administrative procedure. Supplemental Testimony of NRC Performance Appraisal Branch Regarding SMUD Management Controls, fol. Tr. 4233, at 3; Tr. 4254-56.

Rancho Seco administrative procedure governing the requalification training program may be found in the record as CEC Exhibit 35. Licensee has modified the program to include training on the lessons learned from the Three Mile Island accident. Rodriguez Testimony at 12; CEC Ex. 35 at 3. This program is audited regularly by the NRC's Office of Inspection and Enforcement and the Operator Licensing Branch. Wilson on Operator Training at 4; Tr. 3813-3815.

134. Operators at Rancho Seco receive training on the B&W simulator in Lynchburg, Virginia. Witnesses for all parties testified that simulator training is a valuable tool. Rodriguez Testimony at 9; Lewis Testimony at 13; Wilson at Tr. 3859; Bridenbaugh at Tr. 3564. The B&W simulator is very similar in design and layout to the Rancho Seco control room. Staff Ex. 4 at 5-69. The arrangement of controls, and the types of controls for feedwater control and reactor system control are essentially identical to those at Rancho Seco. Rodriguez Testimony at 9. This similarity provides an advantage for Rancho Seco operators. Staff Ex. 4 at 5-69.

135. There are, however, some differences between the Rancho Seco control room and the B&W simulator. For example, the auxiliary feedwater controls are not located in the same positions. The switch layout for adding boron to the coolant system also differs, as does the switch layout for the steam line break failure logic system. Also, some of the balance of plant systems operate differently at Rancho Seco than their counterparts at the simulator. An important difference is that the B&W simulator does not represent the dual drive AFW system present at Rancho Seco. Tr. 3084-98.

136. The simulator training provides the opportunity for the operator to participate in plant operations as a control room operator and as a supervisor of control room operators. The simulator has the capability of introducing over 60 individual casualties in reactor plant systems, including the coolant makeup system, the reactor and its instrumentation, the reactor coolant system, the steam and turbine system, the condensate and feedwater system, and various auxiliary systems. The individual casualties can be combined to create multiple failure accidents or the instructor may fail equipment sequentially. Thus, the simulator gives the operator the opportunity to practice his training and diagnostic skills on complex problems. Rodriguez Testimony at 13, 14. In the case of the post-TMI training, operators were able to observe the course of the various plant parameters while the Three Mile Island accident was demonstrated on the simulator and, in a second simulation, to exercise control to mitigate the accident. *Id.* at 16.

137. CEC believes, however, that the B&W simulator training is not of superior quality. *See* CEC's Proposed Findings, para. 168. In support of this belief, it notes that even though the general conformity of the simulator to

Rancho Seco is advantageous, it is offset by the age and fidelity of the simulator and the infrequent simulation of degraded conditions and multiple failures. *Id.* The Board is of the opinion, however, that the question is whether the overall training of Rancho Seco personnel is adequate and effective to ensure the safe operation of the facility in the event of a feedwater transient. That ultimate issue will be decided by us based, after a review of the evidence, on a finding as to whether all aspects of the training program, in addition to the qualifications and competence of the operating and management personnel, provide reasonable assurance that Rancho Seco can respond safely and appropriately to a feedwater transient. Specifically, our charge with respect to this issue is to make findings on the contentions set forth above and we would note that no party has asserted through its contentions or, in fact, shown that a simulator training program is required to be of "superior quality."[12] Suffice it to say, the adequacy and effectiveness of the simulator training will be taken into account in our resolution of the ultimate issue in question here. We would only note, however, that the B&W simulator training does appear to be effective in that the simulator training received by the Crystal River operators after TMI-2 was credited as being very beneficial in combating the February 26, 1980, Crystal River - 3 incident. Staff Exhibit 4 at 5-8. The Board concurs with CEC that the ideal situation would be for SMUD to have its own simulator which would "mirror" Rancho Seco, but this is a major expenditure of 15 to 20 million dollars and definitely not required by the regulations. CEC Proposed Findings at 180. We agree with CEC that this matter is worthy of further evaluation and would urge SMUD to continue to explore the possibility.

138. Special training was provided to the Rancho Seco operators after the TMI accident to ensure that Items (d) and (e) of the short-term actions, and Item (d) of the long-term modifications set forth in the May 7 Order would be fulfilled. These training requirements were met prior to the restart of the Rancho Seco plant on July 5, 1979. Staff Evaluation at 25, 26; Rodriguez Testimony at 15; Testimony of Robert A. Capra on Implemen-

[12]The only NRC requirement relating to simulatory training is set forth in 10 C.F.R. Part 55, App. A, para. 4(d):

> Simulation of emergency or abnormal conditions ... may be accomplished by using the control panel of the facility involved or by using a simulator .... If a simulator is used ... [it] shall accurately reproduce the operating characteristics of the facility involved and the arrangements of the instrumentation and controls of the simulator shall closely parallel that of the facility involved.

The uncontroverted evidence in this proceeding indicates that these requirements have been met with respect to the use of the B&W simulator.

tation of Long-Term Modifications Established in the Commission Order of May 7, 1979 following Tr. 1163 ("Capra") at 5, 6.

139. Post-TMI training of Rancho Seco operators has been addressed by the Board. In addition to the special B&W post-TMi simulator training, this included further training by the Rancho Seco training staff and by General Physics Corporation, a consultant to SMUD on the sequence of events and causes of the TMI accident, procedure changes made to reflect the lessons learned from the TMI accident, requirements of NRC I&E Bulletins, plant modifications made as a result of the TMI accident, small-break LOCAs, void formation theory, saturated and subcooling operations curves, initiation and recognition of natural circulation, safety features actuation system operation, auxiliary feedwater system operation, control of the reactor trip relay, clarification of technical specifications, and requirements for notification of the NRC. Rodriguez Testimony at 16-18 and Appendix III; Wilson on Operator Training at 5. As we found earlier, each licensed operator was tested on this training by Licensee and audited by the NRC Staff.

140. CEC contends that this special post-TMI training was superficial and not extensive enough given the wide variety of complex subjects. CEC Proposed Findings, para. 155-157. They further assert that the 27 hours of training and testing under this program was not a "substantial addition to the existing training program." *Id.* at para. 157. We again, however, note that it is not our mandate to find whether this special post-TMI training would be a "substantial addition" to the Licensee's training program because we are reviewing the adequacy and effectiveness of the overall training program. Consequently, we need not decide whether this special training substantially complemented the Licensee's existing program. It was implemented to ensure compliance with the requirements of the May 7 Order relating to operating instructions in the event of small-break LOCAs and further simulator training. Simply the Licensee instituted a training program to ensure that post-TMI information was adequately understood by the licensed operators. Wilson on Operator Training at 5. These operators were audited by NRC personnel who concluded that adequate understanding was demonstrated. *Id.* at 7. We find that the special training has proven adequate for its intended purpose—*i.e.,* operators understood the proper actions to be taken in the event of a small-break LOCA.

141. In summary, the Licensee offers a training program that is diverse and comprehensive. It is similar in scope, amount, and type of training to general industry practice, including the training given to the TMI operators. Tr. 3811-3812. We find that the various training programs described above can provide the tools by which operators, management, and other personnel can effectively learn about the fundamental aspects of the facility to ensure

that it is operated in a safe manner. However, as CEC correctly points out, the quantity, scope or type of training is not the total picture. CEC Proposed Finding, para. 162; Tr. 3610-3611 (Bridenbaugh). The effectiveness of the training program is proven by management and operating personnel demonstrating that they know how to and, in fact, can operate Rancho Seco in a safe manner.

2. Understanding of Nuclear Technology and Facility Operation

142. The initial proof of an adequate and effective overall training program is whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of plant operation. As we have found, the cold and hot license training programs, and the ongoing requalification program, include instruction in the fundamentals of nuclear technology, and the theory and principles of plant operation. As a part of these training programs, the operators are examined to assess their understanding of nuclear technology fundamentals. Rodriguez at 23.

143. CEC, however, has questioned whether the training is adequate and has observed that there is ... "no assurance that SMUD operators have an analytical understanding significantly better than that of TMI operators." Prepared Direct testimony of Dale G. Bridenbaugh and Gregory C. Minor Concerning Operator Training and Human Factors Engineering fol. Tr. 3496 ("Bridenbaugh-Minor") at 9.[13] In response to this doubt regarding the ability of operators to demonstrate adequate understanding of nuclear technology and the fundamental aspects of facility operation, the Board and parties conducted a thorough investigation into the sufficiency of that understanding.

144. CEC took the depositions of three operators. The Licensee made seven of the 16 operators available for this examination, from which CEC selected at random a shift supervisor, a senior operator, and an operator. CEC Proposed Findings, para. 27 at 98. These depositions are in the record of this proceeding as CEC Ex. 36, 37 and 38. Based on these depositions, CEC concluded that the shift supervisor displayed a thorough understanding of the plant and its operation, the senior operator a somewhat less complete understanding, and the operator an inadequate understanding. CEC Proposed Findings, para. 169. Some of the facts pointed to by CEC which they assert evidenced a lack of knowledge or inadequate understanding are the following CEC Proposed Findings, paras. 170-172:

[13]While the testimony was jointly sponsored, the Board and the parties were directed to Mr. Bridenbaugh for the testimony on operator training and to Mr. Minor for the testimony on human factors engineering. Tr. 3498.

- The operator gave an incorrect response regarding feed and bleed cooling. CEC Ex. 38 at 18-19.
- the senior operator did not know the OTSG boil dry time. CEC Ex. 36 at 16.
- The senior operator could not recall what his mathematics of dynamic systems course was about. CEC Ex. 36 at 99.
- The senior operator could not recall the substance of his hot licensing training on brittle fracture of the reactor vessel. CEC Ex. 36 at 89.

145. Mr. Bridenbaugh's conclusion noted above comparing Rancho Seco operators with TMI operators also relies, in part, on his review of the operator depositions. Bridenbaugh-Minor at 7, 8; Tr. 3505. He appears also to have based his conclusion on the similarity of training at the two facilities largely on a comparison of the number of hours devoted to classroom and simulatory training. Tr. 3568-3570. He conceded, however, that there could be differences in the quality and content of the training conducted which would not be revealed by just looking at the training programs. Tr. 3537, 3538. Finally, in response to a Board question, he stated (Tr. 3610, 3611):

> I guess I would say that I haven't had the opportunity to make any extensive qualitative analysis of the two programs. I think in order to do that, for example, you would find it necessary to do many of the things that the NRC does, and that is to — or should be doing, at any rate, and that is to sit in on training programs and observe them in operation.
>
> I did not have the opportunity to do that. I think your point is a very valid one, though, and that is, you know, that — comparing absolute hours is not necessarily, you know, a total picture of things ....

146. Mr. Bridenbaugh's comparison also appears to rest upon his finding that Licensee's training program complies with only the letter of existing requirements, and that since the TMI program met the same regulations, and they were both approved by the NRC, there is no substantial difference in the training provided to the operators at the two facilities. Bridenbaugh-Minor at 6; Tr. 3534. The witness went so far as to apply this reasoning to most, if not all, of the utilities with licensed reactors. Tr. 3534.

147. Based on our review of the foregoing and the evidence in this proceeding, we conclude that the record supports a finding that the Licensee's operators do have sufficient technical and analytical knowledge to properly understand the fundamental aspects of nuclear technology and facility operation. First of all, the operators' depositions noted above do not

represent a comprehensive inquiry into their understanding and knowledge and also represent a limited sample of all SMUD operators. These depositions had relatively limited substantive questioning into these matters. A considerable portion of each deposition was devoted to matters such as description of the facility, operator experiences with various transients, equipment availability, descriptions of the SMUD organization, and other matters not germane to the operators' training and knowledge. In addition, this Board is reluctant to give much evidentiary weight to depositions because of our inability to observe the witness's demeanor.[14] We would expect that an operator, not used to giving answers under oath in a legal setting, may not give his or her best answers or have total recall of knowledge.[15] Thus, these depositions do not provide convincing evidence to either support or refute CEC's assertion that the SMUD operators do not possess the requisite technical and analytical knowledge to properly understand the fundamental aspects of nuclear technology and facility question.[16]

148. An attempt was made to elicit from witnesses a comparison between the SMUD training and general industry practice and in particular with TMI training. No witness had made a detailed comparison of the training program. SMUD in rebuttal paragraph 52 stated that it could not present testimony on a detailed comparison without presenting evidence on a multitude of other training programs. This it did not believe was the question before the Board. We agree. Contrary to CEC proposed finding 161, Staff witness Wilson did not testify that Licensee's (presumably current) operator training program is similar in scope, amount, and type of training to general industry practice. Rather, he testified that there were no *substantial* differences. Tr. 3810. Later he compared the SMUD and TMI training programs *prior to the TMI-2 accident,* and testified that while he *assumed* they were fairly similar, he had not made a detailed comparison. Tr. 3811. In general, the training experience would be more dependent on the quality of instruction and the competence of the operators than on a

[14]CEC did not request that the SMUD operators be made available or subpoenaed for the hearing.

[15]This may well explain the incorrect response regarding feed and bleed cooling by the operator since he should have had correct knowledge of this operation.

[16]CEC in its Proposed Findings (para. 173) sought a finding from this Board that:

> [W]hile the depositions do not lead us to conclude that operators at Rancho Seco are less competent than at other facilities, the depositions likewise do not persuade us that their training is *superior* or that they are *more competent* than operators at other facilities. (emphasis added)

We do not adopt this finding because we believe the proposed standard for comparison is inappropriate. It is not supported by the record and not required by any NRC regulations.

strict comparison of the content of the training programs. Second, even though all training programs must meet existing NRC requirements, it does not follow that all the programs are identical. In fact, the record shows that SMUD's program goes beyond existing requirements with respect to (a) simulator training in the requalification program, (Rodriguez at Tr. 3230; Bridenbaugh at Tr. 3524); (b) content of the written requalification examination (Wilson at Tr. 3824), and (c) annual oral requalification examination administered by SMUD management (Rodriguez at Tr. 3448). Thus, we believe that a simple comparison of various training programs does not contribute to the resolution of the issue in question here. It is abundantly clear to us that the proof of adequate and effective training is that the operators *demonstrate* sufficient technical and analytical knowledge and understanding of the fundamental aspects of nuclear technology and facility operation. This knowledge and understanding is audited and observed by the NRC, pursuant to the regulations set forth in 10 C.F.R. Part 55, and the NRC"s operator examination is designed to test an applicant's understanding of the facility design and his familiarity with the controls and operating procedures of the facility. *See* 10 C.F.R., Section 55.20.

149. Staff Witness Wilson from the NRC's Operator Licensing Branch has personally audited Rancho Seco operators and observed them in training. Wilson on Operator Training at 2; Tr. 3821, 3822. Mr. Wilson testified that on the basis of the tests the NRC has conducted and the requalification training which he has witnessed personally, the Rancho Seco operators adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation. Wilson on Operator Training at 7; Tr. 3827. Based on this testimony which we have found reliable and uncontroverted, we conclude that the Rancho Seco operators have demonstrated sufficient technical and analytical knowledge and understanding of the fundamental aspects of nuclear technology and facility operation.

3. New Information

150. CEC Issue 3-2 questions whether Rancho Seco operators are properly apprised of relevant new information, including operating experience at other reactors. Licensee receives new information relevant to the safe operation of Rancho Seco from vendors, the NRC, and from the plant's own operating experience. In the case of significant operating events

at Rancho Seco, reports prepared for submission to the NRC, if pertinent, are provided to operating crews through the Special Order program.[17] As a result of screening by Rancho Seco management, this information may also be reflected in revisions to operating procedures or communicated in memoranda for reading and information. The periodic issue of licensee event reports (LERs) by the NRC is distributed to the Rancho Seco Plant Superintendent and Operations Supervisor. Experiences at other units which are deemed by Rancho Seco management to be directly pertinent to plant operation can then be communicated to the operators through the Special Order program or through short lectures by the Operations Supervisor. In addition, B&W produces a weekly summary of occurrences at B&W reactors, which is provided to Rancho Seco operating crews. Finally, the requalification training program, including B&W simulator training, is used to acquaint operators with operating experience at other plants. Rodriguez at 34, 35.

151. The nuclear industry and the NRC have both undertaken additional efforts, since the Three Mile Island accident, to improve the dissemination and use of nuclear power plant operational data. The Electric Power Research Institute's Nuclear Safety Analysis Center is developing a capability to review systematically available plant event reports and data for transmission to applicable licensees.[18] The new Institute for Nuclear Power Operations (INPO) is also to review and analyze operating experience and relay this information to licensees for incorporation into their training programs. Wilson on Operator Training at 10; Rodriguez at 35. In addition, the Commission has established an agency-wide Operational Data Analysis and Evaluation Office to provide coordination and an overview of all operational data analysis activities performed within the NRC.[19] Wilson on Operator Training at 9.

152. CEC Witness Bridenbaugh's written testimony on this subject emphasizes what he feels is the absence of a procedure requiring that pertinent new information is communicated to operating crews in a manner to make sure that it is understood. Bridenbaugh-Minor at 9, 10.

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[17]The Special Order procedure requires that each shift supervisor discuss with his operating crew the content of each order issued. The shift supervisor must document that this discussion was accomplished. Rodriguez Testimony at 32; Tr. 3402.

[18]EPRI provided Rancho Seco with relevant information on the February 26, 1980 event at Crystal River-3, another plant with a B&W NSSS, which was then communicated to the Rancho Seco operators. Tr. 3300, 3301.

[19]The Board also takes notice of the fact that since the accident at Three Mile Island the Commission has: (1) issued an advanced notice of proposed rulemaking to consider requiring mandatory participation of power reactor licensees in the Nuclear Plant Reliability Data System, 45 FR 6793 (1980); and, (2) amended its regulations to require timely and accurate information from licensees to NRC following significant events at operating nuclear power reactors, 45 FR 13434 (1980).

153. This is a reasonable objective, but the evidence indicates that such means already exist and that improvements are being made. Excessive formality and rigid criteria are not necessarily advantageous here. Licensee's Manager of Nuclear Operations testified that the facility management staff does not want to overload operators with information which is not new or does not add to their understanding of plant operation. Tr. 3305. Careful screening of new information by the Manager of Nuclear Operations, the Plant Superintendent and the Operations Supervisor—who are personally aware of the operators' needs for information, yet sensitive to the overall burdens on the operators—is in our view superior to the establishment of rigid criteria for the communication of new information. *See* Tr. 3446, 3447. The Board finds, then, on CEC Issue 3-2, that Rancho Seco personnel are properly apprised of new information pertinent to the facility's safe operation and ability to respond to transients, and particularly of information on the operating experience of other reactors.

4. Emergency Procedures

154. CEC Issue 3-3 questions whether emergency procedures at Rancho Seco are understood by and available to plant personnel so that they will be implemented effectively in an emergency. Licensee maintains emergency procedures at Rancho Seco in a single volume red binder, distinct from other plant procedures, one copy of which is located in a desk immediately behind the control console in the control room. Consequently, the emergency procedures are available in a manner that allows quick and effective implementation during an emergency. Rodriguez at 32; Wilson on Operator Training at 12.

155. Administrative procedures are in place to ensure that the emergency procedures are kept up-to-date. Wilson on Operator Training at 12. Licensee has changed a number of its emergency procedures since the Three Mile Island accident and made what the Staff believes to be significant improvements to them in response to the Commission's Order of May 7, 1979. *Id.* at 15; Tr. 3850, 3851. Changes to the emergency procedures normally are communicated to operating personnel through the Special Order program. Each licensed operator must review an emergency procedure change and document completion of that review. Rodriguez at 32. The emergency procedures are also the subject of training in the requalification program, where operators practice the procedures during simulator training and are selectively tested on them in the annual oral and written examinations. *Id.* at 33; Wilson on Operator Training at 12, 14. Through its examination process, the NRC Staff also determines whether emergency procedures are understood by licensed personnel. Wilson on Operator Training at 13; Tr. 3840-3845.

156. With respect to emergency procedures, Mr. Bridenbaugh asserted that there may be confusion on the part of operators as to whether emergency procedures and immediate action steps are required to be memorized and on the use of written procedures. He observed that operators need to memorize the immediate action steps. Bridenbaugh-Minor at 10, 11; Tr. 3561, 3562. The record shows, however, that the operators do commit the immediate action steps to memory as Mr. Bridenbaugh suggests. After those steps are taken they refer to the written emergency procedures to determine the subsequent actions which should be taken and to verify accomplishment of all of the immediate actions. Tr. 3443; Wilson on Operator Training at 12; Tr. 3842. CEC is of the opinion that these written emergency procedures do not provide adequate guidance to the operators as to the symptoms to look for, proper operator actions in the event of an emergency, or proper verification procedures.[20] CEC Proposed Findings, para. 210, 211. CEC's opinion in this respect is based on a perusal of CEC Ex. 46, the latest SMUD revision of Emergency Procedure D.5 entitled "Loss of Reactor Coolant/Reactor Coolant System Pressure." It is not based on any examination of the operators. The NRC Staff, on the other hand, has conducted an examination of the Rancho Seco operators with regard to emergency procedures. On the basis of that examination, it is satisified that licensed personnel understand the emergency procedures. Wilson on Operator Training at 13. The Board finds, on CEC Issue 3-3, that the NRC and SMUD adequately assure that emergency instructions are understood by and are available to plant personnel in a manner that allows quick and effective implementation during an emergency. We would also point out that improvements are being made in this area. The B&W licensees have undertaken, through their Owners' Group, the development of abnormal transient operating guidelines (ATOG) in response to the concern that demands on operators are becoming excessive as a result of new requirements. The objective of the program is to simplify the operators' problem of identifying and treating abnormal transients. The guidelines will enable each B&W facility to develop a standardized set of abnormal and emergency procedures and will be based on input data from each of the facilities and transient analyses performed by B&W. The NRC Staff endorses this effort. Staff Ex. 4 at 5-74.

5. Operator Testing

157. FOE Contention III(e) and part of Board Question H-C 32 question whether there are adequate procedures to determine and test the

[20]Verification in this context means ensuring that an automatic action has occurred or manually performing it, if necessary. Staff Ex. 4 at 5-73.

competence of Rancho Seco operators. Individuals who manipulate the controls of a nuclear reactor must first be licensed by the Commission (10 C.F.R. Section 55.3), and the issuance of such a license requires the successful completion of an initial licensing examination administered by the NRC Staff. Wilson on Operator Training at 18. There are 24 licensed personnel (18 senior operators[21] and 6 operators) on the operating staff at Rancho Seco. Tr. 3047, 3048.

158. Through the licensing requirements and training directed by the Commission in its regulations, the additional requirements imposed in the Order of May 7, 1979, and the Staff's verification of compliance with these requirements, this agency has regulated the competence of licensed operators. The cold license, hot license, and requalification training programs at Rancho Seco have been reviewed and approved by the NRC. Wilson on Operator Training at 3. CEC Witness Bridenbaugh is the only witness who questioned the competency of the Rancho Seco operators, although his conclusion merely was that he is not sure they are *better* than the operators who were at TMI. In the foregoing findings of fact we have concluded that Mr. Bridenbaugh's testimony on that score does not contribute to a determination on the issue of whether operators demonstrate sufficient technical and analytical knowledge of facility operation. Further, Mr. Bridenbaugh does not question that Licensee's training of its licensed operators complies with NRC requirements. Rather, he asserts that the NRC standards are inadequate. Bridenbaugh-Minor at 7, 11 and 12; Tr. 3520, 3570. To the extent that Mr. Bridenbaugh finds Commission regulations to be in need of revision, his complaint must be taken to the Commission and cannot be entertained by this Board. *See* 10 C.F.R. Section 2.758.

159. It is significant that Mr. Bridenbaugh's written testimony did not present, and that on cross-examination he could not recall, a specific instance in which a Rancho Seco operator has demonstrated a lack of understanding of what would have to be done to respond to a feedwater transient. *See, e.g.*, Tr. 3586, 3587. NRC Staff Witness Wilson, who has audited and given licensing examinations to Rancho Seco operators, observed requalification training of Rancho Seco operators, and examined hundreds of operators at other plants over a period of six and one-half years, testified that the Rancho Seco operators stack up very favorably with other operators in training programs with which he has experience. Tr. 3878-3881. The Board finds, on the basis of the foregoing findings of fact and in partial response to Board Question H-C 34 and FOE Contention III

[21] A "senior operator" is any individual designated by a facility license under 10 C.F.R. Part 50 to direct the activities of licensed operators. 10 C.F.R. Section 55.4(e).

(e), that the training actions required by the Commission's Order of May 7, 1979, in the context of the other training provided and the testing required, provide reasonable assurance that the licensed operators at Rancho Seco will operate the plant safely in response to feedwater transients.

6. Unlicensed Operators

160. The remainder of Board Question H-C 34 questions whether unlicensed operating personnel will respond adequately to feedwater transients. Unlicensed operators at Rancho Seco assist the licensed operators by starting and stopping motorized equipment, opening and shutting valves, conducting periodic maintenance or checking of equipment, and maintaining plant records. These various activities are directed and supervised by the licensed operators who assist the unlicensed personnel if necessary. Written procedures are located at equipment operating stations to instruct these personnel in their assigned tasks. Unlicensed personnel are allowed to manipulate apparatus and mechanisms which may affect reactivity and the power level of a nuclear power plant only under the direct supervision of a licensed operator present at the controls and only for purposes of training such individuals to obtain necessary experience to become licensed.[22] NRC Staff Testimony of Philip J. Morrill on Training of Unlicensed Operators following Tr. 4141 ("Morrill") at 3.

161. The unlicensed operations personnel are placed in one of three categories according to their experience and competence. The least experienced personnel are "power plant helpers" who are initially assigned to receive on the job training from more experienced personnel and to do odd jobs around the plant. As these personnel become more knowledgeable and experienced, they are assigned greater responsibility for equipment operation by the senior licensed operator on that shift. After approximately a year, a power plant helper may become an "Equipment Attendant" who is generally responsible for working with equipment in the non-safety related portions of the plant. After an additional time of about one year, the unlicensed person may become an "Auxiliary Operator" who usually operates equipment in safety related areas of the plant. These assignments are generally on the basis of seniority, performance (as evaluated by Rancho Seco management) and availability of that job position. Normally, there are between 3 and 7 unlicensed operations personnel on a shift depending on what plant operations are planned. *Id.* at 3, 4.

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[22] See 10 C.F.R. Section 55.9(b).

162. The role of unlicensed operators, however, is minimal in operating the Rancho Seco plant safely in response to a feedwater transient. The auxiliary feedwater system, required in the event of a loss of main feedwater, can be operated from the Rancho Seco control room by licensed operators. In the event that the 24-hour water supply for the auxiliary feedwater system in the condensate storage tank reached a low level, unlicensed operators might be called upon to operate manual valves outside the control room to align the off-site water supply to the auxiliary feedwater pumps. Unlicensed operators have been given training, since the Three Mile Island accident, to perform this manual valving. Each shift supervisor has conducted specific training for the unlicensed operators of his crew, including a "walk through" to affirm valve location and operation, to assure that they can locate and reposition the valves in the unlikely event they are directed to do so to assure an adequate supply of auxiliary feedwater. Unlicensed operators have also been instructed on the proper procedure for taking local control of the auxiliary feedwater system control valve to each steam generator in the unlikely event of a loss of control to all four of the available auxiliary feedwater level control valves. Rodriguez at 37, 38; Morrill at 5; Tr. 4224, 4225.

163. CEC Witness Bridenbaugh testified that under "on-the-job" training, unlicensed operators may not know how or where to perform certain actions the first time they are called upon to perform them, citing the deposition of Rancho Seco licensed senior operator Tipton. CEC Ex. 36 at 113, 114; Bridenbaugh-Minor at 13. Mr. Rodriguez also acknowledged that unlicensed operators may be asked by a licensed operator to perform an operation for which they had never been trained and which they had never before performed. Tr. 3118. Mr. Tipton, however, did not testify that unlicensed operators may not know how or where to perform certain actions the first time they are called upon to perform them. To the contrary, he testified that "...they are instructed either the first time they have to do a task or again if they need refresher." CEC Ex. 36 at 113, 114. CEC believes that this mode of operation is unacceptable and that these personnel should be instructed and "checked off" on these operations before standing shift. *See* CEC Proposed Finding, paras. 200-208.

164. The Board finds that CEC's proposal is not supported by the evidence and, in fact, is an unreasonable method of training unlicensed operators. The evidence indicates that unlicensed operator activities are directed and supervised by the licensed operators, who assist if necessary, and the "on-the-job" training lasts several years. Presumably, if an unlicensed operator is asked to do an activity which he has never done before or received training on, the licensed operator will supervise and

assist. This, to us, is the essence of "on-the-job" training and is an excellent way to gain the necessary experience to become licensed.

The record does not indicate any instances where this type of training has been abused or resulted in the unsafe operation of the facility.[23]

165. Based on the foregoing and in response to Board Question H-C 34 the Board finds that there is reasonable assurance that unlicensed operating personnel are sufficiently trained and will respond adequately to feedwater transients.

7. Management and Technical Personnel

166. FOE Contention III(d) and part of Board Question H-C 32 question whether there are adequate procedures to determine and test the competence of Rancho Seco facility management. The Commission reviews the technical qualifications of applicants for facility operating licenses to engage in the proposed activities in accordance with Commission regulations. *See* 10 C.F.R. Section 50.40(b). The NRC reviews the management and technical organization of the applicant and its technical contractors to assure that on-site facility management and personnel are qualified to act responsibly and competently in the event of an emergency or abnormal occurrence at the plant, to assure that clear management control and effective lines of communication exist between the organizational units involved in the management, operation, and technical support for the operation of the facility, and to assure that the applicant has the necessary technical support for the operation of the facility. Testimony of Frederick R. Allenspach Relating to Management and Technical Competence fol. Tr. 3920 ("Allenspach") at 2-5.

167. SMUD's organizational structure, personnel requirements and technical qualifications were reviewed and found to be acceptable in a Safety Evaluation, dated June 8, 1978, issued as part of the NRC Staff's operating license review for Rancho Seco.[24] *Id.* at 5,6.

168. The key members of Rancho Seco management—the Manager of Nuclear Operations, Plant Superintendent, Engineering and Quality Con-

[23]CEC appears to allege that the high turnover of unlicensed personnel at Rancho Seco during the period October 1978 to October 1979 (10-12 individuals terminated employment) resulted in Rancho Seco being operated by relatively inexperienced personnel. *See* CEC Proposed Findings, para. 206-207. As a result of this turnover and an anonymous allegation that training of unlicensed personnel was inadequate, NRC I&E inspectors interviewed approximately 50% of the unlicensed personnel. Morrill at 4. The investigation disclosed three relatively minor complaints regarding inadequate training and procedures and the Licensee took corrective action. *Id.* at 8. With these actions, the NRC was satisfied that the unlicensed personnel were adequately trained. *Id.* at 9.

[24]The NRC is considering upgraded requirements in the area of management and technical capabilities for licensees of operating reactors. Allenspach at 7-9.

rol Supervisor, Chairman of the Plant Review Committee, and Operations Supervisor—all have senior operator licenses issued by the NRC. They each participated in the Rancho Seco cold license training program, in the special post-TMI training, and continue to participate in the requalification training program. Consequently, as licensed senior operators, these facility management personnel are regularly trained and tested on their knowledge and competence to operate the plant safely. Rodriguez at 19, 20.

169. In addition, Rancho Seco management and supervisory personnel have begun participation in a command and control training program, being presented by a consultant to the District, which will provide further training in the command and control aspects of mitigating various accidents. Rodriguez at 20, 21; Tr. 3385, 3386.

170. Three reactor inspectors from Region V of the NRC's Office of Inspection and Enforcement (Allen D. Johnson, Gerald B. Zwetzig, and Harvey L. Canter) testified on the competence of SMUD to operate the Rancho Seco facility. All of them, in one capacity or another, have reviewed and inspected the operations at Rancho Seco. Mr. Canter has been the NRC's Senior Resident Inspector at the Rancho Seco site since August 1, 1979. Based on personal knowledge and experienced judgment of the SMUD operation, and relying to a large extent on the number of items of noncompliance and reportable occurrences at Rancho Seco since it commenced commercial operation, these inspectors concluded that the "SMUD organization and personnel are competent to safely operate the Rancho Seco nuclear generating station." Johnson Testimony at 11, fol. Tr. 3920. *See also* Zwetzig Testimony at 6, fol. Tr. 3920; Canter Testimony at 8, fol. Tr. 3920.

171. The NRC Staff also submitted testimony by the Performance Appraisal Branch of NRC's Office of Inspection and Enforcement which had completed the major portion of a management appraisal inspection at Rancho Seco while these hearings were in session. Because preliminary findings from this inspection resulted in a number of concerns of the performance appraisal team which might be relevant to the issues before the Board, two members of the team testified on the team's preliminary findings. Supplemental Testimony of NRC Performance Appraisal Branch Regarding SMUD Management Controls, fol. Tr. 4233 ("PAB Testimony"). The performance appraisal team's preliminary concerns relate to management controls in seven of the eleven functional areas reviewed. PAB Testimony at 2.[25]

[25]The witnesses prefaced their testimony by observing that these preliminary concerns may be resolved by the subsequent inspection and review efforts of the team. PAB Testimony at 1.

172. Some of the PAB concerns are not relevant to the subject matter o this proceeding (*e.g.*, fire prevention and protection) except to the exten that they relate to the overall ability of SMUD to properly manage th facility in accordance with the technical specifications at Rancho Seco an other NRC requirements. For example, in the area of training, the PAI witnesses testified that Licensee had not fully implemented requirements i its own procedures for the training of non-licensed personnel. PAI Testimony at 3, as corrected at Tr. 4252. This observation, however, did no relate to the unlicensed operators on the Rancho Seco operating crew which the Board addressed above in its findings on Board Question H-C 34 but to maintenance and technical staff personnel. Tr. 4276, 4277. The PAI witnesses also testified that the Licensee had not fully implemented th training program for licensed operators in that for several operators, th oral exam was not administered within the time frame specified by th Rancho Seco administrative procedure. PAB Testimony at 3; Tr. 4254-56 Licensee explained that this delay was caused by a plant refueling outag which required the services of the operators. Tr. 3447-48.

173. In the area of design changes and modifications, while th testimony was not specifically related to systems at issue here, the witnesse from the Performance Appraisal Branch testified that Licensee's procedure for the review of design changes to Class I systems pursuant to 10 C.F.R Section 50.59 did not comport with the requirements of the license technica specifications in that a second level safety evaluation was not providec where the first level of review by the Supervisor of Engineering and Qualit Assurance resulted in a negative determination under 10 C.F.R. Sectior 50.59. PAB Testimony at 3; Tr. 4247. Licensee described its procedures fo such reviews and disagreed with the performance appraisal team's conclu- sions that they did not comply with the technical specifications. Tr. 3448- 3450. Mr. Johnson, an inspector from Region V of the Office of Inspectior and Enforcement, also described the procedure for such changes requirec by 10 C.F.R. Section 50.59 and the Rancho Seco technical specifications. Tr. 4118, 4119. He is familiar with Licensee's procedures for reviewing design changes to Class I systems, and testified that the position of Regior V, which has been communicated to Licensee, is that the District's procedure complies with the technical specifications.[26] Tr. 3921, 3922. In any event, inspectors from Region V and the performance appraisal tean reviewed some 176 design changes implemented pursuant to a negative

[26]This differing interpretation by the I&E regional office and its Performance Appraisal Branch apparently persists. *See* Tr. 4275. The Board recognizes, and is mindful in attaching weight to statistics on enforcement activities, that licensees may face different interpretations of a single requirement by different NRC inspectors. *See* Tr. 3971-3973, 4280.

Section 50.59 determination by the Supervisor of Engineering and Quality Assurance and concurred in all of the determinations. Tr. 4275, 4276.

174. The Performance Appraisal Branch concerns center about the Licensee's management control systems in the areas reviewed by its team. The fact that concerns exist about a Licensee's management controls does not indicate that the licensee's management is not competent to manage its reactor facility. PAB Testimony at 1, 2; Tr. 4270. While the team spent approximately one man-year of effort in its investigation of Rancho Seco (Tr. 4234), it found no weaknesses in management controls which warranted immediate action. PAB Testimony at 5; Tr. 4271, 4272. Significantly, the team found no concerns in the area of plant operations—*i.e.,* the manner of instructing operating crews, defining responsibilities, providing for communication between operations personnel and management, and providing for management knowledge of problems in the field and their resolution. Tr. 4235, 4236. However, the PAB team felt that Rancho Seco management controls were relatively poor in comparison with the other seven facilities it had inspected. Tr. 4241, 4249.

175. The record on Licensee's management competence appears, on its face, confusing and contradictory as a result of the PAB inspection. The Board and the parties, however, have examined at length Licensee's Manager of Nuclear Operations, the senior management person on the facility site. In addition, five inspectors from the NRC Office of Inspection and Enforcement have testified, providing an up-to-date and ongoing evaluation of the competence of Rancho Seco's management to operate the plant safely. No testimony has been presented that indicates that SMUD personnel are not competent to manage the facility. Although the PAB inspection has uncovered some areas of concern, we believe that many of these concerns relate to the administrative details or "paper work" of running a business. These details may have been deferred or forgotten in the face of all the management work that evolved from the TMI-2 accident, including this proceeding. While these details should not go unnoticed or uncorrected, it is our opinion that they do not constitute a serious cloud on management's ability to operate the Rancho Seco facility in a manner that reasonably assures public health and safety.

176. The PAB team testified that they did not consider immediate corrective action necessary since the deficiencies noted were not significant. Tr. 4272. They also mentioned that SMUD added additional people to its training staff in recent months, a change which should alleviate these problems. Tr. 4257. SMUD is exploring the possibility of putting its training program on a computer system as an aid in flagging due dates. Tr. 4272. It is our further opinion that those areas of concern reflected in the PAB inspection do not indicate a present safety problem that would require

Board action. We are confident that any remedial action necessary by SMUD in response to these concerns, can be effectuated through the normal functions of the Inspection and Enforcement program.

177. Returning to FOE Contention III(d) and Board Question H-C 32, we find that as licensed senior operators, Rancho Seco key management personnel are tested for knowledge and competence in operating the plant. While "tests" in the formal sense are not given on the broader subject of managerial skills and capabilities, the extensive evaluations by the NRC Office of Inspection and Enforcement have been more than adequate for reaching a determination on the competence of facility management. The Board finds that the Rancho Seco facility management is sufficiently competent to provide reasonable assurance that the plant will respond safely to feedwater transients.

J. Instrumentation

178. **Board Question CEC 5-3a:**

> Are the special features and instruments installed at Rancho Seco adequate to aid in diagnosis and control after an off-normal condition engendered by a loss-of-feedwater transient?

Board Question H-C 22:

> What instrumentation is available to give positive indication as to whether or not the coolant is subcooled throughout the core at all times? How does that instrumentation work? In the event that a non-subcooled condition is indicated, what instrumentation would then give reliable information on the water level in the core?

179. The Rancho Seco plant is designed to respond safely to a loss of feedwater transient basically by means of three systems: the integrated control system; the reactor protection system; and the auxiliary feedwater system. Information necessary for the operator to diagnose and respond to a loss of feedwater transient includes: (1) the extent of the loss of feedwater (*i.e.*, whether one or both pumps have been lost or whether control of feedwater flow has been lost); (2) whether the ICS is responding as required; (3) whether the reactor protection system has been called upon to shut the plant down; and, (4) whether the auxiliary feedwater system, if required, is functioning as designed. NRC Staff Testimony of Bruce A. Wilson on Instrumentation for Diagnosis and Control of Off-Normal Conditions, fol. Tr. 3788 ("Wilson on Instrumentation") at 3. As a result of its review of the Three Mile Island accident, the NRC Staff decided that operators should have additional indication of auxiliary feedwater flow

beyond the already available steam generator level indication. Consequently, as one of the short-term actions undertaken to upgrade the timeliness and reliability of the auxiliary feedwater system in response to the Commission's Order of May 7, 1979, Licensee installed, prior to the restart of Rancho Seco, flow meters on each of the auxiliary feedwater lines to give a more direct indication in the control room of the existence and amount of auxiliary feedwater flow to each steam generator. *Id.* at 5; Tr. 1546-1549.

180. Various other additions to and modifications of the instrumentation in the Rancho Seco control room have also been implemented since the Three Mile Island accident. These changes go beyond the requirements of the Commission's Order of May 7, 1979. Tr. 2962-2963, 3351-3356. Specifically, feedwater transient diagnostic instrumentation is available to the operators in the Rancho Seco control room to provide indication of the following parameters: auxiliary feedwater flow; reactor coolant system hot leg, cold leg and average temperature; steam generator level (six channels); steam generator outlet pressure; pressurizer level (three separate temperature compensated level indication channels); reactor coolant system makeup flow; reactor coolant pressure (four narrow range channels and three wide range channels); main feedwater flow to each steam generator; high pressure injection flow; and reactor coolant system loop flow. Licensee's Testimony of Ronald J. Rodriguez fol. Tr. 2948 as amended at Tr. 3351. (Rodriguez Testimony) at 41, 42.

181. As an additional operator aid, two saturation meters were installed in the Rancho Seco control room during the 1980 refueling outage and are now in operation. Those meters provide the operator with a continuous and direct display of the amount of subcooling present in the reactor coolant system. Previously, the operator determined this by comparison of pressure and temperature to a saturation curve. Tr. 3405; Rodriguez Testimony at 44. Each meter receives a wide range pressure signal of 0-2500 pounds from the safety features instrumentation and two hot leg temperature inputs (a range of 120-920°F; one from each reactor coolant loop). The meter itself auctioneers and selects the highest temperature reading it receives, and feeds the temperature and pressure data into a computer for a calculation of subcooling in degrees Fahrenheit. The meter displays to the operator the number of degrees Fahrenheit of subcooling. Tr. 3422, 3423 (Rodriguez); Rodriguez Testimony at 47; Testimony of Paul E. Norian on Adequacy of Pressurizer Instrumentation (Board Question 22), fol. Tr. 1163 ("Norian Instrumentation Testimony"), at 5. CEC Witness Minor, in his written testimony, recommended the installation of a saturation meter at Rancho Seco (Bridenbaugh-Minor Testimony at 16, 17 and 19) because he was unaware of the installation at the time he prepared his testimony. Tr. 3593, 3594. This arrangement does not necessarily assure that all points in the

core are subcooled, but in the opinion of Licensee's Witness Jones, any local hotspots would occasion only local boiling, and the resultant bubbles would mix with the surrounding fluid and condense. Tr. 1141. Thus, cooling of the core would be maintained as long as subcooling is indicated.

182. If, as postulated in Board Question H-C-22, a non-subcooled (saturated) condition exists in the system, there may be voids present, and pressurizer level would no longer be a good indicator of primary coolant inventory. Testimony of Paul E. Norian on Adequacy of Pressurizer Instrumentation fol. Tr. 1169 (Norian on Instrumentation) at 4; Tr. 933; Tr. 1369.

183. As a vivid example of such voiding, during the TMI-2 accident, the pressurizer PORV was stuck open and provided a leakage path for the primary system fluid. Subcooling was lost within a few minutes and the coolant began to flash. Since the leakage path was at the top of the pressurizer, there was an insurge of fluid from the hot leg which maintained a large inventory in the pressurizer. Consequently, the pressurizer level was maintained even though the primary system inventory was continuously depleted until the PORV block valve was closed. Norian on Instrumentation at 3, 4.

184. In answer, then, to the last portion of Board Question H-C-22, there is no instrumentation which gives reliable information on the water level in the core when the primary coolant is not subcooled. It is the opinion of Staff and Licensee witnesses, however, that a non-subcooled condition can reliably be avoided by proper operator action. Rodriguez Testimony at 46-77; Norian on Instrumentation at 6. Rancho Seco Emergency Procedure D5 "Loss of Reactor Coolant/Reactor System Pressure" (CEC Exhibit 46), provides specific guidance to the operator to maintain the reactor coolant in a subcooled condition in the event of a loss-of-coolant accident. *Id.* at 47. High pressure injection control from the control room allows the operator to add inventory as necessary to maintain reactor coolant system pressure and to promote adequate subcooling. *Id.* at 44. By maintaining a minimum of 50°F subcooling in the reactor coolant system and operating high pressure injection pumps to provide an indicated level in the pressurizer—as the procedures direct, and as all of the post-TMI training has taught the operator to do—void formation in the reactor coolant system will be prevented. *Id.* at 47. Because saturation conditions occur before the core becomes uncovered, (Tr. 1755), the key indication the operator needs to guide his actions is the existence of loss of subcooling. It is this condition, and not reactor vessel level, which would dictate required operator actions. Tr. 1755, 1756; Tr. 800-802. In the event conditions degrade to the point where voids are formed, the operator can monitor adequate core cooling by observing installed in-core temperature thermocouples which are located at

the top of the reactor core. Rodriguez Testimony at 47, 48; Tr. 1369, 1370; Tr. 3331.

185. As to whether additional instrumentation is desirable to permit direct detection of water level in the pressure vessel, the opinions of expert witnesses are divided. Licensee, after a review, concluded that there were no designs available for such instrumentation which would give unambiguous indications. Tr. 3332. CEC Witness Minor testified that the operator's ability to diagnose an off-normal condition involving loss-of-coolant would be enhanced by direct indication of vessel water level. But he cautioned that the complexity of such a measurement would make it necessary to carefully research the best method for making such a measurement. Prepared Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor Concerning Training and Human Factor Engineering fol. Tr. 3496 (Bridenbaugh-Minor Testimony) at 15. CEC's Witness Lewis did not advocate vessel level indicators; he stated that, in his opinion, void detectors at high points in the system would be easier to implement and probably more useful. Tr. 484. The Staff's witness merely asserted that the instrumentation to show vessel water level is needed. Norian on Instrumentation, at 5.

186. Obviously, the question of improved instrumentation for inadequate core cooling is unsettled. The important point, however, is that the existing instrumentation is sufficient for the operators to evaluate the state of the primary coolant system and initiate corrective action as needed. Any additional instrumentation to be installed would provide backup to the existing systems and provide further assurance that the core is adequately cooled. Norian on Instrumentation at 6. The Board is aware that, on October 2, 1980, the Commission published a Proposed Rule which has some bearing on this point. Interim Requirements Related to Hydrogen Control and Certain Degraded Core Conditions, 45 FR 65466. This proposed rule would require, *inter alia*, that:

> (3) By January 1, 1982, each boiling and pressurized light-water nuclear power reactor shall be provided with instrumentation *such as a reactor vessel water level indicator* which supplies to the control room a recorded, unambiguous, direct indication, of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovering and give advance warning of the approach of inadequate core cooling.

45 FR 65466 at 65473 [emphasis added].

Thus we find that, although present instrumentation is not able to indicate water level in the core in the presence of voids, the instruments meant to avoid such a condition are presently adequate, and the possibility of enhancing and improving the safety posture by developing and installing level indication or its equivalent is being pursued actively and on an appropriate schedule.

One last point should be mentioned before we leave these two Board Questions: CEC Witness Minor also suggested the need for a dedicated indication of natural circulation. Bridenbaugh-Minor Testimony at 16, 19. On examination by the Board, however, Mr. Minor testified that he did not know whether it was practical to measure such small flow rates, and that additional study would be required to ensure that his proposal would be satisfactory under various conditions. Tr. 3619. Licensee's witness testified that existing temperature instrumentation is adequate to verify natural circulation, and the Staff's witness questioned the practicality of a natural circulation meter. Tr. 3444; Tr. 3894, 3895.

187. We conclude, then, that adequate, state-of-the-art instrumentation is available to cope with a loss-of-feedwater transient at Rancho Seco; that present instrumentation and procedures can also give reasonable assurance that the coolant is subcooled; and that the development and installation of enhanced instrumentation to give vessel level indication is being pursued on a schedule appropriate to its urgency.

K. Control Room Configuration

188. **Board Question H-C 31:**

> Are there features of Rancho Seco's control room design and configuration which make it difficult for operators to avoid a loss-of-feedwater transient?

189. The thrust, and much of the substantive wording, of this issue were taken from a contention propounded by Intervenors Hursh and Castro, who withdrew. It is clear to the Board that a better statement of the issue would have used words such as "diagnose and respond to" in place of the word "avoid," since the configuration of the control room has little to do with whether or not such a transient occurs. NRC Staff Testimony of Bruce A. Wilson on Control Room Configuration, fol. Tr. 3788, (Wilson on Control Room) at 2. Fortunately, witnesses for all parties interpreted the situation correctly and addressed the control room configuration as it bears upon diagnosing and responding to the transient.

190. Upon review of the evidence, we conclude that there are no serious design shortcomings in the Rancho Seco control room. The Rancho Seco control room has compact control consoles which allow operating personnel quick access to controllers for a wide variety of equipment. The overall control room layout minimizes the amount of movement the operator must

make in taking actions involving multiple pumps and valves. Rodriguez Testimony at 40. The compact character of the Rancho Seco control room and the relatively small number of displays makes it substantially different from the TMI control room. On the whole, it is better than TMI's, especially during normal operation. Minor and Bridenbaugh Testimony at 17-18. Indeed, NRC Staff Witness Wilson concludes that the Rancho Seco control room is one of the best. Wilson on Control Room at 4-5.

191. The foregoing findings do not mean that the Rancho Seco control room could not be improved. In 1976, the Electric Power Research Institute (EPRI) published a study comparing the Rancho Seco control room with four others from a human engineering standpoint. "Human Factors Review of Nuclear Power Plant Control Room Design," identified as CEC Ex. 33. The EPRI study identified several human engineering weaknesses with the Rancho Seco control room design. The more substantial problems were:

a. Functionally related control consoles are separated. CEC Ex. 33 at 4-9.

b. Control board is unwieldy. *Id.* at 4-13.

c. Controls and instrumentation are located in areas outside the primary control room area or outside the operators' line of vision. *Id.* at 4-14.

d. Rod monitor display is poorly placed with respect to the reactor control panel. *Id.* 5-36 and Fig. 5-45.

e. Auxiliary feedwater controls are not grouped with main feedwater controls. *Id.* 5-42 and Fig. 5-43.

f. A "B" switch on the functionally grouped safeguards panel is located on the "A" panel. *Id.* at 5-42 and Fig. 5-54.

g. There is no differentiation in appearance between some switch lights and indicator lights on the safeguards panel. *Id.* at 7-9 and Fig. 7-16.

192. SMUD has not made any attempt to change the control room configuration in the time since CEC Ex. 33 was produced. Tr. 2975. The Board believes that in light of the design sensitivities of the B&W system, SMUD should consider eliminating these identified weaknesses, even though the Rancho Seco control room may have fewer defects than most. The Board is confident this will occur, since SMUD has recently contracted for a human factors study of its control room to be undertaken this year. Tr. 2974. Such a study comes at a particularly opportune time, as the study will

evaluate the new instrumentation incorporated at Rancho Seco since the TMI accident.

L. Timetable for Long-Term Modifications

193. **Board Question FOE III(c):**

> The NRC Orders in issue do not reasonably assure adequate safety because there is no reasonable time for implementation of the long-term modifications established in the Commission orders.

The May 7 Order provides that SMUD shall "as promptly as practicable" accomplish the long-term modifications set forth in that Order. May 7 Order at 8. It is the judgment of both the Commission and Licensee that these modifications, unlike the short-term actions, were not required immediately to provide reasonable assurance that Rancho Seco would respond safely to feedwater transients. Dieterich Testimony at 25; Capra Testimony at 8. No witness who appeared at this hearing testified that this judgment was in error. And, while the Commission directed the prompt implementation of the long-term modifications, some of them require detailed engineering analysis and assessment by Licensee, review by the NRC Staff, procurement of components and equipment (some of which may require extensive lead time), installation of equipment and additional training of operators. Capra Testimony at 7. Consequently, these modifications could not reasonably have been subjected to a specified, rigid implementation schedule dictated by the Commission on May 7, 1979. Licensee and the NRC Staff, however, both presented testimony on the status of implementation of the long-term modifications in response to FOE Contention III(c).

194. The testimony presented by both the NRC Staff and Licensee with respect to the long-term modification schedule is uncontroverted. This evidence indicates that the first, second and fourth required long-term items have been completed by the Licensee. Capra Testimony at 3-6. The third long-term modification required Licensee to upgrade to safety grade the anticipatory reactor trip upon loss of main feedwater and/or turbine trip. The NRC Staff approved Licensee's preliminary design for the proposed upgrade on December 20, 1979, which allowed Licensee to proceed toward installation.

195. Based on the foregoing, we find that FOE Contention III(c) has no merit. The evidence indicates that three of the four required long-term modifications have already been implemented and that the remaining modification will be completed in a reasonable time.

196. CEC would have us find that the timetable for long-term modifications is not adequate (CEC Proposed Findings paras 262, 263), in that a proper AFW reliability study and a proper FMEA of the ICS system have not been prepared. We have found that these studies have been adequate for the purposes intended, and we therefore reject the contrary finding proposed by CEC.

M. Hydrogen Control

197. **Board Question H-C 20:**

Does Rancho Seco's present system for coping with hydrogen release in containment provide for:

a. recombiner availability early enough to respond to a situation like that at TMI-2?

b. proper radiological protection of the surroundings if purging is depended upon?

The Board retained (and paraphrased) this question after Intervenors Hursh and Castro withdrew because we were aware that experience at TMI-2 suggested a pressure rise had occurred as a consequence of the generation and combustion of hydrogen in the course of the accident.

198. There are two methods available for removal of hydrogen from a containment building: a purge system and a recombiner. Rancho Seco has a purge system but does not have a recombiner. Licensee's Supplemental Testimony of Robert A. Dieterich In Response to Board Question HC-20, fol. Tr. 1988, at 21; NRC Staff Testimony of Thomas A. Greene on Hydrogen Recombiner, fol. Tr. 2783, (Green Hydrogen Testimony) at 2. After a severe accident, the purge system could not be used for approximately 13 or 14 days because earlier use would lead to large radioactive releases to the environment. Dieterich Testimony at 20; Tr. 2843.

199. A recombiner may be used earlier in an accident sequence than a purge system because the recombiner vents back into the containment building rather than releasing radioactivity to the environment. Dieterich Testimony at 20; Tr. 2842-44. The NRC has recognized the advantage of hydrogen recombiners by requiring them, rather than purge systems, on newer facilities. 10 C.F.R. Section 50.44(g). Rancho Seco's purge system is designed to accommodate the hydrogen generated by a design basis accident in which five percent of the zirconium cladding present in the core reacts with steam to produce hydrogen (Tr. 2156-57) and as required by 10 C.F.R. Section 50.44(d)1.

200. Possession of a hydrogen recombiner was not required by the regulations for plants licensed at the time of and with the characteristics of Rancho Seco. 10 C.F.R. Section 50.44(g); Greene Hydrogen Testimony at 6. Thus Rancho Seco met all the requirements for hydrogen (or "combustible gas") control applicable at the time it was licensed. Indeed, even though the possession of a recombiner was not required at that time, Licensee took the additional precautionary measure of contracting with Arizona Public Service Company (APS) to obtain hydrogen recombiners on a loan basis from APS. Dieterich Testimony at 21; Tr. 2152; Tr. 2848. The APS recombiners could be delivered in approximately 24 hours. Dieterich Testimony at 22.

201. However, the Board feels it is necessary that we consider the time sequence for generation of hydrogen and the amount generated in the light of the accident at TMI-2. During that accident a substantial amount of cladding reacted early in the sequence to produce hydrogen, and the reaction involved something of the order of 30 percent of the zircalloy clad. Tr. 2885. Hydrogen was generated at a rate approximately five hundred times the rate at which present day recombiners can recombine it. Dieterich Testimony at 22; Greene Hydrogen Testimony at 4-5. Tr. 2352-53, 2363; Tr. 2844, 2855, 2886. Further, the buildup of a combustible level of hydrogen occurred in the first few hours. Greene Hydrogen Testimony at 4-5. Thus, the use of a recombiner (or a purge system) would have been of no value.

202. The Commission is aware of the anomaly presented here. Indeed, the Commission has stated:

> The accident at Three Mile Island, Unit 2, resulted in a severely damaged or degraded reactor core with the ... generation of hydrogen from fuel cladding-water reaction well in excess of the amounts required to be assumed for design purposes by the current Commission regulations. (Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations 45 FR 65466).

The "current regulations" referred to are, of course, precisely those to which Rancho Seco conforms.

203. Specifically then, we must find in answer to Board Question H-C 20 that: (a) no recombiner is available which would be capable of coping with a situation like that at TMI-2 and (b) purging could not be safely accomplished at a time early enough to cope with that situation.

204. CEC would have us require the Licensee to install "one or more hydrogen recombiner systems" on the ground that such devices might have

some value. CEC Proposed Findings at 146. We see no support in the record for that notion.

205. Licensee, on the other hand, assures us that "even the amount of hydrogen generated in an accident of the severity of TMI-2 would not result in a dangerous challenge to the Rancho Seco containment" (Licensee's Proposed Findings at 174, footnote omitted, citing Dieterich Testimony at 23) and would have us further believe that even complete reaction of the zircalloy would not generate enough hydrogen to hazard the containment. *Id.* We agree with the first proposition, but we find the statements in the record as to the second situation (*viz:* total clad reaction) to be conclusory and poorly supported. The witness in fact stated that he had not actually done such a calculation. Tr. 2177.

206. We thus find that although Rancho Seco is not protected by recombiners or purging against generation of hydrogen in amounts like those generated at TMI-2, it could withstand combustion of such amounts of hydrogen. As to the possibility that larger amounts might be generated, we note that the Commission's Proposed Rule will require all PWR Licensees to address just that question and its implications for containment integrity. Proposed Section 50.44(c)[3][ii] 45 FR 65466, at 65472. We believe we can rely upon the Commission's implied judgment that operation of Rancho Seco, and that of all other PWRs, in the interim will not present an undue hazard to health and safety.

N. Venting Back Into Containment

207. **CEC Issue 5-1:**

> Whether those systems identified as contributing to the release of radioactivity during the TMI accident, which are outside containment, should be changed to vent into the containment building?

The testimony of four witnesses was received on this issue: NRC Staff Testimony of James Wing on Changing The Systems Outside Containment to Vent into Containment Building, fol. Tr. 2740 (Wing Testimony); NRC Staff Testimony of Jack N. Donohew on Changing the Systems Outside Containment to Vent into Containment Building, fol. Tr. 3168 (Donohew Testimony); Licensee's Testimony of Robert A. Dieterich, ... fol. Tr. 1988 (Dieterich Testimony); Prepared Direct Testimony of Bruce J. Mann Concerning Release of Radioactivity from Containment, fol. Tr. 2926 (Mann Testimony).

208. The issue reflects concern that during the TMI accident, there were diverse pathways for escape of radioactive materials from the TMI containment. Mann Testimony at 1-11, Wing Testimony at 2, Dieterich

Testimony at 17. Thus, the issue raises questions whether similar release paths may exist at Rancho Seco and, if so, whether systems involving such paths should be modified to return leakage to the containment and thus ensure that such releases do not occur at Rancho Seco. We find, for the reasons given below, that the evidence does not support imposition of this, or any similar requirement.

209. A major contributor to release of radioactivity during the TMI accident was the fact that the TMI containment isolated only on high reactor building pressure. This delayed isolation until several hours after the accident began, thus permitting radioactive releases. Dieterich Testimony at 18. The Rancho Seco containment isolates on low primary system pressure (1600 psig), as well as on high reactor building pressure and this isolation signal would come very early in a TMI-type accident sequence. *Id.* at 18-19.

210. However, even after the TMI containment had isolated, additional significant releases from containment to the auxiliary building were experienced because of the necessity to operate certain systems, including the letdown system. Mann Testimony at 14. Indeed, the letdown system was probably the most significant pathway for radioactive releases at TMI (Tr. 3172), so, early containment isolation, while helpful, does not ensure that there will be no releases of radioactivity from the reactor building.

211. SMUD has instituted two programs to attempt to ensure that radioactive releases to the environment will not occur. First, SMUD has identified essential and nonessential systems within containment and has taken steps to ensure that all nonessential systems will be isolated immediately upon either high reactor building pressure or low primary system pressure. Wing Testimony at 3-5. The Board finds that this program should reduce the chance of releases through systems which are not needed after an accident. In addition, SMUD has instituted a leak reduction program concerning its radwaste system, designed to ensure that leakage in that system will be kept to a minimum amount. *Id.* at 3-5.

212. With respect to the concept of "venting back" into the containment those systems outside containment, it appears that no party is advocating that such a system be imposed at Rancho Seco.[27] CEC's Witness Mann suggested that "venting back" might be worthy of study as a way to improve the safe response of the Rancho Seco plant in the event of a severe

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[27]Simply stated, such a system would return into containment any leakage from those systems and components outside containment that have potential radioactive release paths to the environment. At TMI, those systems located outside the containment that released radioactivity to the atmosphere were the waste gas system, the reactor coolant bleed holdup tank relief valve in the letdown system, the fuel handling and auxiliary building sump tanks, the radwaste system pumps, and the makeup and purification system. Wing at 2.

accident, but he admitted that he had not studied the concept. Tr. 2932-2936. Both NRC Staff and SMUD witnesses stated that venting back into containment was unnecessary at this time. Donohew Testimony at 9; Wing Testimony at 9; Tr. 2129, 2136; Tr. 2762-2764; Tr. 3173, 3174. It is also true that implementation of a "venting back" system would require making a number of new penetrations into the containment, and the addition of valves, pumps and pipes subject to leakage. Tr. 2129, 2134-2136. Thus, a "venting back" system could result in a reduction rather than an increase in reactor safety. Tr. 3173-3176. Given the lack of support of this idea by any party, even the party that broached it (CEC Proposed Findings, para. 254-259), given also that the concept would be complex to implement and would not necessarily improve safety, the Board finds that there is no reason to have systems located outside of containment at Rancho Seco modified to vent into the containment building. The "vent back" concept is just one potential design modification to reduce radioactive releases following an accident. We have no reason to believe it would be a very effective one.

O. Controlled Filtered Venting

213. **CEC Issue 5-2:**

Whether the containment building should be modified to provide overpressurization protection with a controlled filtered venting system to mitigate unavoidable release of radionuclides?

This issue was, it appears, raised by CEC primarily as a result of conclusions reached in two reports produced under CEC's aegis: CEC's Draft Staff Report "Underground Siting of Nuclear Power Reactors: An Option for California" (SMUD Ex. 11) and "Analysis of Public Consequences from Postulated Severe Accident Sequences in Underground Nuclear Power Plants" (SMUD Ex. 18).[28]

214. The underground siting study (SMUD Ex. 11) not only examined underground siting but also considered controlled filtered venting (CFV) as it could be applied to conventional, surface-sited plants. Prepared Direct Testimony of Daniel Nix Concerning Controlled Filtered Venting fol. Tr. 2403 (Nix Testimony) at 2.

[28]The reports were prepared in response to three laws constraining the licensing of nuclear power plants in California until after certain determinations could be made. The laws have been declared unconstitutional under the Supremacy Clause by two district courts: *Pacific Gas and Electric Co. v. State Energy Resources Conservation and Development Commission* 472 F. Supp. 191 (S.D. Ca. 1979). These cases have been consolidated and now pend before the Ninth Circuit Court of Appeals.

215. As a result of this study, which concluded that a CFV system was an attractive but less effective alternative to underground siting (SMUD Ex. 11 at X), and as a result of the TMI accident, CEC sought consideration of this issue with respect to Rancho Seco in this proceeding.

216. Licensee argued, however, that this issue was not appropriate for consideration by this Board and, consequently, moved for summary disposition on the ground that the CFV system proposed by CEC would be intended to mitigate accidents more severe than the design basis accident (DBA) for the Rancho Seco containment and that, therefore, the proposal constituted an impermissible challenge to a Commission regulation—the General Design Criteria set forth in 10 C.F.R. Part 50, Appendix A, particularly Criteria 16 and 50. Licensee argued that any modification of the General Design Criteria could only be made by the Commission and would be beyond the power of this Board to direct. The NRC Staff supported Licensee's motion on the ground that the issue was beyond the scope of the May 7 Order, whereas CEC opposed the motion. Be that as it may, this Board denied the motion and a subsequent motion for reconsideration because, in our mind, the issue did not constitute an impermissible challenge to the Commission regulations and was within the scope of the hearing. Tr. 100, 356, 357.

217. Specifically, we believe that a proposed CFV system does not challenge any Commission regulation with respect to containment design criteria, because those regulations only set forth a requirement that the containment remain leak-tight throughout a design basis accident. *See* 10 C.F.R. Part 50, App. A, Criteria 16 and 50. Since a CFV system can enhance containment integrity, as we will find below, it does not conflict with the containment design criteria which we believe set forth minimum standards. In addition, as discussed below, there is controversy surrounding the probability and choice of a design basis accident. Accordingly, we decided to receive evidence and testimony on this issue. Before we explore the function, theory, design and uncertainties of a CFV system, and whether such a system is needed at Rancho Seco at this time, we will briefly explore the question of exactly what eventualities the Rancho Seco containment is designed to protect against.

1. Rancho Seco Containment Design

218. As indicated above, the containment building is designed to hold radioactive materials that may be released during operation of the reactor or in the course of a design basis accident. It is to remain leak-tight. Tr. 2230; 10 C.F.R. Part 50, App. A, G.D.C. 15 and 50. It accomplishes this mainly through its steel lined, reinforced concrete structure. NRC Staff

Testimony of Thomas A. Greene on Containment Overpressurization Protection, fol. Tr. 2783 ("Greene on Containment"), at 3.

219. There are approximately 70 penetrations into the containment. Each penetration contains a line going across the containment boundary, and each line is provided with a redundant set of valves to ensure that the opening can be sealed tightly on demand. Each penetration is sealed to the line it carries with weld material. Those seals are designed to withstand temperatures of at least 286°F. Tr. 2214.

220. There are many ways in which the integrity of the containment theoretically can be breached during an accident. For instance, Table 7 in the Nix Testimony at 15, lists nine categories of PWR accidents that can lead to radioactive releases outside containment.[29] While the nine "release categories" described in that table are intended to represent dominant release sequences for PWRs,[30] the list is not necessarily exhaustive. Tr. 2495. All but the last accident sequence included in the table result in accidents more severe than the design basis accident for Rancho Seco. Tr. 2494, 2495; Nix Testimony at 4, Table 1.

221. Two of the nine release categories in Table 7—PWR-2 and PWR-3—include failure of the containment from overpressurization as the mechanism for radioactive releases to the environment.[31] The CFV systems suggested by CEC Witness Nix and discussed at the hearing are intended to provide protection *only* against these two release categories. Tr. 2495. Accordingly, we will limit our inquiry into the Rancho Seco containment design to determine if it meets present criteria with respect to overpressurization protection.

222. The design of the Rancho Seco facility provides two forms of protection against overpressurization. One is an overpressurization protection system consisting of the containment building spray system and the containment building emergency cooling system. Greene on Containment at 2. the containment building spray system features two separate trains of equal capacity which spray water and sodium hydroxide to remove aerosol fission products released to the containment atmosphere. *Id.* at 2, 5. The containment building emergency cooling system consists of four fan-cooler units and four emergency upper dome circulators. These two systems

[29]Table 7 was taken from the "Reactor Safety Study," WASH-1400 (1975), a report assessing the accident risks in U.S. commercial nuclear power plants. The table summarizes 130,000 accident sequences analyzed in the Reactor Safety Study. Tr. 2493.

[30]The nine PWR release categories set forth in Table 7 of the Nix Testimony are often denoted as "PWR-1," "PWR-2" and so on. That shorthand notation will be used here.

[31]In the PWR-1 sequence, the containment is ruptured by a missile generated by a steam explosion. In the PWR-4, PWR-5 and PWR-8 sequences, there is a failure of containment isolation. In the PWR-6 and PWR-7 sequences, the core melts through the containment building's foundation. Nix Testimony at Table 7.

remove energy from the containment atmosphere following an accident and, if working properly, will prevent the containment from becoming overpressurized during a design basis accident.[32] *Id.* at 2; Tr. 2223-2224, 2264-2265.

223. The second and principal protection against overpressurization is the design of the containment building itself. The Rancho Seco containment building is designed to withstand a "design basis accident" consisting of the pressure loadings resulting from the double-ended rupture of the largest pipe in the primary system. Licensee's Testimony of Robert A. Dieterich in Response to CEC Issue 5-2 fol. Tr. 1983. (Dieterich Containment Testimony) at 3. *See also,* General Design Criterion 50, Appendix A to 10 C.F.R. Part 50. The maximum calculated containment pressure produced in the design basis accident is 52 psig, and the Rancho Seco containment design pressure is 59 psig, offering a 12% margin over the pressure calculated in a design basis accident. Greene on Containment at 3; Dieterich Containment Testimony at 3. The building is also designed to withstand that internal pressure in the presence of wind and earthquake loadings. Tr. 2215.

224. Because of the number of very conservative assumptions and safety margins included in the design, the Rancho Seco containment building would be able to withstand pressures well in excess of 59 psig. Greene on Containment at 7; Dieterich Containment Testimony at 3; NRC Staff Testimony of Dr. James F. Meyer on Containment Overpressurization Protection (CEC Issue 5-2), following Tr. 2786 (Meyer Testimony) at 4; Tr. 2215; Tr. 2830-2832. In fact, two analyses performed by the Structural Branch of the NRC Staff and its consultants showed that a large PWR containment such as Rancho Seco's would withstand, without failure, pressures twice as large as the design pressure, *i.e.,* approximately 120 psig. Tr. 2809, 2868-2871; Tr. 2865-2866, 2900-2901. A more recent study by Sandia Laboratories of large PWR containments has produced a family of containment failure pressures, based on particular loading progressions in the containment, which range from 90 psig to 150 psig. Tr. 2866-2867, 2900-2901. In the course of testing prior to startup, the Rancho Seco containment was pressurized with air up to 69 psig and held there for over a day without detriment. Tr. 2216, 2809.

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[32]Both the PWR-2 and PWR-3 release categories include failure of the containment spray and heat removal systems as part of the scenario leading to overpressurization. Nix Testimony at 15 and Table 7. If these systems only become operative after the containment has reached high pressure and temperature conditions, they may be unable to control the transient completely; however, to the extent they are operational, they will continue to mitigate, if not control pressure and temperature rises produced in an accident such as a core melt. Tr. 2804-2806; Tr. 2223, 2224.

225. All witnesses who addressed the subject stated that there is a great degree of uncertainty on what the actual failure pressure of the Rancho Seco containment would be. There was widespread agreement, however, supported by test data, that the containment would not fail for pressures under 70 psig. Tr. 2688; Tr. 2830, Tr. 2215. As pressure increases, there is an increasing probability that the containment will fail; that probability remains quite low until about 100 psig and then, depending on the containment loading history, it increases dramatically and is influenced by wind loading, earthquake loading, and the accident sequence. Tr. 2810; 2828, 2871, 2872; 2358, 2359.

226. Another source of uncertainty is the form that the containment failure would take. It is possible that, at least for some overpressurization sequences, the containment would not fail catastrophically, but would develop cracks in the concrete that would relieve the containment pressure and then seal back. Tr. 2361; Tr. 2691, 2706-07; Tr. 2867; Tr. 2872, 2873. The releases from such a failure mode would be significantly lower than those generated by a large-scale catastrophic failure, (i.e., one resulting in large permanent openings of the containment structure). Tr. 2867.

227. Because of the foregoing, the Staff and the Licensee have concluded that the Rancho Seco containment building meets the applicable design criteria of 10 C.F.R. Part 50, Appendix A, and is, therefore, safe to operate. Its design is based on a conservative pressure calculation resulting from release of the reactor coolant to the containment atmosphere in the event of a loss-of-coolant accident. Greene on Containment at 7; Meyer at 7; Dieterich Containment Testimony at 2, 3. The uncontroverted evidence presented by the Staff and Licensee shows that the Rancho Seco containment will withstand and remain virtually leak tight during a design basis accident. We agree and note that because of conservative margins designed into the safety systems, certain accidents beyond the DBA can be accommodated. Meyer Testimony at 4. There is, however, no real agreement as to exactly what overpressure or exactly what postulated scenarios the containment can withstand.

228. We now turn to the issue posed by CEC's Issue 5-2. That is, whether the Rancho Seco containment building should be modified to provide further overpressurization protection through use of a controlled filtered venting system (CFVS). This system could be used to mitigate the effects of some severe accidents beyond the design basis accident. Meyer Testimony at 2; Nix Testimony at 2, 14. In our inquiry into this matter we will (1) explore the function, theory and design of a CFVS, (2) analyze the potential risk reduction if a CFVS is utilized, and finally (3) decide whether a CFVS should be installed at Rancho Seco.

2. Controlled Filtered Venting System

a. Function, Theory and Design

229. Controlled filtered venting is a process in which a portion of the containment atmosphere is deliberately released to the environment in a controlled manner through a system of filters and energy absorbers. Such a pressure relief system would be actuated to reduce containment pressure, a pressure which could otherwise fail the containment and thereby allow the uncontrolled and unfiltered release of radiation into the atmosphere. As indicated above, this system would only be used to mitigate the effects of certain severe accidents beyond the design basis accident. *Id.;* Nix Testimony at 8. Various types of CFVSs have been installed or are being installed in Fast Breeder Reactor facilities both here and abroad. For example, the Zero-Power Plutonium Reactor (ZPPR) test facility, the Fast Flux Test Facility (FFTF), and the German SNR-300 prototype LMFBR all have CFVSs or are installing them. Nix Testimony at 15-16; Meyer Testimony at 4-5; Tr. 2239-41.

230. Vent-filter systems for LWRs have received attention since 1975, when Norwegian and Swedish studies on underground siting considered the use of the surrounding soil and rock as a filtering medium. Subsequently, a UCLA study group presented a conceptual design of a vent-filter system for LWRs comprised of a graded sand and gravel bed with downstream HEPA and charcoal filters. Their design included the use of hydrogen burners to minimize the likelihood of hydrogen explosions and air cooling fans to prevent overheating of the charcoal filters. And, of course, the use of a controlled vent-filtered system for core melt accidents was considered in the conceptual study of underground nuclear plants for the California Energy Commission (CEC) mentioned before. SMUD Ex. 11. The CEC design was completely passive, with the principal filtering structure being an underground pressure relief volume filled with crushed rock and gravel. Meyer Testimony at 4, 5; Nix Testimony at 15, 16.

231. The system of SMUD Ex. 11 is a passive system (*i.e.,* it does not require supplied power or signal to actuate it), a system in which the containment atmosphere is discharged through a number of access points or "ports" which, up to the time of system activation, are sealed by metallic discs designed to rupture at a predetermined pressure.[33] Tr. 2614-2616; Nix Testimony at 8, 11. In order to maintain reliability and retain the system's

[33]The discs could also be made temperature-sensitive, so that they would rupture when the containment temperature rose to a certain setpoint. The object of this feature would be to relieve the containment when the containment seal integrity was threatened by high temperatures. Tr. 2623; Nix Testimony at 11. Since the same disc would be subject to rupture on two separate conditions, however, there would be at least two different malfunction modes for each disc. Tr. 2624.

passive nature, which is one of the main advantages perceived by its proponents (*See, e.g.,* Nix Testimony at 10, 11), the discs would have to be designed to rupture at or near one single prescribed pressure. *Id.* at 11. Choosing the proper rupture pressure for the discs would be quite important, for if the discs ruptured at a pressure much below the containment building's catastrophic failure pressure there could be an unnecessary release of radioactivity. Dieterich Containment Testimony at 6; Meyer at 3. And we note that selecting a "catastrophic" failure pressure is complicated by the fact that overpressurization failure might take the form of self-sealing cracks. Pressure relief by that mechanism might be preferable to venting the entire atmosphere through a CFVS. Tr. 2825-26. On the other hand, if the rupture pressure setpoint was too high the system would be ineffectual, for the containment would fail before the discs ruptured. Even if the CFVS was activated prior to containment failure, if its rupture setpoint was too high, pressure relief might not come fast enough to stop the pressure increase before failure of the containment. Meyer Testimony at 3; Nix Testimony at 10.

232. Failure of the discs by overpressurization and/or temperature would allow containment gases to flow through steel piping to the filtration systems. Various systems were examined in the CEC study including use of the unconfined natural soil around the reactor foundation, a system of rock, sand, and charcoal to remove organic iodine. Nix Testimony at 11.

233. The evidence indicates that the technology exists to implement an effective CFVS although some problems remain and some sophisticated systems are very expensive. Meyer Testimony at 6. Some of the conceptual problems remaining are: (a) determining the proper rupture pressure for the discs,[34] (Tr. 2359; Tr. 2826-2829; (b) achieving effective filters that remove the radioactive material and do not become "plugged up," (Tr. 2641-2654); (c) the interactions between a CFVS and the plant's engineered safety features; (d) the possibility of hydrogen ignition due to the system's operation; (e) the potential adverse impact of a CFV discharge because of the temperature reduction caused by the filtration system, which might make the resulting plume less buoyant than an unfiltered one; and (f) the possible exacerbation of low-consequence accidents into high-consequence accidents. Meyer Testimony at 6; Dieterich Containment Testimony at 5, 6; Tr. 2250-2280; Tr. 2691-2692; 2719-2725; Tr. 2821-2825; 2835; 2836.

234. The design performance uncertainties and potential problems previously discussed are reflected in an even greater uncertainty as to the

[34]This problem could be partially eliminated by replacing discs with valves activated by operators from the control room. This solution negates the passive nature of the discs and introduces other failure modes. Tr. 2644-2647; Tr. 2836.

cost of implementing a CFVS. *See* Nix Testimony at 17. The Underground Siting Study estimated a cost of 14 million 1977 dollars for implementing such a system in a new facility. Tr. 2491, 2640; Nix Testimony at 17. The costs of retrofitting an existing facility such as Rancho Seco with a CFVS could include, among other things: the cost of creating a number of new large containment penetrations[35] (Nix Testimony at 17); the cost of building the CFVS to seismic-1 standards[36] (Tr. 2638-2640); the cost of adding expensive filtering materials to attenuate and hold-up radionuclides such as iodine and noble gases (Tr. 2652-2657, 2664; Tr. 2879, 2880); the cost of ensuring reliability if the system is active (as opposed to a passive design), and especially if it is manually operated (Tr. 2816); the cost of developing and licensing the system (Tr. 2287; Tr. 2679, 2680); the cost of down time during installation (Nix Testimony at 17); and the cost of procuring, installing and maintaining the system. A preliminary estimate given by Staff Witness Meyer for retrofitting a CFVS to the Indian Point facility was $15 to $45 million. Tr. 2820.

235. In spite of these conceptual design problems and the large cost uncertainties, both the NRC Staff and CEC agreed that a CFVS could be designed and implemented to vent large volumes of gases and vapor in a controlled manner and to attenuate and hold up virtually any radioactive isotope known to be harmful. Meyer at 5, 6; Nix Testimony at 11. Licensee's witness, on the other hand, stressed the uncertainties and the unbounded costs, even pointing out possible exacerbation of safety problems. Dieterich Containment Testimony at 5, 6; Tr. 2250-2280; Tr. 2691-92; Tr. 2719-2725; Tr. 2821-25. In short, it appears to the Board that there is some dispute among the experts as to whether a practical system can be designed at present.

b. Risk Reduction Potential

236. The risk reduction potential has been discussed at some length by both the Licensee and CEC. *See* Licensee's Proposed Findings, para. 259-271; CEC Proposed Findings, para. 279-297; Nix Testimony at 2-8, 12-15. The thrust of these discussions has been to determine whether the risk (probability of an event multiplied by its consequences) posed at Rancho Seco by accidents more severe than the DBA is sufficiently high to warrant further study or implementation of a CFVS designed to mitigate or eliminate the risk. If the risk is sufficiently small, the usefulness or added

[35]Licensee Witness Dieterich testified that all containment penetrations at Rancho Seco are committed to other uses and there are no available penetrations large enough to be useful for venting. Tr. 2388, 2389. Large new penetrations would have to be made leak tight, resulting in costly plant modifications. Tr. 2384.

[36]CEC Witness Nix estimated that a CFVS designed to seismic-1 standards would cost about 50% more than if such standards did not have to be met. Tr. 2639, 2640.

benefit of a CFVS may be outweighed by its costs or by its negative features that may result from implementation or activation. CEC has concluded that the overpressurization failure accidents are significant contributors to overall public risk. Nix Testimony at 14. This conclusion is derived, in part, from the results of the studies in SMUD Exhibits 11 and 18. These studies indicate that the potential consequences of such an accident are large. Licensee argues that the consequences calculated in the CEC studies are conservatively high but, nevertheless, "not out of line with other risks, both man-made and natural, deemed acceptable by society although not necessarily by all individuals." Licensee Proposed Findings, para. 269. We have no reason to consider in depth these results insofar as accident consequences are concerned for the specific accident sequences which failed containment because we believe it is evident that any catastrophic failure of containment will result in large consequences. The evidence also indicates that a CFVS could reduce these consequences if one can be designed properly. Certainly a filtered venting could only be better than an unfiltered one.

3. Conclusion

237. We are here confronted with disagreement among experts as to the practicality of CFVS design and implementation. At best, the system would be of use only in mitigating certain very severe and very improbable accidents. Tr. 2461; Tr. 2502-05; SMUD Ex. 18 at II-2, II-11, II-12. At worst, for example, if its design were not very carefully thought out, it might exacerbate some accidents. In the period between when our record closed and this writing, the Commission has announced a proposed rulemaking consideration of Degraded or Melted Cores in Safety Regulation, 45 FR 65474, October 2, 1980. In announcing its proposal the Commission set forth specific considerations to be dealt with in the proposed rule, among them:

> Should the NRC require construction, at each nuclear reactor plant site, of a new structure for controlled filtering venting of the reactor containment structure? Would you limit the function of such a new structure to filtering particulates, elemental iodine, and inorganic iodine or would you include adsorption bed systems using charcoal or other processes so that organic iodine and noble gases could also be trapped? What quantities and release rates of gases and particulates would you design such a structure to handle and at what removal efficiency and cost? Do the potential reductions in risk expected from such a structure offset potential increases in risk that may materialize

from incidents such as inadvertent operation or the concentration of hydrogen in the filtering apparatus?

45 FR 65474, at 65476

Clearly the Commission has in mind dealing with just those questions which we have found to be points of disagreement among the expert witnesses we have heard.

238. It seems to the Board unlikely that we would be able, either on the basis of the present record or on the basis of a study prepared especially for this case (as urged by CEC: CEC's Proposed Findings at 196) to reach a more accurate conclusion (or reach a conclusion with greater dispatch) than can the Commission in its proposed rulemaking. The probability of occurrence of an accident in which even a perfectly designed CFVS would be of use is low, the chance of imperfection in a hurried design is great, and steps toward careful consideration of the matter are in progress. We see no need to order a special study, *a fortiori* we see no need to order installation of a CFVS at present.

P. Concluding Findings of Fact

239. This is a special proceeding in which the Commission has vested limited authority with the Licensing Board. We interpret our jurisdiction to determine (1) whether the requirements of the Commission's May 7, 1979 Order provide reasonable assurance that Rancho Seco will respond safely to feedwater transients, and (2) whether SMUD's management and plant operators are sufficiently competent to operate the plant in a safe manner.

240. The Board has evaluated the adequacy of the requirements of the May 7 Order (plus management and operator competency) - including a recognition of additional changes at Rancho Seco since the TMI-2 accident. The short-term actions were completed prior to the restart of the facility. The long-term modifications are not the only changes that have been made and are being made to the facility since it restarted.

241. The record supports, and we have found, that the short-term actions added reliability to the reactor system and increased operator knowledge in order to safely respond to feedwater transients. CEC contends that the short-term actions were quickly devised without appropriate in-depth analysis and are therefore inadequate. *See* CEC Proposed Findings, paragraphs 30, 327-330.

While we can appreciate that these actions were devised and implemented as quickly as possible in order to avoid costly shutdowns, and we can appreciate that some of these actions may, through further review, be modified in favor of a better approach, this Board's review of the record leads us to conclude that the short-term actions were adequate to provide reasonable assurance that Rancho Seco could respond safely to feedwater transients. It is clear that these actions did not eliminate the sensitivities of the B&W nuclear steam supply system which we have discussed in depth. It is also clear that these actions did not provide Rancho Seco operators with a complete and perfect understanding of plant conditions and proper operator actions in the event of a feedwater transient. We are confident that no analyses could be undertaken and no requirement or modification imposed that would provide an absolute certainty that Rancho Seco could respond safely to feedwater transients. Such a requirement is not imposed by the Commission's regulations or by the May 7 Order. Accordingly, we find that the testimony and evidence produced in this proceeding supports a conclusion that the short-term actions imposed by the May 7 Order provide the requisite reasonable assurance that Rancho Seco can be operated in a safe manner.

242. CEC also asserts that the long-term modifications contained in the May 7 Order are deficient in that, again, they were imposed without sufficient analyses. *Id.* at para. 331. As evidence of this deficiency, CEC basically alludes to the "moving target" of additional items to be accomplished. *Id.* However, CEC is careful to point out that these deficiencies do not require an immediate shutdown of the Rancho Seco facility. *Id.* at para. 29.

243. This Board has noted above that review efforts continue with respect to this matter and we deem it important that they be completed as expeditiously as possible. However, we expect that the NRC Staff and industry will always review systems, instruments, equipment, operator procedures and competence in order to improve the performance of nuclear facilities and to provide additional margins of safety. The record compiled in this proceeding contains several suggestions for further modifications and requirements to be imposed on or studied by the Licensee. We have found, however, that the record does not support their adoption at this time, although some may deserve additional study by the NRC and industry on a generic basis. None of them are required to provide reasonable assurance that the Rancho Seco facility will respond safely to feedwater transients or to further enhance management's and operator's understanding and safe operation of the facility. Therefore, the Board finds that the long-term modifications directed by the Commission in its May 7 Order, along with other changes imposed since that Order in response to the investigations of, and lessons learned from, the accident at Three Mile Island, are sufficient to provide such assurance.

III. CONCLUSIONS OF LAW

244. The Board has considered all documentary, written and oral evidence presented by the parties on the questions raised by the Board and on the issues raised by the California Energy Commission. Based upon a review of the entire record in this proceeding and the foregoing findings of fact, the Board enters the following conclusions of law.

245. The actions required by subparagraphs (a) through (e) of Section IV of the Commission's Order of May 7, 1979, are necessary and sufficient to provide reasonable assurance that the facility will respond safely to feedwater transients, pending completion of the long-term modifications set forth in Section II of the May 7 Order.

246. Licensee should be required to accomplish, as promptly as practicable, the long-term modifications set forth in Section II of the Commission's Order of May 7, 1979.

247. These long-term modifications, coupled with the additional changes completed and being undertaken at the facility, including management and operator competency are sufficient to provide continued reasonable assurance that the facility will respond safely to feedwater transients.

IV. ORDER

248. WHEREFORE, IT IS ORDERED, in accordance with 10 C.F.R. Sections 2.760(a) and 2.762, that this Initial Decision shall constitute the final action of the Commission thirty (30) days after the date of issuance hereof, unless exceptions are taken in accordance with Section 2.762 or the Commission directs that the record be certified to it for final decision. Any exceptions to this Initial Decision or designated portions thereof must be filed within ten (10) days after service of the decision. A brief in support of the exceptions must be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff). Within thirty (30) days of the filing and

service of the brief of the appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

THE ATOMIC SAFETY AND LICENSING BOARD

Frederick J. Shon
ADMINISTRATIVE JUDGE

Dr. Richard F. Cole
ADMINISTRATIVE JUDGE

Elizabeth S. Bowers, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
May 15, 1981

[Appendixes A & B have been deleted from this publication but are available at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.]

Cite as 13 NRC 652 (1981) LBP-81-1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Robert M. Lazo, Chairman
Emmeth A. Luebke
Richard F. Cole

In the Matter of **Docket Nos. 50-369-OL**
50-370-OL
(Reopened Operating
License Proceeding)

DUKE POWER COMPANY
(William B. McGuire Nuclear Station,
Units 1 and 2) **May 26, 1981**

Upon completion of a reopened operating license hearing to conside hydrogen generation and control in an ice-condenser containment, th Licensing Board, *inter alia,* (1) finds reasonable assurance that the facilit can be operated without undue risk to the public health and safety wit respect to possible hydrogen generation; and (2) empowers the Director Office of Nuclear Reactor Regulation, upon making the requisite finding with respect to matters not explored by the Licensing Board (and subject t the terms of applicable Commission regulations), to issue full powe operating licenses for Units 1 and 2.

OPERATING LICENSE PROCEEDINGS: RESPONSIBILITY OF LICENSING BOARDS; NRC STAFF

In operating license proceedings, license boards are in general calle upon to decide only the issues in controversy among the parties. 10 CFR § 2.760a. The other matters required to be determined prior to the issuance o an operating license are entrusted to the Director of the Office of Nuclea Reactor Regulation. 10 CFR §§ 2.760a, 50.57.

TECHNICAL ISSUES DISCUSSED:

Hydrogen generation;
hydrogen ignition;
hydrogen control;
containment capability;
polyurethane foam combustion;
operator training;
operating procedures.

APPEARANCES

Mr. William L. Porter, Charlotte, North Carolina and **Messrs. J. Michael McGarry** and **Malcolm H. Phillips, Jr.**, Washington, D.C., for the applicant, Duke Power Company.

Mr. Shelley Blum, Durham, North Carolina, for the intervenor, Carolina Environmental Study Group.

Messrs. Edward G. Ketchen, Stephen H. Lewis, Lawrence J. Chandler and **Joseph F. Scinto**, for the Nuclear Regulatory Commission staff.

Dr. John M. Barry, Charlotte, North Carolina, for Mecklenburg County, North Carolina.

Mr. David Carelock, Charlotte, North Carolina, for the City of Charlotte, North Carolina.

INDEX

SUPPLEMENTAL INITIAL DECISION
(Reopened Operating License Proceeding)

I. INTRODUCTION

1. An Initial Decision (Operating License Proceeding) was issued on April 18, 1979, *Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2), LBP-79-13, 9 NRC 489 (1979) and is incorporated by reference in this Supplemental Initial Decision. On the basis of specific findings of fact and conclusions of law, the Licensing Board ordered that the Director of the Office of Nuclear Reactor Regulation, upon making requisite findings with respect to uncontested matters not embodied in the Initial Decision, was authorized to issue operating licenses for the two units. *McGuire, supra.* 9 NRC at 547-8. However, at that time, the Licensing Board stayed the effectiveness of the Initial Decision "until further order by the Board following the issuance of a supplement to the Nuclear Regulatory Commission ("NRC") Staff's Safety Evaluation Report ("SER") addressing the significance of any unresolved safety issues." The NRC Staff issued the supplement on May 23, 1980 as Supplement No. 3 to NUREG-0422 (May 1980) (Staff Exhibit H). On May 30, 1980, the Duke Power Company (Applicant) moved the Licensing Board to lift its stay of the Initial Decision.

2. On June 9, 1980, Intervenor, Carolina Environmental Study Group (CESG), filed a response opposing Applicant's motion to lift the stay; also a motion requesting the reopening of the McGuire operating license hearing and the admission of new contentions. CESG amended its motion to reopen on August 15, 1980, and advanced four contentions relating to hydrogen generation and control. By Memorandum and Order of November 25, 1980, the Board granted CESG's motion to reopen and admitted CESG's four Contentions and denied the Applicant's motion to lift the stay of the Initial Decision.

3. On November 25, 1980, the Board granted Applicant's motion for a low power license to the extent of authorizing fuel loading, initial criticality, and zero power testing. We denied Duke's request with respect to low power testing at up to 5% of full power. License NPF-9 for fuel loading, initial criticality and zero power testing was issued January 23, 1981.

4. On November 7, 1980, CESG moved to admit two additional contentions relating to Class 9 accidents and to emergency planning for such accidents. On February 13, 1981, the Licensing Board denied CESG's motion to add these two additional contentions.

5. Mecklenburg County on December 23, 1980 and the City of Charlotte on January 13, 1981, requested participation in the reopened hearings as interested government bodies as permitted in 10 CFR §2.715(c). The Licensing Board approved both requests on January 26, 1981.

6. Public evidentiary hearings regarding the issues raised by CESG were held in Charlotte, North Carolina on February 24-27, March 3-6, March 10-13, and March 17-19, 1981. The parties presenting evidence at the hearings were Applicant, NRC Staff, and CESG. Mecklenburg County and the City of Charlotte participated. The decisonal record in this proceeding consists of the transcripts of the evidentiary hearings, all material received into evidence by the Licensing Board during such hearings, and the decisional record established in the previously issued Initial Decision in the operating license proceeding. At the close of the hearing, the Licensing Board granted the Staff's request to hold the record open for the limited purpose of possible inclusion of the results of the Staff review of a discrete issue regarding polyurethane foam. (Tr. 5252). A Staff affidavit was submitted on March 27, 1981 and has been admitted as Staff Exhibit Q. We permitted the parties an opportunity to respond and Applicant filed an affidavit in response which has been admitted as Exhibit 9. On March 27, Mr. Jesse L. Riley, on behalf of CESG, also filed an affidavit. It has been identified as CESG's Exhibit 63. However, it is not responsive to the Staff's affidavit and has not been received in evidence. An index of exhibits is attached as Appendix A.

7. CESG contentions in this proceeding are as follows:

Contention 1: The licensee has not demonstrated that, in the event of a loss-of-coolant accident at McGuire:

1. substantial quantities of hydrogen (in excess of the design basis of 10 CFR §50.44) will not be generated; and

2. that, in the event of such generation, the hydrogen will not combust; and

3. that, in the event of such generation and combustion, the containment has the ability to withstand pressure below or above the containment design pressure, thereby preventing releases of off-site radiation in excess of Part 100 guideline values.

Contention 2: Neither licensee nor NRC staff has demonstrated that a McGuire ice containment will not breach as the result of the rapid combustion of quantities of hydrogen which a dry containment would withstand.

Contention 3: Neither licensee nor NRC staff has demonstrated that the emergency planning radius of 10 miles is sufficient for protecting the public from the radioactive releases of a low pressure, ice condenser containment ruptured by a hydrogen explosion.

Contention 4: Licensee and NRC planning do not provide for crisis relocation which would be required as a result of containment breach and radioactive particle release.

8. CESG Contention 2 attempts to raise as an issue a comparison of the structural capabilities of the McGuire containment and other larger containment structures. The Licensing Board views Contention 2 as an expression of concern regarding the ability of the McGuire containment to withstand the effects of a hypothetical hydrogen combustion. In that such concern is embraced within CESG Contention 1, specific findings regarding Contention 2 are unwarranted. It also became clear during the course of the hearing that the issues raised by Contentions 3 and 4 relating to emergency planning are not reached unless CESG is successful regarding Contention 1. Tr. 2829-34 and 3434-5. Evidence regarding Contentions 3 and 4 was deferred pending this Board's ruling on Contention 1. Tr. 3481-3.

9. In making the findings of fact and conclusions of law which follow, the Board considered the entire record of the proceeding and all the proposed findings of fact and conclusions of law submitted by the parties. Each of the proposed findings of fact and conclusions of law which is not incorporated directly or inferentially in this Supplemental Initial Decision is rejected as being unsupported in law or fact or as being unnecessary to the rendering of this Decision. The Board is guided in this operating license proceeding by Appendix A, Section VIII of 10 CFR Part 2, which in subsection (b) provides that the Board will make findings on matters in controversy among the parties.

II. MATTERS IN CONTROVERSY

10. CESG Contention 1 is the same as the contention the Board drafted as Contention 11 in the Three Mile Island Restart proceeding to comply with Commission policy. In the TMI Restart proceeding, the Commission issued two decisions regarding the proper scope of litigation of matters involving hydrogen generation [*Metropolitan Edison Company* (Three Mile Island Nuclear Station, Unit No. 1, CLI-80-16, 11 NRC 674 [1980] and Order of September 26, 1980 [Docket No. 50-289 (Restart)])]. The Commission decisions discussed below, are applicable to this proceeding.

11. The issue of hydrogen generation was brought before the Commis sion in the form of two certified questions:

1. Whether the provisions of 10 CFR 50.44 should be waived o exceptions made thereto in this proceeding where a *prima faci* showing has been made under 10 CFR 2.758 that hydrogen ga generation during the TMI-2 accident was well in excess of th amount required under 10 CFR 50.44 as a design basis for the post accident combustion gas control system for TMI-1.

2. Whether post-accident hydrogen gas control should be an issue i this proceeding where post-accident hydrogen gas control wa perceived to be a serious problem and was in fact a problem durin the TMI-2 accident. [*Three Mile Island*, 11 NRC at 674-5.]

12. In its ruling the Commission declined to waive or except th hydrogen generation provisions in 10 CFR §50.44. This regulation limits th amount of hydrogen, generated during the course of a loss-of-coolan accident to hydrogen associated with a five percent metal-water reaction. I must be taken into account in the design of nuclear reactor containmen systems. The Commission, in its Memorandum and Order in the Three Mil Island case stated:

> The Three Mile Island accident has in fact raised a safety issu regarding hydrogen control measures following a loss-of-coolan accident that should be addressed. The Commission believes tha quite apart from 10 CFR 50.44 hydrogen gas control could properly b litigated in [the Three Mile Island Nuclear Station, Unit No. 1 proceeding under 10 CFR Part 100. Under Part 100, hydrogen contro measures beyond those required by 10 CFR 50.44 would be required i *it is determined that there is a credible loss-of-coolant accident scenari entailing hydrogen generation, hydrogen combustion, containment breac or leaking, and offsite radiation doses in excess of Part 100 guidelin values.* The design basis assumptions of 10 CFR 50.44, in particular th assumption that hydrogen generation following a loss-of-coolan accident is dependent on ECCS design as opposed to actual ECC operation, do not constrain the choice of credible accident sequence used under 10 CFR 100.11(a) *Union of Concerned Scientists* v. *AEC* 499 F.2d 1069, 1090 (D.C. Cir. 1974). Thus we answer the secon certified question in the affirmative. (emphasis added)
>
> We answer the first certified question in the negative. We are of cours aware that the Three Mile Island accident resulted in hydrogen bein generated far in excess of the hydrogen generation design basi

> assumptions of 10 CFR 50.44. This was because the operator interfered with actual ECCS operation with the result that the safety system did not operate as designed and as 50.44 assumed it would operate. However, this is a safety issue that is not peculiar to Three Mile Island Unit 1 — it is an issue that is common to all light water power reactors because operators generally have the physical capability to interfere with automatic ECCS operation. The proper response to this issue is not waiver of the rule under 10 CFR 2.758 because this case presents no "special circumstances" but rulemaking to either amend or suspend the present rule. The Commission is planning a broad *rulemaking proceeding* that will address the general question of possible safety features *to deal with degraded core conditions.* This rulemaking proceeding will include measures to deal with hydrogen generation following a loss-of-coolant accident. (emphasis added)
>
> * * *
>
> ... the hydrogen control issue can be litigated under 10 CFR Part 100. Under Part 100 the likelihood of an accident entailing generation of substantial (in excess of 10 CFR 50.44 design basis) quantities of hydrogen, the likelihood and extent of hydrogen combustion, and the ability of the reactor containment to withstand any hydrogen combustion at pressures below or above containment design pressure would all be at issue. A critical issue here would be the likelihood of an operator interfering with ECCS operation.
>
> However, after the Three Mile Island accident the Staff has given licensees explicit instructions not to turn off prematurely the ECCS system. As noted above, it was operator interference with ECCS operation that was the root cause of the hydrogen generation problem at Three Mile Island Unit 2. In our view this instruction which had not been issued when 50.44 and General Design Criterion 50 were promulgated, compensates for the less conservative analytical framework of Part 100, and serves as a basis to sustain the present hydrogen generation assumptions of 50.44 at least for the interim until the degraded core rulemaking can be completed. 11 NRC at 675-6.

The Board has limited its scope to consideration of credible accidents. The degraded core rulemaking is viewed as providing a forum for the treatment of other accidents.

13. The Commission has provided guidance with these rulings. The regulations recognize and allow some measure of hydrogen production and the Commission granted the parties the right to litigate whether excessive

amounts of hydrogen can be generated. Permissible limits for hydrogen are stated in Section 50.44. The Commission, in its TMI ruling provided a way to consider this issue under 10 CFR Part 100. The interpretation is that a party must prove a credible accident that will give rise to the production of excessive hydrogen. A party must show a credible condition wherein the core is inadequately cooled for a sufficient period of time. CESG has raised the hydrogen generation issue, and under the Commission's ruling, it is considered to have the burden to establish a credible accident scenario involving hydrogen production resulting in offsite doses in excess of 10 CFR Part 100 limits. The burden is further clarified by Commissioners Gilinsky and Bradford in their dissent to the Commission's September 26, 1980 Order denying reconsideration of CLI-80-16:

> Moreover, Chairman Ahearne and Commissioner Hendrie are, in effect, saying that even after experience has amply demonstrated the adequacy of safety regulations covering the internal components of the reactor, the burden is still on a challenger to lay out a specific accident sequence to the Commission which leads to containment failure and public radiation exposure in excess of those permitted by Part 100. *TMI (Restart),* supra, Order, dissenting opinion, slip op. at p.2 (September 25, 1980).]

14. Part 100 is a siting regulation, and it establishes radiation limits at a certain boundary from the plant surrounding the "exclusion" area. These radiation exposure limits are 25 rem to the whole body, or 300 rem to the thyroid from iodine exposure.

15. In summary, the question of whether there is a credible loss-of-coolant accident involving hydrogen generation, hydrogen combustion and breach or leakage of the containment, with consequent offsite doses in excess of the Part 100 guideline values, is litigable under 10 CFR Part 100, notwithstanding the provisions of 10 CFR §50.44. The Licensing Board admitted CESG's Contentions 1 through 4 on the basis of this Commission precedent.

III. FINDINGS OF FACT

16. The Applicant offered evidence regarding Contention 1 concerning the lack of credibility of a TMI-type accident sequence. The evidence related principally to a sequence characterized as S2D, which is a small break LOCA sequence with the break occurring anywhere in the primary coolant system; not just a TMI-2 small break sequence caused by an open relief valve initially actuated because of a loss-of-feedwater event. Cross examination was permitted into other sequences in which the evidence

suggested a relevance or similarity to the TMI-type events or to the S2D sequence. (Tr. 3086-89, 3374, 4065).

17. The excessive hydrogen produced during the TMI-2 accident was a direct result of a reaction between the zirconium in the fuel cladding and steam and/or water during a loss-of-coolant accident which led to an inadequate core cooling situation. The Emergency Core Cooling System (ECCS) is designed to prevent an inadequate core cooling situation which could result in high temperatures of the core and excessive hydrogen production. The sequence of events which occurred at TMI-2 was: (1) a loss of feedwater transient resulting in high reactor coolant system pressure which was relieved by the pressurizer relief valve; (2) failure of the relief valve to close resulting in a continued loss-of-coolant; (3) premature operator interference with the emergency core cooling system resulting in inadequate cooling of the reactor core and excessive core temperatures; and (4) production of hydrogen from the reaction of approximately 45% of the zirconium clad and steam in the presence of the excessively high temperatures. If the operators at TMI-2 had not prematurely terminated the ECCS operation there would not have been excessive hydrogen generation. (App. Panel I: Canady, Reed and Barron, following Tr. 2864, Tr. 2870-3, 3086-89, 3374-5, 4468-79 and Staff Testimony Regarding Hydrogen Control following Tr. 4353).

18. Applicant has made improvements at the McGuire facility subsequent to the TMI accident in the areas of personnel, equipment, procedures and training to effectively preclude improper operator termination of the McGuire ECCS. New technical specifications at McGuire require the following personnel changes: (1) a Senior Reactor Operator must be present in the control room at all times, in addition to a Reactor Operator; and (2) a Technical Advisor to the Shift Supervisor must be present on all shifts and available to the control room within ten minutes. Applicant's current staffing procedures require two licensed reactor operators in the control room instead of one. The second reactor operator may be absent for short periods of time. (App. Panel 1 at 2).

19. The technical and diagnostic capability in the control room has been increased by adding a Technical Advisor to advise the Shift Supervisor. The Shift Technical Advisors have been selected from among the group of licensed Senior Reactor Operators at McGuire, all of whom have received additional simulator training and have received additional academic training, including instruction in heat transfer, fluid flow, thermodynamics, and plant transients. The Shift Technical Advisor provides additional evaluation and assessment of both normal and unanticipated transients. The Senior Reactor Operator must be in the control room at all times the plant is above the cold shutdown mode of operation. The Assistant Shift

Supervisor has been assigned administrative duties, relieving the Shift Supervisor of duties which could detract from his management responsibility for safe operation of the plant. These changes provide additional expertise in the control room. An ECCS termination decision will be made by the Senior Reactor Operator with input available from the Shift Technical Advisor and the Reactor Operator. This change provides greater assurance that the ECCS will not be prematurely terminated by operator action. There are an acceptable number of both reactor operators and senior operators to run the McGuire facility. The Applicant is engaged in a long-term hiring and training program to obtain additional reactor operators and senior operators, including additional personnel to account for attrition. (App. Panel I, Staff Exh. I, pp. 22-32; Tr. 2880-4, 3007-13, 3028-32).

20. Equipment modifications have been made at the McGuire plant subsequent to the TMI-2 accident. These include installation of a subcooling monitor to monitor the approach to an inadequate core cooling situation. Alarms are also provided to warn the operator of an approach to a potential inadequate core cooling condition. Applicant is planning to install a reactor vessel level measurement system at McGuire which is designed to monitor the water level in the reactor and provide further indication of an approach to an inadequate core cooling situation. These modifications and additions will increase the assurance that operation of the plant will be done in a safe manner. Pressurized water reactors are operated at temperatures below the saturated temperature. This is the temperature at which water will boil at a corresponding pressure. Equipment modifications were designed to warn the operator of operation beyond the normal operation approaching the saturated temperature that could lead to inadequate core cooling. New instrumentation such as a subcooling monitor and associated equipment would monitor the approach to inadequate core cooling conditions. The subcooling monitor is part of the McGuire in-plant computer system. It will display to the operator on the computer video screens, the actual system conditions on a temperature vs. pressure graph. It will indicate to the operator what the core conditions are relative to a saturation curve. (App. Panel I, Tr. 2925-31, 2983, 3009-12)

21. The subcooling monitor is basically software, which utilizes hard-wired inputs that are scanned at a minimum of once per minute and gives the operator a video display of the information. The problem with computer backlog due to the delay in the printers that occurred at TMI, which did not have a video display, will not occur at McGuire. The operators are thoroughly trained through the use of steam tables to calculate margins to saturation. Thus, they compute margins even if the monitor were to fail. There is no problem similar to that which occurred at

TMI due to computer backlog with respect to input to the control room via the subcooling monitor. (Tr. 2884-7, 3009-11, 3032-4).

22. On redirect, it was shown that a hydrogen bubble would not occur in the reactor vessel and would not interfere with natural circulation. Applicant has installed a redundant reactor coolant venting system located directly off the top of the reactor vessel head. This system allows for detection of any hydrogen bubble formed in the reactor vessel and for venting any hydrogen off into the pressurizer relief tank rather than directly into the containment atmosphere. (Tr. 3061-3, 3092-3).

23. Operator training has been substantially enhanced. Subsequent to the TMI-2 accident, the training program has been revised and expanded to reflect the lessons learned from the accident. Revisions include the addition of a TMI-type accident scenario in the simulator portion of the training program. Priority is given to the avoidance of an inadequate core cooling situation. Candidates for the operator training program are employed by Applicant for several years before being selected. Screening of candidates for the training program includes consideration of aptitude test results, seniority, training grades throughout employment, and employment evaluations. The operator training program is lengthy, over 2½ years, and intensive. The formal program uses an effective mix of formal classroom presentations, research reactor training, on-the-job training using task lists, simulator operation and both written and oral examination. The program includes instruction in power plant operating practices and nuclear theory. Specific course material includes electric theory, heat transfer, fluid flow, thermodynamics, chemistry, physics, mathematics, health physics, reactor theory, nuclear systems, transients, radioactivity theory, radiation detection and instrument control theory. The operators receive training in monitoring and reading of indicators in the control room and on actions to detect and respond to false readings. A demonstration of a TMI accident, and also natural circulation and ECCS operation have been added to the simulator program. The instructors conducting the operator training program at McGuire are, with the exception of one Reactor Operator, all holders of a Senior Reactor Operator license. Prior to receipt of a Reactor Operator license, each candidate is required to comply with all appropriate provisions of 10 CFR Part 55, to include successful completion of an extensive operator examination. Each licensed operator is required to undergo annual requalification training that consists of formal classroom presentations and simulator operations. The requalification training includes approximately 48 hours per year on the simulator responding to simulated emergencies which includes the TMI-accident scenario. Applicant testified that, during operations, McGuire would use five operational shifts: 3 in shift

operation, 1 off duty and 1 in training. (App. Panel I, p. 6-8, Tr. 2852, 2872-80, 2993-4, 3000-7, 3028-9, 3039 FSAR Sect. 13.2)

24. The direct, prefiled written "Testimony of Jesse L. Riley Regarding Hydrogen Generation, Combustion, and Containment Response" with attached "Professional Qualifications of Jesse Riley", bound into the record following Tr. 3780, was rejected by the Licensing Board as evidence in this proceeding. (Tr. 3967-69). The Licensing Board also rejected Mr. Riley's oral direct testimony given on the record at Tr. 3767-3811, 3816-3824, 3864-3875. (Tr. 3967-69). The proferred testimony concerned the generation and combustion of hydrogen, effects of such combustion on the containment structure, the McGuire containment design loading, chemical reactions, reactor systems, and reactor operator performance. The Board concluded that Mr. Riley was not qualified to testify as an expert on matters relating to strength of containment structure, particularly structural engineering aspects, and was also not qualified on hydrogen burning or detonation. (Tr. 3967, 3969). See also (Tr. 3875-3967).

25. CESG's direct case regarding operator training consisted of the testimony of five psychologists. CESG subpoenaed Dr. John Philip Brockway, Dr. Gary Thomas Long, Dr. James Richard Cook, Dr. Edward Leo Palmer, and Dr. John Edward Kello. Their testimony was directed toward general psychological phenomena under certain work conditions, such as boring tasks, information overload, fatigue, stress, cognitive dissonance, group think, risky-shift, obedience to authority, massed and distributive learning, all or none learning, overlearning and amnesia. According to their professional qualification statements, these witnesses had no background with respect to operation of nuclear power reactor facilities and did not relate the general phenomena discussed to nuclear power plant operations or to control room activities. (Fol. Tr. 3624, 3835, 3845, 3853 and 3978; Tr. 3624-39, 3635-8, 3977-82; App. Exh. 8, Tr. 4674).

26. In rebuttal, Applicant presented a panel of psychologists, Dr. Lewis F. Hanes, Dr. Julian M. Christensen, Dr. Eric F. Gardner, Mr. Robert M. Koehler, and Mr. Richard J. Marzec. These witnesses were able to provide a link between theory and practice. The testimony of this panel of psychologists demonstrated that the psychological phenomena mentioned by CESG's psychologists had been taken into account in the structure of its operator training program, the structure of the control room, the organization of the operating personnel, and in the function and duties of the control room operator, or that those concepts do not apply to nuclear power plant operators or operation. The testimony showed that there is little risk of operator error in a nuclear power plant that would affect safe operation of the plant from the effects of such psychological phenomena as, cognitive dissonance, risky shift, group think, forgetfulness, information

overload, boredom, or mental or physical fatigue. Obedience to authority is considered to be a positive attribute for an operator who is required as a matter of safety to follow established safety procedures. It was pointed out that application of psychological concepts and theories, developed on an experimental basis, to real world operation of nuclear power plant operations must be done with extreme caution. Dr. Hanes testified that due to "chunking", an individual with training can handle increasingly greater amounts of information that may seem complex to the outsider. Chunking involves an ability to grasp increasingly large chunks of complex sets of information as training and experience is increased. (Tr. 4715-4841).

27. Stress on operators is not considered to affect the safe operation of the McGuire station. Stress levels in operators of nuclear plants are generally low and the effects of stress on safe operation are reduced to even lower levels by operator training and experience. An operator is required to follow specific procedures. Individuals who do not perform well under stressful conditions during the testing and training phases of the operator training and qualification program are eliminated. Supervisors of operating personnel are trained to recognize any aberrant behavior due to such things as fatigue that might occur in its operators. The training program which mixes training, work experience, and simulator training is considered to be a good example of the learning concept of "articulation" and how Applicant includes such phenomena as "distributive learning" in its training process. (Tr. 4740-72, 4781-99).

28. Changes have been made in administrative and operating procedures. They include: (1) a redefining of the Shift Supervisor's primary responsibilities and duties to emphasize safe operations of the plant; (2) new shift turnover checklists; (3) more stringent restrictions on overtime; and (4) more stringent controls on verification of system availability. These changes are intended to reduce the possibility of any operator error which could lead to premature termination of the ECCS. They are designed to enhance the ability of the reactor operators to operate the plant in a safe and efficient manner. Operating lines of authority and management responsibility have been clarified and formalized. Procedures have been provided to require verification of safety system availability when systems are removed from or returned to service and notification of operators when safety systems are removed from or returned to operation. Detailed shift turnover procedures have been provided, including the use of detailed checklists to assure that current information is provided to the oncoming shift. (App. Panel I, Tr. 2951, 2986-93).

29. Significant changes have been made in Applicant's emergency procedures which will reduce the likelihood of premature termination of ECCS and the possibility of inadequate core cooling. The revised

emergency procedures provide specific criteria for terminating ECCS operation. Applicant testified that prior to the TMI-2 accident, emergency procedures required an operator to first identify the accident which was in progress and then take corrective actions based on the particular procedure for that accident. Subsequent to the TMI-2 accident new emergency procedures have been developed at McGuire requiring operator actions in response to the operator's subjective determination of the accident in progress. Emergency procedures now require that before the operator can terminate the ECCS, four specific criteria must be verified as being within acceptable limits. The specific criteria are:

(1) Reactor coolant system pressure is greater than a specified minimum value and increasing, and

(2) Pressurizer level is greater than a specified minimum value, and

(3) The reactor coolant system is subcooled by greater than 50°F, and

(4) Adequate auxiliary feedwater flow for core heat removal is injected into at least one non-faulted steam generator.

In any of the four criteria are not met, the procedure directs that ECCS operations cannot be terminated. If the four criteria are met, an inadequate core cooling situation cannot exist and generation of excessive hydrogen is impossible. All licensed operators have been thoroughly instructed in the use of such procedures. Operator compliance with these procedures precludes the premature operator termination of ECCS operations thereby preventing inadequate core cooling and excessive hydrogen generation. (App. Panel I, p. 5, Tr. 3013-16)

30. Applicant testified that in the unlikely event of operator premature termination of ECCS operations, emergency procedures require that readings of the four ECCS termination/reinitiation criteria parameters be continuously checked and recorded in a log every 15 minutes for a two-hour period following such termination. These log entries must be independently verified. If any of the parameters are not within acceptable ranges, the procedures require that ECCS operation be reinitiated. In the event of a TMI-type accident at McGuire, if ECCS operation was prematurely terminated, the operator would have over 2 hours to reinitiate ECCS operation before generating an amount of hydrogen in excess of that produced by a 2% zirconium-water reaction. Applicant concluded that even in the incredible event that ECCS operation was prematurely terminated, it is incredible to further assume that within the two-hour period ECCS operation would not be reinitiated prior to generation of hydrogen in excess

of that produced by a 2% zirconium-water reaction. (App. Panel II, Canady, Muench and Barron following Tr. 3045).

31. In its cross-examination CESG asked a wide range of questions concerning the cause of the TMI event, selection of personnel and manning, operating procedures, and equipment and instrumentation, in addition to operator training. The questioning explored the above areas in some detail. The cross-examination failed to establish in any of these areas any material basis for concluding McGuire would be operated in such a manner that ECCS would be improperly terminated in a TMI-2 accident. (Tr. 2865-91, 2920-78, 4385-4480, 4520-27).

32. The Staff agrees with Applicant and testified that equipment changes, enhanced operator training, technical competence and improved operating procedures have substantially reduced the likelihood of recurrence of an event at McGuire such as TMI-2. (Staff Ex. K at 3). In its proposed findings, CESG generally disagrees with these conclusions by Applicant and Staff.

33. The Board finds that actions taken by Applicant, subsequent to the TMI-2 accident, are such that in the event of a TMI-type accident at McGuire the likelihood of ECCS operations being prematurely terminated by the control room operating staff is so remote that such an accident is not credible. In the unlikely event of premature termination of ECCS, the Board finds that current emergency procedures provide reasonable assurance that ECCS will be reinitiated within sufficient time to prevent the generation of hydrogen in excess of the design basis of 10 CFR §50.44.

IV. ADDITIONAL EVIDENCE PRESENTED

34. Considerable additional evidence was received in the record of this reopened proceeding on which separate findings of fact have not been made because such findings have now been determined to be unnecessary to the decision. We recognized during the evidentiary hearing at the close of the first phase of Applicant's case that if we held then, as we now hold here, that CESG's failure to establish a credible accident scenario resulting in hydrogen generation leading to offsite doses in excess of 10 CFR Part 100, should preclude CESG from further litigating the issue of excessive hydrogen generation. Such a ruling would have terminated the hearing at that point. However, in order to build a complete record and provide for the contingency that careful reflection of the evidence after the close of the hearing would lead us to a finding adverse to Applicant regarding CESG's burden of establishing a credible TMI-type accident scenario at McGuire, we elected to receive the additional evidence offered concerning the

McGuire containment response to a TMI-type accident. Pertinent portions of such evidence are summarized below.

Hydrogen Control

35. Evidence regarding the McGuire Plant response to a TMI-type accident involved five major areas: containment structural capability; containment systems that mitigate the effects of excessive hydrogen generation; the ignition and burning of hydrogen; transitions to detonation; and pyrolysis of polyurethane foam.

A. Containment Structural Capability

36. The McGuire containment vessel is a freestanding welded steel structure with a vertical cylinder to which are welded horizontal and vertical stiffeners, a hemispherical dome and a flat base. The vessel is 115 feet in diameter and 171.25 feet high. (App. Ex. 5B at 4-4 and 4-7).

37. Three independent structural analyses were conducted to determine the McGuire containment vessel functional capability (the maximum point at which the containment can be reasonably assured of retaining its leak tight integrity). Tr. 3746-49. One was undertaken by Applicant and reported in Priory Testimony following Tr. 3654 and in Applicant's Exhibit 5B, Chapter 4. Based on its analysis, Applicant concluded that the functional capability of its McGuire containment is 67.5 psig. (Priory Testimony following Tr. 3654 at 2).

38. A second analysis was conducted by Applicant's Consultant, Mr. R. S. Orr and produced a functional capability figure of 68 psig, essentially the same as the Applicant's calculation. (Testimony of R. S Orr Regarding the McGuire Structural Integrity, following Tr. 3654 at 1 and 2, Tr. 3656).

39. The third independent analysis was undertaken by Ames Laboratory of Iowa State University acting as consultant to the NRC staff and is identified as the Staff Analysis. The predicted ultimate strength of the containment shell for the uniform static pressure case was calculated to have a mean value of 84 psig and a standard deviation of 12 psig. Because of anticipated deformations in the shell at 84 psig, the Staff considers the mean pressure minus 3 standard deviations, 48 psig, is the appropriate lower bound capacity at which leak tightness will be assured. (Staff Ex. K following Tr. 4353 at 27 through 33).

40. The probability of containment failure at 48 psi, the lowest of the functional capacity estimates, was calculated to be 4×10^{-5} per occurrence. (Tr. 4894, 4942-43). This multiplied by the probability of a TMI type accident (10^{-5} to 10^{-6} per year) results in an overall probability of failure due to a TMI-type accident of 10^{-10} to 10^{-11} per reactor year. (CESG Ex. 61; Tr. 4943-45).

41. CESG presented testimony of Joe E. Lanford, a sturctural engineer, and former Duke Power Company employee, who testified that he observed excessive grinding around a weld on one of the McGuire containment structures causing a gouge in the base metal of about 1/8 inch deep and several inches long. He brought this to the attention of Duke officials but did not know whether any corrective action was taken. Mr. Lanford opined that if such a gouge had not been repaired, "the containment would be, to some degree large or small, depending on the damage, weakened from the design load capacity." (Tr. 3827-32).

42. Testimony from Applicant and Staff witnesses failed to specifically identify Mr. Lanford's alleged faulty weld. Staff witness Herdt opined that it was not established whether the weld was completed or not. If the weld was not completed (i.e., pre-inspection) and a gouge was noticed, the welder could repair that section and no records of a gouge and/or repair would appear on any Q/A records. (Tr. 4980-81).

43. Both Applicant and Staff witnesses described the welding inspection and Q/A procedures. (Tr. 4847-4853; Tr. 4971, 4972). Significant among the procedures are the required visual and radiographic inspections of every butt weld. (Tr. 4847-4852). Visual inspection is conducted with an instrument capable of detecting width changes in the metal thickness to 1/32 of an inch. (Tr. 4859). Radiographs would detect changes in metal as small as 15/1000 of an inch. (Tr. 4861). Both Applicant and Staff testified that excessive grinding as described by Mr. Lanford would or should have been detected in both visual and radiography inspections and repaired. (Tr. 4850-2; 4972, 4981-2).

44. The record further shows that, even if the gouge described by Mr. Lanford as 1/8 inch deep and several inches long was, for whatever reason, undetected and not repaired, it would have an insignificant effect on containment capability. Because of its ductility, steel can tolerate small imperfections such as the gouge described by Mr. Lanford. (Tr. 4896).

B. Hydrogen Mitigation System Description

45. During the TMI event, hydrogen released to containment was ignited by existing ignition sources within containment. This resulted in a pressure spike of 28 psig. The emergency hydrogen mitigation system (EHM) installed at McGuire consists of 62 igniter assemblies (46 in the lower compartment, 8 in the ice condenser upper plenum, and 8 in the upper containment). The igniters are designed to intiate hydrogen gas burning at low concentrations thereby preventing gas buildup to the detonable range. The igniters work in combination with other containment systems, including the ice condenser system, the containment air return

system, the hydrogen skimmer system and the containment spray system. (Testimony of David L. Canup following Tr. 3488).

C. Ignition and Burning of Hydrogen

46. Applicant's witnesses testified that the ignition and burning of hydrogen generated in a TMI-type accident will occur by: (a) a continuous burn at the top of the ice condensers; (b) a series of burns initiated in the lower containment; or (c) a combination of (a) and (b). They further said that the Case (a) scenario is the likely scenario and the pressure rise in containment resulting from that would be only a few psi. (Tr. 3353-7). Of the three scenarios, the peak pressure in containment would result from a Case (b) scenario (multiple burns of hydrogen in the lower compartment). (Testimony of William Rasin, David Goeser, Bela Karlovitz, Bernard Lewis and Edward McHale Regarding Hydrogen Generation and Ignition following Tr. 3144 (Lewis Panel) at 10).

47. Applicant analyzed the Case (b) scenario. (*Id.* at 2, 3). The actual conditions assumed by Applicant for this scenario are described in the testimony and involve a small-break loss-of-coolant accident assuming failure of ECCS and a 75-80% zirconium-water reaction (1550 pounds of hydrogen). *Id.;* Applicant's Ex. 5A at 2-3 through 2-5 and Ex. 5B at Table 6, Accident Scenario JVD 12; Tr. 3202-3).

48. Applicant's analysis of the Case (b) scenario, the most severe of the three cases considered, resulted in a peak pressure of less than 16 psig. (Lewis Panel at 2, 3). This is considerably below the containment shell functional capability as determined by three independent structural analyses. See Containment Structural Capability §IV. A. *Supra.*

D. Transitions to Detonation

49. Steam inerting of hydrogen and transitions to detonation received considerable attention. Certain of the results of hydrogen ignition tests conducted by Livermore, raised the possibility of hydrogen not burning in lower steam concentrations than generally reported. (Tr. 3210-18; Staff Ex. K following Tr. 4353 at 15, 16). Dr. Marshall Berman of Sandia National Laboratory, a Staff consultant voiced concern over the possible steam inerting of the lower compartment, followed by removal of the steam in the ice condensers, resulting in high concentrations of hydrogen in the ice condenser possibly resulting in detonable mixtures. (Tr. 4083). Below about 18 volume percent, hydrogen is not detonable. (Tr. 3155, 3260). Both concentration and geometry are involved in detonation considerations. CESG maintained that the configuration of the ice condenser was such that analogies could be made to situations where detonations have been observed to occur. References were made to the concern of Dr. Roger A.

Strehlow (Tr. 3412) and through Dr. Berman, the work of Dr. John Lee of McGill University, a Sandia consultant. (Tr. 4199-200).

50. Applicant's testimony regarding Dr. Strehlow's concern clearly points out that even if higher concentrations were present, the geometry necessary for transition to detonation does not exist in the ice condenser. The testimony reflects that there are no long, narrow confined passageways in the ice condenser; rather the area between baskets is open, there are holes along the baskets and that the lattice frame would not confine any substance from flowing through the condenser. (Tr. 3489-94). CESG also raised the possibility of transition to detonation in the ductwork of the air handling units. The record reflects that the necessary geometry is not there. (Tr. 3613-7).

51. Dr. Berman's concerns of detonations in the upper plenum were strongly influenced by Dr. John Lee's experiments using stoichiometric concentrations of combustible gas (propane or methane) in large tubes open at one end with periodic obstacles (baffles at various distances) inside the tube wherein substantial overpressures (detonations) occurred. (Tr. 4095-97 as modified at 4243). Applicant's Witnesses, Lewis and Karlovitz, after having consulted with Dr. Lee by telephone, testified that the open, unconfined geometry of the upper plenum of the ice condenser was such that the results produced by Dr. Lee could not be produced in the upper plenum region. (Tr. 5050, See also 5057-8). Dr. Lewis further stated that if concentrations of hydrogen and steam in the lower compartment began to increase rapidly such that steam inerting occurred, the resulting concentration of hydrogen flowing through the ice condenser would increase correspondingly. (Tr. 5051-6, 5102). When the hydrogen/air mixture passing through the ice condenser enters the upper plenum and reaches flammable concentrations, ignition and burning would occur. As the concentration of hydrogen flowing through the ice condenser increased, the flame would settle into the ice condenser at a level where such mixture was just flammable and continue to burn. Thus, due to the presence of igniters in the upper plenum regions there would never be a detonable mixture of high hydrogen concentration in the upper plenum region. (Id.) The scenario posed by Dr. Berman and his interpretation of Dr. Lee's work is not probable on the bases of hydrogen concentration or necessary geometry. (Tr. 5050-1).

E. Pyrolysis of Polyurethane Foam

52. CESG raised a question concerning the effect of pyrolysis on decomposition of polyurethane foam insulation as a result of hydrogen burning in the ice condenser. The ice condenser is insulated from the

containment wall with 27,000 pounds of polyurethane covered with sheet steel.

Applicant presented a panel of witnesses consisting of Dr. Lewis, Mr. Rasin and Dr. Leonard S. Edelman. Dr. Lewis testified that, assuming a TMI-type accident which resulted in generation of hydrogen and a continuous burn in the ice condenser, the flame temperature inside the ice condenser would be theoretically about 1400°F (an $8\frac{1}{2}$% hydrogen concentration) with a flame height of at most about 1 centimeter. Within a few feet upward the hot gases would have cooled to about 400-500°F. Dr. Edleman testified to the characteristics of the polyurethane and the effect of heat on the foam. Applicant performed a conservative heat transfer analysis using the flame and hot gas temperature noted by Dr. Lewis. The analysis was conservative because it ignored the effect of ice in the condenser and assumed a 6 inch flame. The result was the volatization of 250 pounds of foam and the addition of 3×10^6 Btu of heat energy, which would have an insignificant effect on containment pressure. (Tr. 5136). Applicant testified that if the foam was not enclosed but totally exposed to oxygen and a flame, it would burn as long as there was sufficient oxygen to support combustion. In the sealed configuration at McGuire there is little oxygen to support combustion and the foam itself does not generate free oxygen for such support. An experiment conducted in such a configuration resulted in an inability to sustain combustion even when a large opening was present. (Tr. 5041, 5068-77, 5106-41, 5145-60, 5180-92, 5215-23).

53. In cross-examination, CESG determined that the amount of oxygen initially present in the containment is only sufficient to burn about 9,000 lbs. of the 27,000 lbs. of the polyurethane initially present in the containment. The net volume of the combustion gas would be 100,000 cf at standard conditions. If the heat of combustion were removed, the presence of this gas would cause a pressure increase of about 8% in the containment. If, instead of burning, the polyurethane were to completely gasify, it would result in 250,000 cf at standard conditions and an increase in containment pressure of about 20%, about 3 psi. The air handling ducts are not leak-tight and some of the volatilized gases from the pyrolysis were assumed to be released into the ice condenser atmosphere where they could contribute to the burning already occurring there. The heat of the combustion of the foam at 12,000 Btu's/lb was used as an approximation of the energy contribution. The resulting total energy contribution to the containment would be a maximum of 3 million Btu's, which is compared to the 80 million Btu's contributed by hydrogen burning under the S2D scenario. This additional energy contribution will not significantly increase the total pressure rise in containment. (Tr. 5121-60, 5128-25).

54. At the close of the evidentiary hearing, the Staff asked for additional time to review the record on pyrolysis and, if necessary, file testimony on that subject. The Staff filed an affidavit which has been admitted as Staff Exhibit Q. The Board permitted the parties the opportunity to respond to Staff Exhibit Q. Applicant filed an affidavit which has been admitted as Duke Ex. 9. CESG filed an affidavit (Ex. 63) on the same date as Staff but not in response to Staff Ex. Q. In Ex. Q, the Staff found the information sufficiently complete in the record and did not have substantial information to add. The Staff has no further concerns about the pyrolysis of foam. (Tr. 5245-6, 5252).

V. 10 CFR Part 2 - Appendix B

55. The Atomic Safety and Licensing Board has heard and decided, as necessary, all issues that have come before it. But for the provisions of Appendix B to 10 CFR Part 2, we would authorize the Director, Office of Nuclear Reactor Regulation, upon making requisite findings with respect to matters not embraced in our Intial Decision of April 18, 1979 or in this Supplemental Initial Decision, to issue full-term, full-power, operating licenses to McGuire Nuclear Station, Units 1 and 2.

56. In our analysis of the evidence, we have not identified any serious, close questions which we believe may be crucial to whether a license should become effective before full appellate review is completed. However, the Board has heard considerable testimony regarding ongoing research concerning hydrogen mitigation systems and has taken official notice of the contents of the program instituted by TVA to demonstrate by January 31, 1982 that an adequate hydrogen control system is installed at the Sequoyah facility and will perform its intended function in a manner that provides adequate safety margins.* The staff has taken the position that this requirement should also apply to McGuire, and we concur that such a requirement is appropriate with respect to the matters which we have considered in this proceeding. Since the decision to be reached by January 31, 1982, however, also entails matters beyond the scope of this proceeding (e.g., consideration of results of refined CLASIX calculations, results of verifications of equipment survivability, etc.), we consider such a requirement to be within the purview of the Director of the Office of Nuclear Reactor Regulation.

* "Research Program on Hydrogen Combustion and Control Quarterly Progress Report", Tennessee Valley Authority - Sequoyah Nuclear Plant, December 15, 1980 (Tr. 5227).

VI. CONCLUSIONS OF LAW

57. In an operating license proceeding, the Board is called upon to decide only the issues in controversy among the parties. 10 CFR §2.760a. Other matters required to be determined prior to the issuance of an amendment to the zero-power operating license for Unit 1 authorizing full-power operation or of an operating license for Unit 2 are entrusted to the Director of the Office of Nuclear Reactor Regulation. 10 CFR §§2.760a, 50.57.

58. Based upon the foregoing findings of fact, which are supported by reliable, probative, and substantial evidence in the record, and upon consideration of the entire evidentiary record in this reopened proceeding, the Board makes the following Conclusions of Law in supplementation of the Conclusions of Law reached in its April 18, 1979 Initial Decision:

(1) There is reasonable assurance that in the event of a TMI-type accident at McGuire, substantial quantities of hydrogen (in excess of the design basis of 10 CFR §50.44) will not be generated.

(2) As to Contentions 1 and 2, the actions taken and the procedures adopted by Duke Power Company subsequent to the TMI accident, provide reasonable assurance that (a) in the event of a TMI-type accident at McGuire, the likelihood of ECCS operations being prematurely terminated by the control room operating staff is so remote that such an accident scenario is not credible; (b) in the unlikely event of premature termination of the ECCS, operations will be reinitiated within sufficient time to prevent the generation of hydrogen in excess of 10 CFR §50.44; and (c) the McGuire facility can be operated without undue risk to the public health and safety with respect to possible hydrogen generation resulting from accidents of the type which occurred at TMI-2.

(3) Because the Board has found that quantities of hydrogen in excess of the design basis of 10 CFR §50.44 will not be generated, breach of containment and offsite doses in excess of 10 CFR Part 100 guideline values resulting from hydrogen combustion in a TMI-type accident at McGuire are not credible events. Accordingly, the premise for CESG Contentions 3 and 4 has not been established and there is no need to make specific findings with respect to those contentions.

(4) The NRC Staff has issued Supplement 3 to the McGuire Safety Evaluation Report (Staff Exhibit H) which addresses the significance of the unresolved generic safety issues as they relate to the

McGuire facilities and has provided a reasonable foundation for its several conclusions in conformity with the Board's April 18, 1979 Initial Decision.

VII. ORDER

WHEREFORE, IT IS ORDERED that the stay of the Licensing Board's April 18, 1979 Initial Decision is lifted and the Director, Office of Nuclear Reactor Regulation, is authorized, upon making requisite findings with respect to matters not embraced in the Initial Decision of April 18, 1979 or this Supplemental Initial Decision, in accordance with the Commission's regulations, to issue to Duke Power Company operating licenses (or in the case of Unit 1, an amendment to NPF-9, if appropriate) for a term of not more than forty (40) years, authorizing operation of the McGuire Nuclear Station, Units 1 and 2, at steady state power levels not to exceed 3,411 megawatts thermal; such licenses may be in such form and content as is appropriate in light of such findings.

In view of the Commission's Rules of Practice limiting the Board's jurisdiction in a contested operating license proceeding, the Board has made findings of fact and conclusions of law on matters actually put into controversy by the parties to the proceeding. In addition, the licenses will not be issued until the Director, Office of Nuclear Reactor Regulation has made the findings reflecting its review of the application under the Atomic Energy Act, which will be set forth in the proposed licenses, and has concluded that the issuance of the licenses will not be inimical to the common defense and security and to the health and safety of the public. Further, the license will not be issued until directed by the Commission after the appropriate Appendix B to 10 CFR Part 2 stay review process, if such is applicable.

Exceptions to the Initial Decision of April 18, 1979 and to this Supplemental Initial Decision and requests for a stay may be filed within 10 days after the service of the Supplemental Initial Decision. A brief in support of the exceptions should be filed within 30 days thereafter (40 days in the case of the Staff). Within 30 days after the service of the brief of appellant (40 days in the case of the Staff) any other party may file a brief in support of, or in opposition to, the exceptions.

THE ATOMIC SAFETY AND LICENSING BOARD

Emmeth A. Luebke
ADMINISTRATIVE JUDGE

Richard F. Cole
ADMINISTRATIVE JUDGE

Robert M. Lazo, Chariman
ADMINISTRATIVE JUDGE

Issued at Bethesda, Maryland,
this 26th day of May, 1981

[Appendix A has been deleted from this publication
but is available at the
NRC Public Document Room,
1717 H Street, N.W.,
Washington, D.C.]

Cite as 13 NRC 677 (1981) LBP-81-14

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
Marshall E. Miller, Chairman
Dr. Emmeth A. Luebke
Dr. Oscar H. Paris

In the Matter of **Docket Nos. 50-250-SP**
50-251-SP
(Proposed Amendment to Facility Operating License to Permit Steam Generator Repairs)

FLORIDA POWER AND LIGHT COMPANY
(Turkey Point Nuclear Generating Station, Units 3 and 4) **May 28, 1981**

The Licensing Board grants the staff's motions for summary disposition of all of intervenor's admitted contentions and cancels the evidentiary hearing originally scheduled to consider the licensee's proposal to repair its steam generators. The Board retains jurisdiction over the subject of radioactive solid wastes produced in connection with the repairs, directing the parties to furnish it with relevant information and recommendations.

NEPA: SCOPE OF REVIEW

The scope of a NEPA environmental review performed in connection with a nuclear facility license amendment is somewhat limited; the analysis is directed to a consideration of the extent to which the action under the proposed amendment will lead to environmental impacts beyond those previously evaluated.

NEPA: PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

Individual actions which have discrete and readily discernible local effects and are not part of a comprehensive federal proposal or national program do not require a programmatic environmental impact study.

RULES OF PRACTICE: INTERVENTION PETITIONS (PLEADING REQUIREMENTS)

Under 10 CFR § 2.714(b), an intervention petition must include the bases for each contention set forth with reasonable specificity. Contentions must be sufficiently detailed and specific to demonstrate that the issues raised are admissible and that further inquiry is warranted, and to put the other parties on notice as to what they will have to defend against or oppose.

RULES OF PRACTICE: SUMMARY DISPOSITION

Under 10 CFR § 2.749, once a motion for summary disposition has been made and supported by affidavit, the opposing party may not rely on mere allegations, but rather must demonstrate by affidavit or otherwise that a genuine issue exists as to a material fact.

NEPA: RULE OF REASON

The environmental review mandated by NEPA is subject to a rule of reason; it need not include environmental matters which are only remote and speculative possibilities. *Vermont Yankee Nuclear Power Corp. v. NRDC,* 435 U.S. 519, 551 (1978); *NRDC* v. *Morton,* 458 F.2d 827, 837-38 (D.C. Cir. 1972).

NEPA: SCOPE OF REVIEW

The environmental analysis of a license amendment is focused only upon the changes arising from the amendment. The consideration of alternatives in a license amendment proceeding does not include the evaluation of alternatives to the continued operation of the plant, even though the amendment might be necessary to enable continued reactor operation.

TECHNICAL ISSUES DISCUSSED:

Steam generator degradation;
Steam generator repair procedures;
Occupational radiation exposure;
Steam generator storage;
Steam generator repair waste storage;
Design basis tornado;
Probable maximum hurricane.

MEMORANDUM AND ORDER
(Granting Summary Disposition of All Contentions, and Canceling Evidentiary Hearing)

This proceeding involves a proposed program for the repair of steam generators at Turkey Point Nuclear Units 3 and 4. The Nuclear Regulatory Commission (NRC) gave notice on December 13, 1977, that it was considering license[1] amendments which would "authorize the licensee to repair the steam generators now in use at each facility, replacing major portions of such steam generators with new components, and to return the units to operation using the steam generators, so repaired."[2]

Any person whose interest may be affected was given an opportunity to intervene by filing a request for a hearing in the form of a petition for leave to intervene, by January 13, 1978. Such petitions to intervene were stated to be governed by 10 CFR §2.714, and were required to identify "the specific aspect or aspects of the subject matter of the proceeding as to which he wishes to intervene and setting forth with particularity both the facts pertaining to this interest and the basis for his contentions with regard to each aspect on which he desires to intervene."[3]

The *Federal Register* notice establishing an opportunity for hearing on the proposed issuance of amendments to a facility operating license further stated:

> "Contentions shall be limited to the matters within the scope of the amendments under consideration. A petition that sets forth contentions relating only to matters outside the scope of the amendments under consideration will be denied."[4]

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[1] Facility Operating Licenses Nos. DPR-31 and DPR-41.
[2] 42 *Fed. Reg.* 62569.
[3] *Id.*
[4] *Id.*

No petitions for leave to intervene were filed during the 30-day period established by the notice. On February 9, 1979, more than a year after the expiration of the intervention period, Mark P. Oncavage (Intervenor) filed an untimely request for a "full hearing." After receiving numerous filings, responses by the Staff and the Licensee (FPL), and amendments, a divided Board ruled that after balancing the five factors set forth in 10 CFR §2.714(a)(1) for considering nontimely petitions, the intervention petition would be allowed.[5]

After receiving various filings, the Board entered an Order Relative to Contentions and Discovery on September 25, 1979. This Order clarified the language in the admitted contentions and ruled on the remaining contentions. Revised Contention 1 was stated to read as follows:

> "Section 102(2)(C) of the National Environmental Policy Act (42 U.S.C. §4332(2)(C) or 10 CFR §51.5 requires the preparation of an Environmental Impact Statement prior to the issuance by the Nuclear Regulatory Commission of amendments to the operating licenses for Turkey Point Units Nos. 3 and 4 (Facility Operating Licenses Nos. DPR-31 and DPR-41) authorizing the Licensee to repair the steam generators now in use in each facility."

The Staff at that time took the view that an environmental impact statement (EIS) was not required under the National Environmental Policy Act (NEPA)[6] and 10 CFR Part 51, and that an environmental impact appraisal (EIA) would be adequate. On June 29, 1979, the Staff issued an EIA with appropriate notice to the public. However, the Staff subsequently decided to prepare an EIS as a matter of discretion, following a Commission Memorandum and Order directing the issuance of an EIS in connection with the Surry steam generator repairs.[7] In December, 1980, the Staff issued its Draft Environmental Statement (DES) and circulated it for comment. The Final Environmental Statement (FES) was issued as NUREG-0743 in March, 1981.

A prehearing conference was held March 24, 1981, for the purpose of establishing, with precision and finality, the contentions which would frame the issues for trial. The Chairman of the Board requested counsel for each party to address the viability and phrasing of each contention, in order to

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[5]10 NRC 183 (1979). A dissenting opinion was filed by one Board Member (10 NRC at 211-12), and separate opinions on the weight to be given Factor (iii) were filed by the other two Board Members (10 NRC at 193 & 200).

[6]The National Environmental Policy Act of 1969, Pub L. No. 91-190, 83 Stat 852 as amended by Pub. L. 94-83, 89 Stat 424, 42 U.S.C. §§4321 *et seq.*

[7]Virginia Electric Power Co. (Surry Nuclear Power Station, Units 1 and 2), CLI-80-4, 11 NRC 405 (1980). *See also* letter from Staff counsel to the Board, dated March 6, 1980.

avoid having the parties "coming in with issues or new matters in an untimely fashion,"[8] to determine which issues were still viable, and "in order to have in one place the precisely phrased contentions that we are going to trial on."[9] The Intervenor's contentions were then renumbered and read into the record, and also set forth expressly in the prehearing order as "currently refined or revised."[10] Those contentions as thus set forth with finality, and those contentions alone, control the issues to be adjudicated in this proceeding. The only possible exception is the leave granted to the Intervenor "to file on or before April 20, 1981, appropriate amendments to Contention 1 in order to plead with specificity the respects in which the FES (due to be filed by the Staff by April 1) does not legally or factually comply with NEPA (Tr. 36, 38-9, 43)."[11] The filings made by the Intervenor regarding Contention 1 pursuant to this order, and the responsive motions and answers filed by the other parties, are discussed more fully *infra* at pages 7-8, 14, 24-28.

Summary disposition motions were filed and, without opposition by the Intervenor, were granted as to Contention 14 as originally numbered,[12] and Contentions 2, 3, 5, 6, 7, 8, as renumbered.[13] Summary disposition of Contention 4A was granted by our Order entered May 7, 1981. That leaves for consideration in this proceeding only the amendments to Contention 1, and Contention 4B.

I. DESCRIPTION OF PROPOSED STEAM GENERATOR REPAIRS

The six steam generators at Turkey Point Units 3 and 4 have all undergone a significant amount of degradation since they began operation in 1972 and 1973, respectively. The wastage and denting phenomena have led to the tube wall thinning, support plate flow slot hourglassing and plate ligament cracking, tube denting, stress corrosion cracking and several instances of reactor coolant leakage through cracked tubes. As of November, 1980, tube plugging for various reasons has resulted in removing about 20% of the steam generator tubes in Unit 3 and about 24% of the tubes in Unit 4 from continuing service. Additional plugging would

[8]Transcript of Prehearing Conference held in Homestead, Florida on March 24, 1981 (Tr.) at 5.
[9]Tr. 6-7.
[10]Memorandum and Order entered April 2, 1981, pp. 2-5.
[11]*Id.*, at 3-4.
[12]*Id.*, at 5-6.
[13]Memorandum and Order (Granting Motions for Summary Disposition), entered April 29, 1981, at p. 2.

result in operating at a reduced power rating and at an economic disadvantage.

FPL plans to repair all six steam generators in Turkey Point Units 3 and 4. The Unit 4 steam generators have the most tubes plugged and therefore, would be repaired first. The repair of Turkey Point Unit 3 steam generators is expected to begin about one year later. Since FPL experiences operating peaks of longer duration in the summer, and the repair is expected to take from six to nine months per unit, the repair should be started in the fall to be completed before the next summer peak demand.

The proposed repairs will consist of replacing the lower assembly of each steam generator, including the shell and the tube bundle, and refurbishing and partially replacing the steam separation equipment in the upper assembly. Prior to the repair work, the unit will be shut down and all systems will be placed in condition for long-term shutdown. The reactor vessel head will be removed for defueling. All of the normal procedures for fuel cooling and fuel removal will be followed. The fuel will be removed from the reactor and placed in the spent fuel storage facility, and then the reactor vessel head will be replaced.

The equipment hatch will be opened and access control will be established. A special curtain, which would be able to reduce the size of the opening in the containment in case of an accident, will be installed in place of the door for ease of deployment. A special vent exhausting through an HEPA filter will be constructed. The biological shield wall and a section of the operating floor concrete and structural steel will be removed to provide access to the steam generator. Guide rails will be installed for transporting the lower assembly through the equipment hatch.

After this preparatory work, the cutting of system piping will begin. This will include cutting and removal of sections of steam lines, feedwater lines, and miscellaneous smaller lines for the service air and water and the instrumentation system. The steam generator will then be cut at the transition cone, and the upper shell will be removed and will be refurbished inside containment. After the channel cut at the bottom, the lower assembly will be lifted from its support to the working level where it will be welded shut.

Following this, the steam generator lower assembly will be lowered and placed in position on a transport mechanism. This mechanism will carry the assembly through the equipment hatch. A transporter will carry it to the steam generator storage facility on the site. The other two steam generator lower assemblies will be lifted from their location, welded shut, and lowered through the same hatch where the first steam generator was removed.

After removal and storage of all three steam generator lower assemblies, their replacements will be transported from the temporary storage location to the equipment hatch. The same machinery used to remove the lower assemblies will be used to install the new assemblies in their cubicles. The steam generator lower assembly will be reinstalled and rewelded to the old bottom section. The upper assembly with its refurbished internals will be mounted on the lower assembly. After welding the two assemblies together, the piping will be reconstructed. Following these major repair activities, there will be cleaning, hydrostatic testing, baseline inservice inspections, and preoperational testing of instruments, components and systems. The reactor will then be refueled and startup tests will be performed. The performance of the repaired steam generators will be tested for moisture carryover and verification of thermal and hydraulic characteristics (NUREG-0743, Final Environmental Statement, March, 1981 at 1-1 to 3-4).

II. CONTENTION 1

The Intervenor's "Amendment to Contention 1", Filed April 20, 1981, consists of 17 numbered amendments to the original contention, which purport to "plead with specificity the respects in which the FES....does not legally or factually comply with NEPA."[14] The Staff filed its Objections to Proposed Amended Contention 1 and Third Motion for Summary Disposition on April 27, 1981. The Staff opposed the proposed amendments on both procedural and substantive grounds, asserting that they failed to plead with specificity the respects in which the FES did not comply legally or factually with NEPA. It also asserted that such pleadings presented no genuine issues of material fact warranting adjudication, and sought summary disposition under 10 CFR §2.749. The Licensee filed a response in support of the Staff's objections and motion for summary disposition on April 30, 1981. The Intervenor filed an Answer Opposing the Motion for Summary Judgment on May 19, 1981.

The Intervenor's numbered amendments to Contention 1 will be considered *seriatim*, regarding both their adequacy as contentions and their viability when challenged by the Staff's motions for summary disposition.

The first two amendments assert that the Staff has failed to comply with two provisions[15] in the 1978 guidelines of the Council on Environmental Quality (CEQ).

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[14]Tr. 27-28, 35; Memorandum and Order, dated April 2, 1981, p. 4.
[15]40 CFR §§1501.7 and 1505.2.

Amendment 1 states:

The EIS failed to follow section 1501.7 of the NEPA regulations in tha the Staff failed to invite interested persons to participate in a scopin process in which the scope of the EIS was to be decided.

Amendment 2 states:

No record of decision was prepared for the Turkey Point Project i violation of 40 CFR 1505.2.

The Commission's own regulations implementing NEPA are set fort in 10 CFR Part 51. The Commission has consistently taken the positio that the substantive requirements of the CEQ guidelines are not bindin upon the NRC because it is an independent regulatory agency.[16] Th Executive Order issued by the President stated generally that federa agencies shall comply with the regulations issued by CEQ "except wher such compliance would be inconsistent with statutory requirements."[17] Th Commission has proposed revisions in 10 CFR Part 51 which voluntaril take the CEQ guidelines into account, but until the proposed revisions ar adopted, the present regulations remain in effect.[18] A final rule has not ye been adopted by the Commission. Accordingly, the Staff was governed b the provisions of 10 CFR Part 51, not the CEQ regulations as alleged by th Intervenor, in preparing and issuing a Final Environmental Statement.

Moreover, the Intervenor and the public have had extensive opportunities for input to the environmental review process in this proceeding including the scope of the Environmental Impact Statement suggested b CEQ guidelines (40 CFR §1501.7). On June 29, 1979, the Staff issued a Environmental Impact Appraisal with appropriate notice to the public. I December, 1980, the Staff issued its Draft Environmental Statement fo public comment. A large number of comments including those of th Intervenor were received and were specifically addressed by the Staff in it FES, which was issued as NUREG-0743 in March, 1981.[19]

The scope of a NEPA environmental review performed in connectio with a nuclear facility license amendment is somewhat limited, and it is no as broad as that conducted in the prior NRC licensing proceedings. Such a analysis is directed to a consideration of the extent to which the actio

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[16] May 31, 1979 letter from NRC Chairman Joseph M. Hendrie to Charles H. Warren, Chairman CEQ (Attachment to Staff's Motion dated April 27, 1981).
[17] Executive Order No. 11, 991 (3 CFR 123), reprinted 42 U.S.C. §4231 (1977).
[18] *Fed. Reg.* 13739-40 (March 3, 1980).
[19] FES at 8-1 to 8-26.

under the proposed amendment will lead to environmental impacts beyond those previously evaluated.[20] The Appeal Board in this regard has stated:

> "Nothing in NEPA or in those judicial decisions to which our attention has been directed dictates that the same ground be wholly replowed in connection with a proposed [license] amendment.... Rather, it seems manifest to us that all that need be undertaken is a consideration of whether the amendment itself would bring about significant environmental consequences beyond those previously assessed and, if so, whether those consequences (to the extent unavoidable) would be sufficient on balance to require a denial of the amendment application. This is true irrespective of whether, by happenstance, the particular amendment is necessary in order to enable continued reactor operation...."[21]

Accordingly, in this case the scope of environmental review does not extend to a reconsideration of the impacts of the continued operation or alternatives to such operation of Turkey Point, as they have been previously assessed in NRC licensing proceedings. Such avoidance of replowing the same ground applies to a reconsideration of alternative energy sources, or energy reduction measures, including conservation.

The FES prepared and filed by the Staff in March, 1981 (NUREG-0743), contains a description of the proposed steam generator repair method (§3), as well as an evaluation of its environmental effects, alternatives thereto, and postulated accidents (§§4, 5 and 6). It contains a reasoned consideration of all comments received on the DES, including those made by the Intervenor (§8). The FES concludes that the proposed action will not significantly affect the quality of the environment, that its benefits outweigh the costs, and that the overall cost benefit would not be improved by any of the alternatives (§6). The scope of the FES therefore encompasses the environmental impact analysis required by NEPA and implemented by 10 CFR Part 51.

Section 1505.2 of the CEQ guidelines, regarding the preparation of a public record of an agency's decision, is not applicable under Amendment 2 because an agency decision is not made by the Staff. That adjudicatory decision is made for the agency by a Licensing Board, subject to review by the Appeal Board and by the Commission itself. As stated by the Staff, it has made its recommendations and believes that an adequate record has

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[20]Consumers Power Co. (Big Rock Point Nuclear Plant), ALAB-636, (March 31, 1981); Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451 (1980).

[21]Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 46 fn. 4 (1978).

been developed for a favorable decision on the FES by the Board. For the foregoing reasons, Amendments 1 and 2 do not plead cognizable contentions, and they are also subject to summary disposition.

Amendment 3 alleges that programmatic EIS is required "as a result of the steam generator repairs that would be required nationally." However no legal or factual basis is shown for such a conclusion. The instant steam generator repairs are not part of a comprehensive federal proposal or national program which would require a programmatic NEPA review. The environmental impacts associated with the Turkey Point repairs will only occur on a local, not a national basis.[22] Such individual actions with discrete and readily discernible local effects do not require a programmatic environmental impact study.[23] Amendment 3 does not start a cognizable contention, and it is also subject to summary disposition.

Amendments 4 and 15 involve essentially the same subjects and therefore will be considered together.

Amendment 4 states:

> The final EIS fails to comply with NEPA in that the EIS does not address (to the fullest extent possible) all environmental effects of proposed actions as well as all irreversible and irretrievable resources.

Amendment 15 states:

> The EIS fails to discuss the irreversible and irretrievable commitment of resources in the proposed action.

These amendments merely refer generally to some phrases taken from Section 102 of NEPA, but fail to relate them to the Turkey Point steam generator repairs in any meaningful manner. There is no specificity or concreteness as to the way in which "environmental effects" or "irreversible and irretrievable commitment of resources" were allegedly not properly addressed by the Staff in the FES.

Under 10 CFR §2.714(b), an intervention petition must include "...the bases for each contention set forth with reasonable specificity." This requirement of pleading with particularity and specificity was also set forth in the notice of opportunity for hearing on the Turkey Point proposed license amendment, *supra* page 2 (42 *Fed. Reg.* 62569). These basic requirements make it incumbent upon intervenors to set forth contentions

[22]Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 267-8 (1979); Virginia Electric and Power Company (Surry Nuclear Power Station, Units 1 and 2), DD-79-19, 10 NRC 625, 639-42 (1979).
[23]Kleppe v. Sierra Club, 427 U.S. 390, 399, 402, 410 (1976).

which are sufficiently detailed and specific to demonstrate that the issues raised are admissible and that further inquiry is warranted, and to put the other parties on notice as to what they will have to defend against or oppose.[24] Although intervenors are not required to plead evidence, it is nevertheless necessary for contentions to set forth the reason or bases for their assertions with reasonable particularization or specificity.[25]

Mississippi Power and Light Company (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 AEC 423, 425-26 (1973) does not hold, as the Intervenor argues, that a contention fulfilled the particularity requirement by stating that "the alternatives of conserving electricity or utilizing other methods of producing energy have not been adequately considered." If the Intervenor's "Response to NRC Staff Objections to Proposed Amended Contention 1 and Licensee's Motion to Dismiss Contention 1", p. 3 had merely continued this quotation from Grand Gulf, it would then have read as follows:

> "At the prehearing conference, petitioner's counsel stated that the *basis for that contention* is that the amounts expended by the applicant on advertising greatly exceeded (by a factor of 11) that devoted to research and development, and that he intended 'to introduce evidence that there are geothermal sources in the Middle South Utilities System area that could be utilized' (Tr. 66-67). We agree with the Licensing Board *that, given this particularization,* the contention is adequate." (6 AEC at 426) (Emphasis added)

It should be noted by contrast that in the instant proceeding, the Intervenor failed to particularize Contention 1 at the Prehearing Conference on March 24, 1981, although repeatedly invited to do so by the Board (Tr. 12-15, 24, 26-30, 34-36, 43-45). A subsequent prehearing conference scheduled for April 27-28, 1981, was canceled upon motion of counsel for the Intervenor on the stated grounds that is "was no longer necessary."[26]

Of course under 10 CFR §2.749, once a motion for summary disposition has been made and supported by affidavit, the opposing party may not rely on mere allegations, but rather must demonstrate by affidavit or otherwise that a genuine issue exists as to a material fact *(Virginia Electric and Power Company* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451, 453 (1980)).

[24]BPI v. Atomic Energy Commission, 502 F.2d 424, 429 (D.C. Cir. 1974); Duquesne Light Co. (Beaver Valley, Unit No. 1), ALAB-109, 6 AEC 242, 245 (1973); Philadelphia Electric Co. (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-216, 8 AEC 13, 20-21 (1974).
[25]Houston Lighting and Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542, 547-9 (1980).
[26]Memorandum and Order (Canceling Prehearing Conference), entered April 23, 1981.

Amendment 4 also asserts that the FES is defective because it does nc address "all" environmental effects of proposed actions. This is not correct statement of the applicable law. The environmental revie mandated by NEPA is subject to a "rule of reason", and it need not includ review of environmental matters which are only remote and speculativ possibilities.[27] The Appeal Board has held that environmental impac statements need not discuss remote and speculative environmental impact of the proposed project itself,[28] quoting with approval the followin statement by the Court of Appeals in *Trout Unlimited:*

> "An EIS need not discuss remote and highly speculative conse quences.... A reasonably thorough discussion of the significant aspect of the probable environmental consequences is all that is required b an EIS."[29]

Finally, the FES in this proceeding does in fact contain a full and fai evaluation of the reasonably calculable environmental impacts of th proposed steam generator repairs (FES, §4; Appendices A-D). The FE also considers and discusses the irreversible and irretrievable commitmen of resources, in accordance with the requirements of NEPA (FES, §4.3.1) Amendments 4 and 15 are inadequate to plead contentions, and they ar subject to summary disposition.

Amendment 5 states:

> The EIS fails to look at the socio-economic effects upon Florida Powe and Light rate payers. Such effects must be examined fully within th EIS because the project entails direct significant environmental effect which are intertwined with the socio-economic effects.

This contention amounts to a generalized claim, without explanatio or definition, that the FES fails to consider the socio-economic effects upo the Licensee's ratepayers. In fact, the FES analyzes in some detail th economic costs of the Turkey Point steam generator repair project (§4.2) This study covers the costs of the repairs, and shows a substantial net dolla savings when repair costs are compared with the cost of continued

[27]Vermont Yankee Nuclear Power Corp. v. NRDC, 435 U.S. 519, 551 (1978); NRDC v Morton, 458 F.2d 827, 837-8 (D.C. Cir. 1972).

[28]Public Service Electric and Gas Company (Hope Creek Generating Station, Units 1 and 2) ALAB-518, 9 NRC 14, 38 (1979).

[29]Trout Unlimited v. Morton, 509 F.2d 1276 at 1283 (9th Cir. 1974). *Accord:* Environmental Defense Fund v. Hoffman, 566 F.2d 1060, 1067 (8th Cir. 1977); Concerned About Trident v. Rumsfeld, 555 F.2d 817, 828 (D.C. Cir. 1977); Sierra Club v. Hodel, 544 F.2d 1036, 1039 (9th Cir. 1976); Carolina Environmental Study Group v. United States, 510 F.2d 796, 799 (D.C. Cir. 1975).

›peration in a derated mode. The estimated net savings of $380,000,000 are ›ased largely on the costs of replacement capacity, which are described vith supporting data. The contention does not give a basis for nor any ›articularization of reasons for its bare assertions, contrary to the 'equirements of 10 CFR §2.714(b), discussed *supra*.

If this contention is intended in some manner to raise an issue over vho will bear the costs of the proposed repairs, that is a matter for the ıppropriate state agencies to decide, and it is beyond the scope of NRC urisdiction in this proceeding (FES, §8.6.24). Amendment 5 does not ıdequately plead a contention, and it is subject to summary disposition.

Amendment 6 states:

> The EIS contains no glossary or table of definitions and consistently uses terminology beyond the ken of lay people.

There is no NEPA requirement that an EIS must contain a glossary of erms. Steam generator repairs to a nuclear power plant obviously involve ›ome technical matters. However, the meaning of most terms in the FES ːan be determined from their context and relationship to the subjects liscussed. The courts have discussed this language problem as follows:

> "[An EIS] serves as an environmental full disclosure law, providing information which Congress thought the public should have concerning the particular environmental costs involved in a project. To that end, it 'must be written in language that is understandable to nontechnical minds and yet contain enough scientific reasoning to alert specialists to particular problems within the field of their expertise....' It cannot be composed of statements 'too vague, too general and too conclusory...'."[30]

The FES appears on its face to achieve the terminological balance ›ought between reasonably informing the public and yet alerting specialists o particular technical matters. None of the commentors on the DES, with he sole exception of the Intervenor, indicated any problems with lefinitions or the use of technical terms (FES, §§8.1, 8.6.3). Amendment 6 loes not adequately plead a cognizable contention, and it is subject to ›ummary disposition.

Amendment 7 asserts that the estimates of worker exposure provided in he FES "are unreasonably low". The Intervenor has failed to set forth any ›ases or reasons for this assertion; thus the contention fails to meet the

[30]Silva v. Lynn, 482 F.2d 1282, 1284-85 (1st Cir. 1973); Sierra Club v. Morton, 510 F.2d 813, ›20 (D.C. Cir. 1975).

requirement of 10 CFR §2.714(b) that the bases for a contention be stated with reasonable specificity. Moreover, the FES identifies the occupational radiation exposure associated with the proposed repair as the major environmental impact (See FES, §2.4). Occupational exposure was thoroughly and extensively addressed in the FES. The expected exposure was compared to the actual exposure which occurred during the steam generator repair at Surry, and adjusted upward in light of that experience. As a result of that upward adjustment, FPL changed its planned procedure so as to reduce occupational exposure (See FES, §§4.1.1 and 5). In addition the Intervenor addressed occupational exposure in his comments on the DES, and the Staff responded fully to those comments (See FES, §§8.6.8 and 8.6.13). Thus there is no genuine issue to be heard as to the facts set forth on occupational exposure in the FES, and Amendment 7 is subject to summary disposition.

Amendment 8 asserts that the analysis of deaths and health effects that are expected to result from the repair activity is based on "out-moded scientific information". Again, the Intervenor has failed to set forth the basis for this assertion and thus the contention fails to meet the requirements of 10 CFR §2.714(b). With regard to the facts, the health effects predicted in the FES are based on the 1972 report of the National Academy of Sciences' Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR Committee), "The Effect on Populations of Exposure to Low Levels of Ionizing Radiation" (See FES, §4.1.1.6). The 1972 BEIR report was updated by the more recent report, "The Effect on Populations of Exposure to Low Levels of Ionizing Radiation - 1980". This 1980 report is used as the basis for additional estimates presented in Appendix B of the FES (See FES, pp. B-1 through B-4).[31] Thus there is no genuine issue to be heard as to the facts with respect to this contention, and it is subject to summary disposition.

Amendment 9 states:

> The economic analysis in the EIS is invalid in that it fails to consider the possibility that replacement or repair of the steam generators may be necessary a second time.

In fact §3 of the FES does consider the possibility of the need to replace or repair the steam generators again, and concludes that "a number of changes have been made in the materials, the design, and the operating procedure for the replacement steam generators to assure that the corrosion

[31]The health effects of ionizing radiation predicted in the 1980 report by the BEIR Committee are less severe than those predicted by the 1972 report.

and denting problems will not recur." Section 6(3) states that the new steam generator design "incorporates features that will eliminate the potential for the various forms of tube degradation observed to date." In responding to the Intervenor's comments on the DES, the FES further states that it "is assumed that the life of the repair is the remainder of the plant life, or about 30 years. There is no guarantee of this plant life; however, the Staff safety review found no reason to doubt that the steam generators would last the life of the plant" (FES §8.6.24).

No basis has been shown for this contention. It should be noted that the Intervenor's original Contention 11(a) alleged that the Licensee had "failed to consider the cost of future recurring steam generator repairs." The Board rejected that contention then because it found "no basis for this speculation."[32] There is still no basis shown for such speculation. Amendment 9 does not adequately plead a cognizable contention, and it is subject to summary disposition.

Amendment 10 states:

> The entire EIS fails to comply with a good faith consideration as is required under NEPA.

This statement is wholly conclusory and without the allegation of any factual or other bases or reasons. It does not purport to raise any factual issue, and it lacks the specificity and particularization of reasons for its bare assertions required by 10 CFR §2.714(b). It is therefore not admissible as a contention. In addition, the FES contains a good faith, objective and reasonable consideration of the subject areas as mandated by NEPA.[33] Amendment 10 is subject to summary disposition.

Amendments 11 and 13 both purport to address the consideration of alternatives in the FES.

Amendment 11 states:

> The analysis of alternatives is inadequate under NEPA.

Amendment 13 states:

> The EIS fails to adequately discuss the alternatives to the proposed action.

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[32] Order Relating to Contentions and Discovery, dated September 25, 1979, at p. 5.
[33] Environmental Defense Fund, Inc. v. Andrus, 619 F.2d 1368, 1375-77 (10th Cir. 1980); Manygoats v. Kleppe, 558 F.2d 556, 560-61 (9th Cir. 1977).

These assertions are bare conclusions, devoid of any description of bases or reasons for the statements. There is no identification of any alternatives which should have been considered but were not. Neither is there any description of alleged inadequacies or deficiencies in the analysis of those alternatives which were considered in the FES. It has been held that the "discussion of environmental effects of all alternatives need not be exhaustive, but it must be such that sufficient information is contained therein to permit a 'rule of reason' designation of alternatives beyond the primary proposal."[34]

The Supreme Court has discussed this question of NEPA consideration of alternatives as follows:

> "[T]he term 'alternatives' is not self-defining.... Common sense also teaches us that the 'detailed statement of alternatives' cannot be found wanting simply because the agency failed to include every alternative device and thought conceivable by the mind of man.... It is still incumbent upon intervenors who wish to participate to structure their participation so that it is meaningful, so that it alerts the agency to the intervenor's position and contentions.... Indeed, administrative proceedings should not be a game or a forum to engage in unjustified obstructionism by making cryptic and obscure reference to matters that 'ought to be' considered and then, after failing to do more to bring the matter to the agency's attention, seeking to have the agency determination vacated on the ground that the agency failed to consider matters 'forcefully presented.' "[35]

Further, the FES in fact considers various alternatives in substantial detail, including continued operation without repair, replacement by plant of another design, or the chosen alternative of repair of generators (FES, §5; Table 5.1, Options considered). The steam generator repair alternatives which were analyzed included retubing (§5.2), tube sleeving (§5.3), replacement of entire generator (§5.4), and the proposed method of replacement of the lower assembly (Table 5.2). Six alternative methods for the disposal of the steam generator lower assemblies, which comprise the largest source of radioactive waste, were also analyzed (§5.5, Table 5.3). The Staff answered the comments of the Intervenor on the DES regarding alternatives, pointing out the range of reasonable alternatives it had considered, but noting that alternatives to plant operation itself, rather than

[34]Environmental Defense Fund, Inc. v. Andrus, *supra*, 619 F.2d at 1375. *See also* Natural Resources Defense Council, Inc. v. Morton, 458 F.2d 827, 836-7 (D.C. Cir. 1972).
[35]Vermont Yankee Nuclear Power Corp. v. Natural Resources Defense Council, 435 U.S. 519, 551-54 (1978).

the proposed repairs, were beyond the scope of required environmental review (FES §8.6.13). The FES therefore contains a good faith reasonable review of alternatives as required by NEPA. Amendments 11 and 13 do not adequately plead cognizable contentions, and they are subject to summary disposition.

Amendment 12 states:

> The final EIS as a whole fails to adequately address the impact of the steam generator repair on the human environment because it tends to explore the positive effects that the repair will have while down-playing the negative impact.

The bare assertion lacks the essential elements of pleading the bases of contentions with reasonable specificity, as required by 10 CFR §2.714(b). No issues are framed by this allegation. In addition, the FES makes a reasoned cost-benefit analysis showing that the benefits of the continued safe production of power for the public outweigh the described costs of the proposed repairs, both environmental and economic (FES, §6). It also shows that the overall cost benefit would not be improved by any of the alternatives *(Id.)*. Amendment 12 fails to plead an admissible contention, and it is subject to summary disposition.

Amendment 14 states:

> The EIS fails to adequately discuss the relationship between local short term use of man's environment and maintenance and enhancement of the long term productivity.

This contention lacks the requisite descriptions of bases with reasonable specificity, contrary to the requirements of 10 CFR §2.714(b). In addition, the Turkey Point plant site is the primary environmental resource involved in this proceeding, and it is and has been wholly dedicated to the nuclear generation of electricity. Such committed land usage was considered and approved in the operating license FES in 1972 (OL-FES, §VII). The instant proposed steam generator repairs do not change or materially alter the size, use or environmental impacts of this facility or its site. Amendment 14 does not plead a cognizable contention, and it is subject to summary disposition.

Amendment 16 states:

> The final EIS fails to adequately discuss the environmental impact of a hurricane if one occurs during the repair process.

We assume that Intervenor means to refer to the environmental impact resulting from the interaction of a hurricane with steam generator repair activities. We find *infra* with respect to Contention 4B that a hurricane during the repair activity would not be likely to cause a release of radioactivity to unrestricted areas. Moreover, Staff attested that the worst-case accident during the repair would not result in the release of radioactivity to unrestricted areas in excess of the limits imposed by 10 CFR Part 20. Were such an accident to occur during a hurricane, wind and turbulence would further reduce airborne concentrations (Staff Affidavit at 7). Thus there is no genuine issue to be heard as to the environmental impact of a hurricane interacting with repair activities, and Amendment 16 is subject to summary disposition.

Amendment 17 states:

The final EIS fails to consider the long term effects of a nuclear waste building next to biscayne bay (sic).

We assume that "nuclear waste building" refers to the steam generator storage compound (SGSC). We have already found, in granting summary disposition of Contention 4A, that the location and design of the SGSC would prevent damage to the SGLAs during storms. We also found that corrosion would not cause leaks to develop during the anticipated storage period on site (See Order dated May 7, 1981). Finally, we note specifically here that the SGSC will have a 6-inch thick concrete floor which would inhibit release of radioactive liquid, should it leak from the SGLAs (Staff Affidavit at 4). From these facts we conclude that there is no genuine issue to be heard as to the facts relating to long term effects of the SGSC next to Biscayne Bay, and that Amendment 17 is subject to summary disposition.

III. INTERVENOR'S ANSWER OPPOSING MOTION FOR SUMMARY DISPOSITION

The Intervenor on May 19, 1981 filed his Answer Opposing the Motion for Summary Disposition, which had been filed by the Staff on April 27, 1981. The Staff's motion had also opposed the Intervenor's April 20, 1981, proposed Amendment to Contention 1. The Intervenor on May 12, 1981, filed a pleading captioned "Response to NRC Staff Objections to Proposed Amended Contention 1 and Licensee's Motion to Dismiss Contention 1." Both the Staff and the Licensee on May 18 filed motions to strike this pleading on the grounds that it constituted an unauthorized reply to their answers to the proposed amendment to Contention 1, which were permitted by 10 CFR §2.714(c). Inasmuch as the Intervenor's answer to the summary disposition motion covers the points raised in his May 12 response to objections and an alleged motion, it is unnecessary to determine whether the pleadings previously filed by the Staff and the Licensee were motions, answers, objections or something else.

The Intervenor first argues that his Contention 1, although definitively read into the record by the Board[36] and stated with finality in our prehearing conference order,[37] nevertheless should be considered as including his original Contention 10. Contention 1, as set forth without objection in our prehearing conference Order entered April 2, 1981, read as follows:

> "Section 102(2)(C) of the National Environmental Policy Act (42 U.S.C. §4332(2)(C) or 10 CFR §51.5 requires the preparation of an Environmental Impact Statement prior to the issuance by the Nuclear Regulatory Commission of amendments to the operating licenses for Turkey Point Units Nos. 3 and 4 (Facility Operating Licenses Nos. DPR-31 and DPR-41) authorizing the Licensee to repair the steam generators now in use in each facility (Tr. 11-54)."[38]

> Original Contention 10 read as follows:

> "The Commission's NEPA Analysis is inadequate in that it fails to adequately consider the following alternative procedures:
> a. Arresting tube support plate corrosion
> b. In-place tube restoration (sleeving)
> c. In-place steam generator tube replacement (retubing)
> d. Derating
> e. Decommissioning
> f. Bioconversion
> g. Conservation
> h. Solar energy
> i. Natural gas
> j. Coal".

As discussed *supra* at pp. 3-4, it was intended that Contention 1, as phrased on March 24, 1981, was the only such contention before the Board. No mention was ever made, at the prehearing conference (March 24) or after the prehearing Order (April 2), that the Intervenor contended that original Contention 10 was included in or to be read with Contention 1 as

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[36]Tr. 5-7, 9, 11-15, 19-21, 24-28, 33-36, 43-44, 54.
[37]Memorandum and Order (Prehearing Conference, March 24-25, 1981), entered April 2, 1981, pp 3-4.
[38]*Id.*

rephrased. Counsel for the Intervenor was expressly told at the prehearing conference that "if you want to plead with some specificity now by rephrasing Contention 1 we would allow you to do so, but you persist in telling us you think that is sufficient. So, I am giving you warning, it is wholly lacking in specificity as a contention. And if you want to stand on it, do it at your peril."[39]

After some further colloquy, counsel for the Intervenor stated that after the FES was filed he was prepared "to file with the Board what issues — what contentions we intend to assert to prove that the final EIS does not legally and factually comply with NEPA...."[40] Accordingly, the Intervenor was granted leave to file an amended Contention 1 to supply the specificity it then lacked.[41] The subsequently issued Order also stated:

> "The Intervenor is also granted leave to file on or before April 20, 1981, appropriate amendments to Contention 1 in order to plead with specificity the respects in which the FES (due to be filed by the Staff by April 1) does not legally or factually comply with NEPA (Tr. 36, 38-9, 43). The Staff is granted leave to file a motion for summary disposition of Contention 1 as thus amended, on or before May 1, 1981 (Tr. 44-5, 47, 50). The Intervenor shall file its response to the Staff's motion for summary disposition of Contention 1 as amended, by May 20, 1981 (Tr. 52)."[42]

The 17 proposed amendments to Contention 1 filed by the Intervenor pursuant to leave granted, did not include original Contention 10. It was only after the Staff and the Licensee objected to the lack of specificity in the proposed amendments that the Intervenor first attempted to inject the argument that Contention 10 was always a part of Contention 1. This attempted evasion of the final framing of contentions at and following the prehearing conference cannot be allowed. The Intervenor has been previously admonished that our procedural rules and orders must be complied with.[43] We decline to permit this further departure from our orders and directives, and hold that original Contention 10 is not a part of, nor is it to be read in conjunction with, Contention 1 as stated in our controlling prehearing conference order establishing the issues in this proceeding.

[39]Tr. 35.
[40]Tr. 36.
[41]Tr. 43.
[42]Memorandum and Order, entered April 2, 1981, p. 4.
[43]Memorandum and Order, entered April 7, 1981, p. 2 (..."Because of the urgencies of time...we will treat the Intervenor's motion on the merits. However, in the future it is expected that procedural rules will be complied with.")

It is interesting to observe that of the 10 (subparagraphs a-j) alleged defects in the NEPA analysis which the original Contention 10 (which we have rejected) purported to assert, only three are included in the Intervenor's Statement of Genuine Issue of Fact, which accompanied his Answer Opposing the Motion for Summary Judgment (sic), dated May 19, 1981. This statement on genuine issues of material fact reads as follows:

> "1. Whether the Final Environmental Statement adequately addresses the alternatives of derating, conservation and solar power."[44]

This statement of genuine issues only addresses subparagraphs d, g and h of original Contention 10, so apparently the remainder are abandoned.

The thrust (and some of the flavor) of the Intervenor's attempts to inject original Contention 10 into the issues framed for hearing, may be discerned from portions of his May 12, 1981 filing, denominated Response to NRC Staff Objections to Proposed Amended Contention 1 and Licensee's Motion to Dismiss Contention 1. It was there stated, in regard to the pleading of Contention 1, that "The Intervenor is not required to voluntarily disclose its entire case to the Staff and Licensee, but through proper Rules of Procedure the process will disclose to the Staff and Licensee the theory of the Intervenor's case concerning Contention 1" (p. 4). It was further stated that the "evidence will show that conservation and solar energy would allow the derating and decommissioning of the Turkey Point Plant" (p. 5).

It is clear that the Intervenor's efforts to assert contentions regarding conservation and solar energy are irrelevant and beyond the scope of issues that may be considered in this license amendment proceeding. We have already discussed (pp. 2 and 10, *supra*) the controlling principle that an amendment proceeding is limited to a consideration of those issues "directly arising from the proposed change."[45] An amendment proceeding cannot be converted into a vehicle for the reconsideration of previously analyzed environmental impacts from the construction and operation of a new nuclear plant.

The environmental analysis of an amendment is focused only upon the changes arising from the amendment.[46] The consideration of alternatives in an amendment proceeding does not include the evaluation of alternatives to

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[44]Paragraph 2 of this statement of genuine issues of material fact, relating to alleged radioactive releases to unrestricted areas from storage of waste produced during repairs combined with hurricanes, is discussed in Section IV, dealing with Contention 4B, *post*.

[45]Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-245, 8 AEC 873, 875 (1974).

[46]Consumers Power Co. (Big Rock Point Nuclear Plant), ALAB-636, 13 NRC 322, (March 31, 1981).

the continued operation of the plant, even though the amendment might be necessary to enable continued reactor operation.[47] Energy conservation and solar energy are alternatives to the operation of Turkey Point, rather than alternatives to the proposed steam generator repairs. Hence they are beyond the scope of this proceeding, as they were the subject of prior NRC consideration in operating license proceedings.[48]

It has been held that the need for power is not a cognizable issue in a license amendment proceeding, where it had been explored at the prior construction permit and operating license proceedings.[49] Since an evaluation of the need for power accounts for electric energy saved through conservation or the use of solar power, a consideration of such alternatives in this proceeding would amount to an irrelevant reconsideration of the need for power from Turkey Point.[50] Such issues are beyond the scope of this proceeding.[51]

Finally, it should be recalled that the Intervenor submitted his untimely petition to intervene more than a year after the expiration of the intervention period (*supra*, p. 2). In support of showing his ability to make a contribution to this proceeding under the five-factor test for nontimely filings under 10 CFR §2.714(a)(1), the Intervenor asserted that he had experts who would testify as follows:

> "The three major areas to be addressed by these witnesses were identified as '(1) the long term on site storage of steam generator lower assemblies in an earthern floor facility; (2) the occupational radiation exposure, and (3) the release of liquid effluents containing radioactivity into a closed cycle cooling canal.' " (Supplemental Submission of Petitioner Mark P. Oncavage, June 5, 1979, p. 2). None of these "three major areas" which formed the basis of the intervention remains in issue, and no expert opinions or testimony have been profferred on these issues. This probably is due in part to the Licensee's responses to the concerns voiced by the Intervenor. For example, the originally proposed Steam Generation Storage Compound (SGSC) was to be an earthen floored structure with one end closed by concrete stop logs. The SGSC was to be located in the lay-down area at an elevation of

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[47]Northern States Power Co. (Prairie Island Nuclear Generating Plants, Units 1 and 2), ALAB-455, 7 NRC 41, 46-47, fn. 4 (1979); Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 266, fn. 6 (1979).

[48]Final Environmental Statement, July 1972, §X.

[49]Portland General Electric Co. (Trojan Nuclear Plant), ALAB-534, 9 NRC 287, 289 (1979).

[50]Consumers Power Co. (Midland Plant, Units 1 and 2), CLI-74-5, 7 AEC 19, 24 (1974); Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 46 fn. 4 (1978).

[51]42 *Fed. Reg.* 62569.

about five feet MLW. Licensee now plans to make the SGSC a concrete floored building which will meet local hurricane-resistance design standards, and it will be founded on an engineered fill of crushed, compacted limestone at an elevation of 17.5 feet MLW (FPL Affidavit). Again, FPL originally planned to replace the steam generator assemblies using a pipe-cut method, similar to the method being used in the Surry SGS repair. Surry's experience caused FPL to increase its estimate of occupational exposure from 1300 person-rem per unit to 2985 person-rem per unit. Primarily because of the high occupational dose associated with the pipe-cut method, FPL determined that an alternative, the channel-cut method, should be used. The channel-cut method results in an estimated occupational exposure of 2084 person-rem per unit (FES 4.1.1.3 and 4.1.1.4).

IV. CONTENTION 4B

Contention 4B states:

> There are likely to occur radioactive releases, (from the steam generator repair) to unrestricted areas which violate 10 CFR Part 20 or are not as low as reasonably achievable within the meaning of 10 CFR Part 50 as a result of a hurricane or tornado striking the site during repairs.

The parties were also put on notice by the Board's February 23, 1981 "Order Accepting Negotiated Schedule" that the Board intended to hear evidence on the relationship between the repair schedule and the hurricane season. These matters were addressed in the "Affidavit of Robert F. Abbey, Jr. on Contention 4B" filed by Staff (Staff Affidavit) and the "Affidavit of F. G. Flugger and H. H. Jabali and P. K. Wan on Contention 4B" filed by Licensee (FPL Affidavit).

The proposed steam generator repair for Unit 4 is scheduled to begin in late October, 1981, and end about June, 1982. The repair for Unit 3 is scheduled to begin in late October, 1982, and end about June, 1983 (FPL Affidavit at 10, Staff Affidavit at 5). The Atlantic hurricane season begins June 1 and extends through November 30 (FPL Affidavit at 4, Staff Affidavit at 2). Based on observations from 1886 through 1977, the median beginning date of the hurricane season is June 26, and the median ending date is October 29 (FPL Affidavit at 4). Observations from 1871 through 1978 in a 50-mile segment of coastline encompassing Turkey Point show that the earliest recorded hurricane made landfall on September 8 and the latest occurred on October 21 (Staff Affidavit at 1-2). Thus, although the proposed repair schedule is not based on the timing of the hurricane season, it does not substantially coincide with the historical hurricane season in southeastern Florida (FPL Affidavit at 10, Staff Statement of Material Facts at 2).

The tornado season in Florida is less well defined. Within 125 nautical miles of Turkey Point, 253 tornadoes were reported in the period 1950 through 1980 (Staff Affidavit at 2). These storms occurred throughout the year, but the peak month for tornadoes was June (*Ibid*). The high frequency of severe tornadoes characteristic of the midwest is not expected in Florida because meterological conditions in peninsular Florida differ from those in the midwest (FPL Affidavit at 6-8). While midwestern tornadoes often have windspeeds up to 300 mph or even more, tornadoes in southern Florida rarely have windspeeds above 200 mph (FPL Affidavits at 6-9, Table 1). The greatest inferred windspeed for a tornado within 125 nautical miles of Turkey Point is between 207 and 260 mph, an Intensity Class 4 storm on the Fujita Scale (Staff Affidavit at 5; FPL Affidavit, Table 1); the Licensee attests to evidence showing that this particular storm probably had windspeeds that were low in the Class 4 range (FPL Affidavit at 8-9).

The probability of occurrence of hazardous windspeeds at Turkey Point is very small. Staff estimated the probability of the site experiencing hurricane winds of 150 mph to be about 5×10^{-4}/yr and the probability of tornado windspeeds of 260 mph to be about 1.5×10^{-7}/yr (Staff Affidavit at 5). Licensee estimated the probability of a tornado with 200 mph winds occurring at the site to be 1.6×10^{-6}/yr (FPL Affidavit at 9). We conclude that the probability of these events occurring during the repair is somewhat less than the estimates above, because the repair activities will take place during a period less than a year in length.

The matter of hurricanes and tornadoes at Turkey Point is addressed in the Affidavit of Leonard G. Pardue on Contention 4B (Pardue Affidavit) attached to the Intervenor's Answer Opposing the Motion for Summary Judgment. The Pardue Affidavit predicts storm surges of 13-18 feet during a Category 4 hurricane (using the Saffir/Simpson Hurricane Scale) and a surge of more than 18 feet during a Category 5 hurricane. Whether these values are in terms of mean low water (MLW) or mean sea level (MSL) is not revealed. The Pardue Affidavit predicts that a "major hurricane" could produce a storm surge 15 feet above *MSL*, however. This compares with the estimate by FPL and Staff that a PMH would produce a storm surge of 18.3 feet above *MLW*. With regard to the chance that a hurricane will occur at Turkey Point, the Pardue Affidavit estimates the probability of a "major hurricane" occurring in a 50-mile segment of Florida coast in which Turkey Point is located to be 5×10^{-2} per year. This value compares with Staff's estimate of 5×10^{-4} per year probability that a 150 mph hurricane wind will occur at the site. The large coastal segment and greater wind range (from 111 mph up) considered by the Intervenor may account for the greater probability value given in the Pardue Affidavit.[52] We need not reconcile these different estimates, however, to reach a result with regard to the motion for summary disposition of Contention 4B, for reasons which are explained below.

Licensee's schedule for the proposed steam generator repair was not based on the timing of the hurricane season or the probability of tornado occurrence (FPL Affidavit at 10). FPL attests that consideration of the occurrence of a hurricane or tornado does not alter the safety evaluation of the repair activity reached by FPL or the NRC Staff (*Ibid.*, FPL Affidavit at 11). The physical work associated with removal and replacement of the steam generator lower assemblies (SGLAs) will occur within the reactor building; the reactor building is designed to withstand a tornado and the probable maximum hurricane (PMH) (FPL Affidavit at 10). During the repair the spent fuel will be removed from the reactor building and placed in the spent fuel complex, a structure independent of the reactor building and also designed to withstand a tornado and the PMH. If a wind-borne missile should enter the open equipment hatch of the reactor building during a hurricane or tornado, the missile could not impact the nuclear fuel or cause any other accident not previously evaluated (FPL Affidavit at 11). Water-borne missiles could not enter the open equipment hatch during the tidal surge associated with a PMH because the bottom of the hatch opening is at an elevation of more than 28 feet MLW (Steam Generator Repair Report, Figure 3.2-4). The storm surge during a PMH would reach a stillwater level of 18.3 feet MLW, with waves on the engineered fill of the reactor building cresting to less than 22.5 feet MLW (Affidavit of Richard B. Codell on Contention 6(a), (b), (c), and (e), accompanying the NRC Staff Second Motion for Summary Disposition, dated March 23, 1981, at 2-3).

As the SGLAs are removed from the reactor building, steel support saddles will be affixed to them (FPL Affidavit at 11). The SGLAs will then be placed temporarily in a laydown area at an elevation of 17.5 feet MLW or moved into the Steam Generator Storage Compound (SGSC) (*Ibid*). Neither tornadic nor PMH winds would be sufficient to move an SGLA temporarily located on support saddles in an open area because they weigh 185 tons (FPL Affidavit at 11-12; Staff Affidavit at 5). Nor would a tornado-borne missile be able to penetrate the steel wall of an SGLA (FPL Affidavit at 12). If the SGLAs are in the SGSC when the site is struck by a

[52]Staff also provided a summary of wind hazard probabilities for Turkey Point which ranged from 1.0×10^{-1} per year for the threshold hurricane wind speed of 73 mph to 1.0×10^{-4} for a hurricane wind speed of 167 mph. An estimate of 1.0×10^{-2} obtained for speeds of 105/110 mph agrees well with the Pardue Affidavit estimate. *See* Staff Affidavit at 4, Table 1.

tornado or PMH, they will be adequately protected from storm winds and tidal surge (FPL Affidavit at 13-14; *See* Codell Affidavit cited above and Licensee's Answer Supporting NRC Staff Motion for Summary Disposition of Contention 4A with supporting affidavits).[53]

Notwithstanding the fact that no radioactive release is to be expected from the SGLAs as a result of a storm at Turkey Point during the repair activity, both Licensee and the NRC Staff analyzed the hazard associated with such release were it to occur. It was shown in the FES (NUREG-0743) that given the worst-case accident involving a 12-foot drop of the SGLA, the radioactive release would be within 10 CFR Part 20 limits at the site boundary (FPL Affidavit at 15). Under storm conditions wind and turbulence would increase the dilution and further reduce airborne concentrations (FPL Affidavit at 16; Staff Affidavit at 7). Thus, if an SGLA were breached during a storm the resulting hazard would be insignificant.

From the foregoing, we find the following material facts as to which there are no genuine issues to be heard:

1. The proposed repair schedule does not substantially coincide with the historical hurricane season in southeastern Florida, and the probability of a tornado occurring at the site during the repair activity is remote.
2. Physical work associated with removal and replacement of the steam generator lower assemblies will be conducted inside the reactor building, which is designed to withstand a tornado or hurricane.
3. A steam generator lower assembly outside the reactor building would be unmoveable by tornado or hurricane winds or wind-driven water.
4. A tornado-borne missile could not penetrate the steel wall of a steam generator lower assembly.
5. Steam generator lower assemblies will be adequately protected from tornadoes and hurricanes when stored in the steam generator storage compound.
6. If a radioactive release from the steam generator lower assemblies should occur during a storm, the radiological consequences will fall within the permissible radiation levels of 10 CFR Part 20, levels

[53]Contention 4A, which stated that the SGLAs would be damaged by storm tides or seawater while stored in the SGSC, was summarily dismissed by us in our Order dated May 7, 1981. We granted the motion for summary disposition of that contention because the SGSC will be founded on engineered fill with a finished grade of 17.5 feet MLW, and the storage compound will comply with the design requirements of the Code of Metropolitan Dade County, Florida, with respect to wind loadings. Additionally, the facts showed that the SGLA walls would not be penetrated by corrosion during the period of storage on site.

which are applicable to normal reactor operation, rather than accident conditions. Accordingly, Contention 4B is subject to summary disposition.

V. FURTHER PROCEEDINGS

A. Termination of Evidentiary Hearing

The Board has now granted summary disposition of all of the Intervenor's admitted contentions.[54] There are therefore no cognizable contentions that remain to be heard, and hence there is no necessity to hold an evidentiary hearing.

The authority for terminating the evidentiary hearing, originally scheduled to commence June 2, 1981,[55] is to be found in the Appeal Board's decision in *Virginia Electric and Power Company* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC 451 (1980). In that case, the "Licensing Board granted the Applicant's motion for summary disposition of all issues in its favor and, accordingly, authorized the issuance of the license amendment" (11 NRC at 452). The Appeal Board affirmed this action granting summary disposition in its entirety. After reviewing the record regarding alternatives to proposed spent fuel pool modifications, it held that the Licensing Board "...correctly declined to order a hearing to explore further the Intervenors' suggested alternatives" (11 NRC at 456). After reviewing the service water cooling system contention, the Appeal Board stated:

> "...at no juncture did [intervenors] point to anything which might cast doubt upon the Applicant's thesis that, even should the postulated accident conditions occur, the facility's cooling system would remain capable of maintaining the pool water temperature at a level which posed no threat to the public health and safety. *In these circumstances, there was nothing to be heard*" (11 NRC at 461). (Emphasis supplied)

The Appeal Board has described its *North Anna* decision as follows:

> "That the Section 2.749 summary disposition procedures provide in reality as well as in theory, an efficacious means of avoiding unnecessary and possibly time-consuming hearings on demonstrably

[54]Original Contention 14 (Memorandum and Order dated April 2, 1981); Contentions 2, 3, 5, 6, 7 and 8 (Memorandum and Order dated April 29, 1981); Contention 4A (Order dated May 7, 1981); and Contention 1, amended Contention 1, 17 proposed amendments to Contention 1, and Contention 4B are summarily dismissed by the instant Memorandum and Order.

[55]Notice of Prehearing Conferences (Supplements to Schedule), dated March 10, 1981, p. 2; 46 *Fed. Reg.* 17318.

insubstantial issues is amply reflected by our recent decision in *Virginia Electric and Power Company* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-584, 11 NRC at 451. In that proceeding, involving an application for an operating license amendment to permit the expansion of the capacity of a spent fuel pool, the Licensing Board summarily resolved in the applicant's favor all of the intervenors' contentions.... More specifically, because, *in response to the applicant's motion for summary disposition*, the intervenors had not demonstrated that a genuine issue of fact existed respecting the environmental superiority of any of their suggested alternatives, we held that as a matter of law none of these alternatives had to be further explored at an evidentiary hearing."[56] (Emphasis in original.)

In the instant case, we have held that the alternatives of conservation and solar power, which allegedly "would allow the derating and decommissioning of the Turkey Point Plant,"[57] are beyond the scope of this proceeding as a matter of law. Accordingly, since all of the Intervenor's contentions have been summarily dismissed, there is nothing to be heard and no necessity for an evidentiary hearing.

The Intervenor argues that 10 CFR §51.51(b)(1) requires a public hearing at which the Staff will offer the FES into evidence.[58] Section 51.52(b)(1) provides in pertinent part as follows:

> "In a proceeding *in which a hearing is held* for the issuance of a permit, license, or order, or amendment to or renewal of a permit, license, or order, covered by §51.5(a), *and matters covered by this part are in issue*, the staff will offer the final environmental impact statement in evidence. Any party to the proceeding may take a position and offer evidence on the aspects of the proposed action covered by NEPA and this part in accordance with the provisions of Subpart G of Part 2 of this chapter." (Emphasis supplied)

As the italicized portions of this section show, the FES is to be offered into evidence only if a hearing is held. It does not itself require the holding of a hearing if one is not otherwise required. This section further provides that it applies if NEPA "matters covered by this part are in issue." Inasmuch as all contentions have been summarily dismissed, there is no necessity for a hearing, and there are no NEPA matters in issue.

[56]Houston Lighting and Power Company (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542, 550-51 (1980).
[57]Intervenor's Statement of Genuine Issue of Fact, dated May 19, 1981.
[58]Intervenor's Response to NRC Staff Objections to Proposed Amended Contention 1 and Licensee's Motion to Dismiss Contention 1, dated May 12, 1981, at pp. 5-6.

Consequently, the provision concerning offering the FES into evidence is not applicable.

B. Retention of Jurisdiction Concerning Radioactive Solid Wastes

There remains one matter for which the record is not sufficiently developed to enable the Board to rule with finality. This subject concerns the alleged storage on site of low level solid waste in "loosely stacked, sealed drums in roped off areas" (Affidavit of Douglas King, dated May 13, 1981, par. 4, 7). It is asserted that the amount of radioactive solid waste to be generated from the proposed repairs ranges from 1100 to 2300 cubic meters per unit, according to the FES (*Id.*, at par. 8).[59] It is further asserted in this affidavit that the availability of the Barnwell disposal site is limited, and that the outdoor storage of solid waste in drums is unreasonable in view of the likelihood of hurricanes or tornadoes (*Id.*, at par. 9-10).

The lack of an adequate record on this subject is probably attributable to the short time available to develop Contention 4B and the underlying data. At the prehearing conference on March 24, 1981, the Board permitted the Intervenor to amend Contention 4 by adding paragraph B, which raised the question of radioactive releases during the period of repairs (Tr. 56-60). This action was taken over objections of Staff and the Licensee that it injected new matters and issues when a trial was imminent (Tr. 61-72).

The Board, making a liberal construction of NRC discovery practice, also permitted the Intervenor to make a discovery site inspection and to perform some environmental sampling, subject to reasonable limitations.[60] It was contemplated that such inspection would be conducted expeditiously in view of the tight discovery and trial schedule, and that the parties would report promptly any significant discoveries. However, the Board received only a somewhat cryptic footnote from the Licensee on April 20, 1981, indicating that the site inspection had been conducted on April 19 and that some undescribed samples had been sent to an independent laboratory for analysis.[61] No other information regarding this site sampling has ever been received by the Board.

The only other information regarding observations made at the Intervenor's April 19 site inspection came on May 21, in the form of an affidavit by Douglas King executed on May 13, contained in Intervenor's Answer Opposing the Motion for Summary Judgment (sic) dated May 19,

[59]FPL estimates that this solid waste will contain 130 to 270 curies of radioactivity (FES, 4.1.2.2).

[60]Memorandum and Order, entered April 2, 1981, pp. 6-10.

[61]Licensee's Response to Intervenor's Motion to Continue or Deny Summary Disposition, dated April 20, 1981, at p. 3, fn. 9.

1981. That affidavit describes several hundred, loosely stacked drums apparently containing low level solid wastes. However, due to the posture of the filings made by the several parties and the time pressures of preparing for hearing, no information on this subject has been received from the Licensee or the Staff.

The Board wishes to keep the record open on the subject of solid wastes, their storage on site in drums, or their transportation or other disposition. Accordingly, all parties are requested and directed to furnish reasonably detailed and concrete information on these matters, by affidavits or other means tending to establish reliability. The parties are also requested to state their positions regarding what action, if any, the Board can or should take in this regard, including possible license amendment conditions.

Such written information should be lodged with the Board (not merely mailed) on or before 4 p.m., Monday, June 15, 1981.

ORDER

For all the foregoing reasons and based upon a consideration of the entire record in this matter, it is this 28th day of May, 1981

ORDERED

1. That the Staff's motions for summary disposition are granted as to all of the Intervenor's admitted contentions (Contentions 1, 2, 3, 4A, 4B, 5, 6, 7, 8, and originally numbered 14), and each of the said contentions or amendments thereto is dismissed with prejudice.

2. That the evidentiary hearing originally scheduled for June 2, 1981, is unnecessary, and it is hereby canceled.

3. That the parties are directed to file by 4 p.m., June 15, 1981, detailed information concerning the handling, storage, transportation or other disposition to be made of low level solid waste that may be produced at the Turkey Point facility as a result of the proposed steam generator repairs.

4. That the parties are further directed to state their positions as to whether the Board can or should take any action regarding solid waste resulting from steam generator repairs at Turkey Point, including the imposition of license amendment conditions.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

Dr. Emmeth A. Luebke
ADMINISTRATIVE JUDGE

Dr. Oscar H. Paris
ADMINISTRATIVE JUDGE

Marshall E. Miller, Chairman
ADMINISTRATIVE JUDGE

Cite as 13 NRC 708 (1981) **LBP-81-15**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Hugh K. Clark, Chairman
Dr. George A. Ferguson
Dr. Oscar H. Paris

In the Matter of **Docket Nos. 50-461 OL**
50-462 OL

ILLINOIS POWER COMPANY, *et al.*
(Clinton Power Station, Units 1 and 2) **May 29, 1981**

The Licensing Board issues an order admitting an intervenor and an interested state to this operating license proceeding. The Board also accepts certain of intervenor's contentions for litigation.

RULES OF PRACTICE: INTERVENTION PETITIONS (PLEADING REQUIREMENTS)

A contention need not plead evidentiary facts; it is enough if a contention alleges its basis with reasonable specificity. The merits of an issue are not to be considered at the pleading stage.

MEMORANDUM AND ORDER
(Admitting Petitioners, Accepting Contentions, and Ordering a Hearing)

I. SUMMARY

Illinois Power Company, *et al.* (IP or Applicants) filed an application with the Nuclear Regulatory Commission (Commission) for operating licenses to operate Units 1 and 2 of the Clinton Power Station.

Prairie Alliance, *et al.*, (Prairie Alliance) filed a petition to intervene and requested a hearing. Prairie Alliance has standing to intervene and has presented a number of allowable contentions. For the reasons set forth herein, Prairie Alliance is admitted as a party to the proceeding.

Bloomington-Normal Chapter of Prairie Alliance also filed a petition to intervene. Later it withdrew its petition, having decided to collaborate with its parent organization, Prairie Alliance.

The State of Illinois (Illinois) requested permission to participate in the hearing pursuant to 10 C.F.R. § 2.715(c). Its request is granted.

A hearing will be held on Applicants' request for the said operating licenses.

II. PROCEDURAL HISTORY

A. Application for Operating Licenses

On September 29, 1980, the Commission gave notice in the *Federal Register,* 45 *F.R.* 6437-9, of the receipt of an application by Applicants for licenses to operate Units 1 and 2 of Clinton Power Station. In part, this notice provided for the filing of petitions to intervene and requests for a hearing by October 29, 1980.

B. Prairie Alliance's Petition to Intervene and Request for Hearing - Ruling on Prairie Alliance's Standing to Intervene

A petition to intervene and a request for hearing, dated October 29, 1980, was filed by Prairie Alliance on behalf of itself and its members, including Stanley Elsasser, Rebecca Elsasser, Joanne Schwart, Jean Foy, Caroline Mueller, and Allen Samelson, The Petition described Prairie Alliance as a not-for-profit organization incorporated under the laws of Illinois and concerned with nuclear power and its relationship to the community and the environment. It has approximately 350 members, most of whom live, work, and own property within 35 miles of the Clinton Power Station.

The Petition further states that Stanley Elsasser and Rebecca Elsasser reside and own property at 817 East Main Street, Clinton, Illinois, approximately six miles west of the Clinton Power Station. It is also alleged that the health, ownership of property, work, and life-style of these persons will be affected by the licensing for operation of the Clinton Power Station.

Both the N.R.C. Staff (pp. 1-3 of brief of November 18, 1980) and Applicants (p. 2 of brief of November 10, 1980) concede that Prairie Alliance has standing to intervene. Prairie Alliance's Petition meets the criteria set forth by the Appeal Board in *Virginia Electric Power Company* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-522, 9 NRC 54 (1979). The Board rules that Prairie Alliance has standing to intervene.

C. The Petition of the Bloomington-Normal Chapter of Prairie Alliance to Intervene and Request for Hearing - Ruling on Petitioner's Standing to Intervene

By a letter dated October 29, 1980, the Bloomington-Normal Chapter of Prairie Alliance (Petitioner) requested a hearing and petitioned for the right to intervene. In a telephone conference between the Board, the Petitioner, Prairie Alliance, the Applicant and the Staff, on December 2, 1980, this petitioner expressed an intent to consolidate its efforts with its parent organization, Prairie Alliance. The Petitioner, as a separate entity, took no further part in this proceeding. The Board rules that this Petitioner has shown no standing to intervene. This Petitioner is dismissed as a separate party in this proceeding.

D. The Petition of Illinois to Participate in This Proceeding and Request for Hearing - Ruling of Board

By pleading dated October 29, 1980, Illinois requested a hearing and sought permission to participate therein as an interested State pursuant to 10 C.F.R. § 2.715(c). The request is granted. Its participation will be governed by the provisions of 10 C.F.R. § 2.715(c).

E. The First Special Prehearing Conference

On January 14, 1981, Prairie Alliance, which is not represented by counsel, filed a timely supplement to its petition to intervene. The supplement proposed 42 contentions. None of the proposed contentions met the specificity requirements of 10 C.F.R. § 2.714. At the First Special Prehearing Conference held on January 29, 1981 in Urbana, Illinois, a number of the proposed contentions were discussed. The futility of discussing the remainder of the proposed contentions became apparent.

In its answer of November 18, 1980 to Prairie Alliance's petition to intervene, the Staff stated that the petitioner had identified areas of interest sufficient to meet the aspect requirements of 10 C.F.R. § 2.714. In the hope of expediting this proceeding so that the substance of Prairie Alliance's contentions might be addressed, the Board approved a suggestion by Staff that it meet with Prairie Alliance in an attempt to explain what the Staff considered to be formal deficiencies in Prairie Alliance's proposed contentions. Counsel for the Applicants were invited to participate in such meeting but declined.

The time for filing revised contentions by Prairie Alliance was extended, to allow Prairie Alliance an opportunity to rewrite its contentions after meeting with the Staff.

E. The Second Special Prehearing Conference

Prairie Alliance filed its revised proposed contentions on March 30, 1981. Illinois' Brief of April 9, 1981 supported the revised contentions. Applicants' Brief of April 11, 1981 opposed such contentions. The Staff's Brief of April 11, 1981 objected to some of these contentions and did not object to others.

At the Second Special Prehearing Conference in Champaign, Illinois on April 14, 1981, Prairie Alliance orally presented its revised contentions one by one. Applicants, Staff, and Illinois commented on each revised contention. Applicants also made general comments concerning the contentions. These general comments will be discussed below.

III. THE CONTENTIONS

A. Applicants Object to a Number of the Revised Proposed Contentions as Having no Basis in Fact

Applicants argued that each contention must be specific and factually supported. It is now well settled that a contention need not plead evidentiary facts. It is enough if a contention alleges its basis with reasonable specificity. Also, the merits of an issue are not to be considered at the pleading stage. *See, Philadelphia Electric Company* (Peach Bottom Atomic Power Station, Units 2 and 3), ALAB-216, 8 AEC 13, 20 (1974); *Duke Power Company* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-128, 6 AEC 399, 401 (1973); *Mississippi Power and Light Company* (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 AEC 423 (1973); *Houston Lighting and Power Company* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-590, 11 NRC 542 (1980); *Commonwealth Edison Company* (Byron Nuclear Power Station, Units 1 and 2), LBP-80-30, 12 NRC 683 (1980); and other cases cited in the latter. Applicants' argument is rejected.

B. Issues Decided at the Construction Permit Stage

Applicants allege that many of the revised proposed contentions are concerned with issues decided at the construction permit stage. A detailed review of the contentions reveals that only two contentions, 6 and 17, are in this category. Contention 6 is allowed because of information not available at the Construction Permit stage. Contention 17 is denied. Applicants' allegation does not deserve further discussion.

C. Issues not Pertinent to the Clinton Power Station

Applicants allege that several of the revised proposed contentions raise issues which could only pertain to facilities with designs radically different from the design of the Clinton Power Station. This allegation goes to the problem of proof. It is not pertinent at the pleading stage. Moreover, this general allegation is not raised with respect to any specific contention.

D. Premature Issues

Applicants object to a number of the revised proposed contentions on the ground that they are premature. It is true that a number of the contentions relate to requirements made since the FSAR was filed. Some of these contentions have been allowed. After the Staff's SER is filed, but not later than the prehearing conference, these contentions will be reconsidered. The Board has been lenient as to these contentions since it is impossible for the intervenor to be more specific at this stage of the proceeding.

E. Untimeliness of Revised Proposal Contentions

Applicants argue that the revised proposed contentions were filed after the due date and hence are untimely. The Staff did not concur (Tr. 88). Under 10 C.F.R. § 2.714(a)(1), untimely contentions will not be granted without a determination that the request should be granted based upon a balancing of five factors set forth in the regulation. Prairie Alliance was granted by the Board an opportunity to revise its contentions. It would be unconscionable to now hold that the revised contentions were late. Moreover, the earlier contentions were so broad that it would be difficult to broaden them further. Without doubt the revised contentions are more specific than the earlier ones. Applicants have not pointed out any specific revised contention which is not within the scope of the earlier contentions. The Applicants' argument is rejected.

F. Rulings on the Specific Revised Proposed Contentions

Of necessity, the proposed contentions have been drafted using information now available, including Applicants' Final Safety Analysis Report

(FSAR) and Operating License Environmental Report (OL-ER). Since these documents were filed, there have been extensive changes in the requirements and regulations concerning operating licenses. These changes will be reflected in the Staff's Safety Evaluation Report (SER) now scheduled for issuance in January 1982 and in the Staff's Final Environmental Statement (FES) now scheduled for issuance in March 1982 (Tr. 111). After discovery and the availability of the SER, the supplement thereto (SSER), and the FES, the allowed contentions will be subject to reconsideration as to both scope and allowability.

Clarifying changes have been made by the Board in some allowed contentions. The allowed contentions have been given new numbers. As revised and renumbered, these contentions are set forth in Appendix A attached hereto.

Contention 1. This contention questions the adequacy of Applicants' emergency planning, the requisites of which were extensively amended and upgraded by the Commission effective November 3, 1980, 45 *F.R.* 55402 *et seq.*, August 19, 1980. Contention 1 *is admitted*, except for Part (g) which *is denied* as too vague to meet the requirements of 10 C.F.R. § 2.714. Part (g) *is denied* for the additional reason that it is outside the scope of emergency planning.

Contention 2. This contention is concerned with the management and technical qualifications of Applicants. These are appropriate matters for consideration. Contention 2, as revised by the Board, *is allowed.*

Contention 3. This contention is concerned with the Applicants' financial qualifications. This is an appropriate area for consideration. This contention, as revised by the Board, *is allowed.*

Contention 4. This contention challenges the Applicants' security planning. This matter deserves consideration. The contention *is allowed.*

Contention 5. This contention is concerned with beyond design basis, or "Class 9", accidents. The Commission's Policy Statement of June 13, 1980, 45 *F.R.* 40101, requires the N.R.C. Staff, not the Applicants, to consider the environmental consequences of accidents beyond design basis in the Environmental Statement. Moreover, this contention lacks the specificity required by 10 C.F.R. § 2.714. The contention *is denied* without prejudice to the proffer of a specific contention after Prairie Alliance has had a chance to study the Staff's FES and SER.

Contention 6. Anticipated transients without scram (ATWS) is the subject of this contention. It is alleged that faulty welds on a number of control rod drive tubes raise the likelihood that an ATWS can occur. Contention 6, as revised and renumbered by the Board, *is allowed.*

Contention 7. Questions as to possible deficiencies in control room design, in light of current requirements by the N.R.C., are raised by this contention. As modified and renumbered by the Board, Contention 7 *is admitted.*

Contention 8. This contention states that full consideration has not been given to systems interactions, specifically, multiple equipment failures, minor failures, and failures of non-safety related systems that interact with safety systems. To the extent that the contention intends to address accidents beyond design basis, it duplicates Contention 5 and on that ground *is inadmissible.* Additionally, it *is not allowed* because bases for the contention have not been set forth with the specificity required by 10 C.F.R. § 2.714.

Contention 9. This contention, which is concerned with social, economic, and psychological effects of plant operation, *is denied* as not having bases set forth with the specificity required by 10 C.F.R. § 2.714. Parts (f) and (g) which relate to psychological effects, *are further denied* as being inappropriate for litigation. On December 5, 1980, the Commission announced that, pending a reconsideration at a later time, "requests to admit contentions based on psychological stress are effectively denied.", CLI-80-39, 12 NRC 607 (1980).

Contention 10. This contention questions whether the Clinton Power Station's units meet the General Design Criteria requirements of 10 C.F.R. Part 50, Appendix A. Part (a)(2) alleges new information and Part (c) alleges changed conditions. These two parts *are admitted.* The remaining parts *are denied* as too vague to meet the requirements of 10 C.F.R. § 2.714.

Contention 11. The radiation monitoring system is challenged in this contention. The contention *is denied* because a basis for it has not been set forth with the specificity required by 10 C.F.R. § 2.714.

Contention 12. This contention addresses reactor coolant pressure boundary leaks. As orally modified by Prairie Alliance at the prehearing conference (Tr. 166), it *is admitted.*

Contention 13. Concern is expressed by this contention that radiation exposure levels may be inadequately maintained. Parts (c) and (d) *are admitted* as revised and renumbered by the Board. The remaining parts *are denied* as too vague to meet the requirements of 10 C.F.R. § 2.714.

Contention 14. Questions are raised by this contention concerning the emergency core cooling system. Parts (a) and (c) *are admitted* as renumbered. Part (b) *is denied* as too vague to meet the specificity requirements of 10 C.F.R. § 2.714.

Contention 15. This contention concerns the effects of low-level radiation released in several different ways. Part (a) refers to releases caused by accident sequences. It *is inadmissible* because Commission policy does not require treatment in the ER of beyond design basis accidents. Part (b) relates to occupational doses and *is inadmissible* because the occupational doses of all workers on the site will be governed by the acceptable levels specified in 10 C.F.R. Part 20. Parts (c) and (d), which relate to atmospheric effluents, *are admitted* for litigation. Part (e), which is concerned with the residual risks of low-level radiation released from Units 1 and 2 of the Clinton Station, *is also admissible.** *See, Public Service Company of Oklahoma* (Black Fox Station, Units 1 and 2), CLI-80-31, 12 NRC 264 (1980).

Contention 16. This contention questions the health and safety effects of radioactivity released during the transportation of radioactive fuel and waste to and from the Clinton Station and during the fuel cycle required for Units 1 and 2. Parts (a), (b), and (c) fail to meet the specificity requirements of 10 C.F.R. § 2.714 and *are inadmissible* on that ground. To the extent that they relate to off-site transportation of fuel, they *are inadmissible* as being irrelevant to this proceeding. *See, Pennsylvania Power & Light Company, et al.* (Susquehanna Steam Electric Station, Units 1 and 2), LBP-79-6, 9 NRC 291, 315 (1971) and cases cited therein. Part (d) relates to the transfer system for spent fuel on site; it is both relevant and specific and *is admitted* as renumbered.

Contention 17. The cost-benefit analysis for Units 1 and 2 is alleged to be grossly inaccurate in this contention. Parts (a), (b), and (c) relate to the construction permit stage and *are therefore denied. See, Niagra Mohawk Power Corporation* (Nine Mile Point Nuclear Station, Unit 2) ALAB-264, 1 NRC 347, 357 (1957); *Carolina Power and Light Company* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3, and 4), CLI-79-5, 9 NRC 607 (1979); and *Commonwealth Edison Company* (Byron Nuclear Power Station, Units 1 and 2), LBP-80-30, 12 NRC 683, 691 (1980). Parts (d), (e), and (f) *are denied* as too vague to meet the specificity requirements of 10 C.F.R. § 2.714.

*Staff interpreted Part (e) as being a challenge to Table S-3 of 10 C.F.R. Part 51 and inadmissible on that ground. The contention, however, relates to releases from the reactors themselves, not the fuel cycle activities required for the reactors.

Contention 18. This contention alleges that environmentally superior and less costly alternatives make operation of the Clinton Station unnecessary. This matter was fully explored at the construction permit stage; the contention *is denied* as improper for consideration at the operating license stage. *See, Illinois Power Company* (Clinton Power Station, Units 1 and 2), 2 NRC 579 (1975); 3 NRC 135 (1976); and 4 NRC 27 (1976).

Contention 19. This contention is a list of generic issues. The Staff's SER is scheduled to be filed in January 1982. The SER must, and will, address generic issues in detail, including the nexus of such issues to the Clinton Station. If, after receipt of the SER, Prairie Alliance wishes to raise one or more generic issues, revised contentions having the required specificity can be filed at that time. As the contention now stands, it *is denied* as being too vague to meet the requirements of 10 C.F.R. § 2.714.

ORDER

For the foregoing reasons and based upon a consideration of the entire record in this matter, it is this 29th day of May, 1981

ORDERED

1. That Prairie Alliance is admitted as a party to this proceeding;

2. That the revised contentions of Prairie Alliance set forth in Appendix A hereof are accepted for litigation, and all other contentions are denied;

3. That Bloomington-Normal Chapter of Prairie Alliance is denied admission as a separate party to this proceeding;

4. That the State of Illinois is permitted to participate in this proceeding as an interested State pursuant to 10 C.F.R. § 2.715(c);

5. That a hearing shall be held on Applicants' request for licenses to operate Units 1 and 2 of the Clinton Station;

6. That discovery on the accepted contentions shall begin forthwith and proceed on the following schedule: First round discovery requests must be filed not later than June 26, 1981; Responses to such requests must be made not later than July 27, 1981, or 4 weeks after receipt of discovery requests, whichever is the earlier date; Guidance as to the timing of further discovery will be given by the Board as needed;

7. That, pursuant to the provisions of 10 C.F.R. § 2.740(e)(3), responses to requests for discovery shall be supplemented as information becomes available to render the responses current and accurate.

Judge George A. Ferguson concurs in, and participated in the drafting of, this Memorandum and Order. He was prevented from signing it because of attendance at another proceeding.

THE ATOMIC SAFETY AND
LICENSING BOARD

Hugh K. Clark, Chairman
ADMINISTRATIVE JUDGE

Oscar H. Paris, Member
ADMINISTRATIVE JUDGE

APPENDIX A

There follows the list of contentions which are accepted for litigation in this proceeding.

1. Clinton Power Station (CPS) should not be licensed to operate until a safe and feasible emergency plan has been developed which complies fully with current NRC requirements. *See* 10 CFR Part 50, Appendix E, NUREGs-0696 and -0654. The emergency plan currently proposed by Illinois Power Company (IP) as delineated in the Final Safety Analysis Report (FSAR), is insufficient in the following respects:

(a) IP has failed to adequately incorporate emergency planning for a plume exposure pathway emergency planning zone (plume EPZ) of a minimum ten-mile radius from the CPS and an ingestion exposure pathway emergency planning zone (ingestion EPZ) of a minimum fifty mile radius from the CPS, as required by 10 CFR Part 50, Appendix E. This planning should include, at a minimum, consideration of the following items peculiar to the CPS site vicinity and region:

(1) Problems posed in effecting termination of activities at outdoor recreational facilities within the plume EPZ and ingestion EPZ;

(2) Difficulties posed by "special facilities" which, because of the nature of the populace, the number of people involved or the means of available communication and transportation, give rise to especially acute problems in emergency response actions. Included in this category are universities and other schools, nursing homes, mental health facilities, prisons and jails, children's camps, state parks, industrial parks, and other such facilities located within the plume EPZ and ingestion EPZ;

(3) The severe, but not uncommon, weather conditions, such as heavy snowfalls, sleet storms, and tornadoes which occur in the site vicinity and plume and ingestion EPZs throughout the year.

(b) IPC has not demonstrated concrete coordination plans with the appropriate state and local agencies involved in emergency planning and response actions. Thus far IP has failed to effect meaningful agreements with "17 named agencies as well as others such as local hospitals and physicians" as required by the NRC Staff in the Construction Permit Safety Evaluation Report, Section 13.4. *See* FSAR Emergency Plan, Sections 5.5.3, 5.5.4, B6, B7, and B9.

(c) The emergency plan lacks sufficient detail in the area of emergency preparedness training. For example, the plan does not state who will provide the training of local services personnel or how often that training will be provided. The same is true of training plans for accident assessment personnel and the "Emergency Response Organization". Additionally, there is no provision for emergency training of security personnel or a radiological orientation training program for local services personnel, including local news media persons, as required by 10 CFR Part 50, Appendix E.

(d) As required by 10 CFR Part 50, Appendix E, the emergency plan fails to identify or describe the following items:

(1) The special qualifications of non-IP employees who will be utilized in emergency training operations or recovery;

(2) The criteria for determining the need for notification and participation of local, state and federal agencies;

(3) An analysis of the time required to evacuate or provide other protective measures for various sectors and distances within the plume exposure and ingestion EPZs for both transient and permanent publics;

(4) A sufficient identification of the persons who will be responsible for making off-site dose projections;

(5) An adequate description of how off-site dose projections will be made and how the results will be transmitted to appropriate government entities;

(6) Plans for yearly dissemination to the public within the plume exposure and ingestion EPZs of basic emergency planning information, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency;

(7) An identification of the appropriate state and local government officials within the EPZ which will require notification under accident conditions.

(8) A demonstration that state and local officials have the capability to make a public notification decision promptly upon being informed of an emergency condition.

(e) The requisite protective actions necessary to assure isolation of people from the plume and ingestion EPZs in case of an off-site or general emergency or other serious accident is not described with sufficient detail in the Emergency Plan. *See* FSAR Emergency Plan, Section 5.4.3.1.

(f) IP has failed to provide adequate emergency support facilities for the CPS. The FSAR lacks documentation concerning compliance with the current regulatory requirements for the Technical Support Center, the Operational Support Center, the Emergency Operations Facility, the Safety Parameter Display System, and the Nuclear Data Link. *See* NUREG 0696.

2. The CPS should not be licensed to operate until IP has demonstrated, as required by 10 CFR 50.34(b) and Part 50, Appendix B, that it possesses sufficient management and technical qualifications to assure that the CPS

will be (a) maintained in a safe condition while operating normally, or (b) safely operated and controlled in the event of an abnormal occurrence or emergency, or (c) permanently shut down and maintained in a safe condition.

Repeated Quality Assurance (QA) and Quality Control (QC) problems are noted in NRC Region III Inspection Reports. Specifically, IP's QA and QC program is consistently deficient in its ability to assure (1) a sufficient number of experienced personnel, (2) integrity of welding procedures, and (3) numerous other QA and QC functions. These incidents, among others, raise serious questions as to IP's management and technical capabilities to operate, backfit, and permanently shut down the CPS in compliance with regulatory requirements.

3. In noncompliance with 10 CFR 50.33(f) and Part 50, Appendix C, IP has not demonstrated that it possesses or has reasonable assurance of obtaining the funds necessary to pay the estimated costs of operation, plus the estimated cost of permanently shutting the facility down and maintaining it in a safe condition.

Since Construction Permit issuance, IP has placed an increasing reliance on external financing of construction of the CPS, mainly in the form of bonds carrying high interest rates and common stock for which relatively high dividends must be paid. These facts call into serious question IP's capability to maintain the operation and permanent shut-down of the CPS in a way that provides assurance of public health and safety.

4. The CPS should not be licensed to operate until IP has developed and demonstrated an adequate security plan which complies with 10 CFR 73.55. The FSAR does not give adequate assurance that all regulatory requirements have been or will be met prior to operation. *See* FSAR, p. 1.8-25, Regulatory Guide 1.17, Revision 1.

5. The CPS is especially vulnerable to anticipated transients without scram (ATWS) due to the faulty welds during construction which have caused "burn through/suck back" on a number of control rod drive tubes. These defects have not been adequately analyzed or repaired. The CPS should not be licensed to operate until IP has completed an ATWS analysis for (1) redundancy, (2) systems interaction, (3) loss of coolant accident, and (4) incidents such as those experienced in other GE boiling water reactors.

6 (PA, #7). The design and fabrication of the CPS control room layout and instrumentation have not been modified to meet current regulatory requirements in NUREGs-0660, -0694, -0737. Specifically:

(a) The CPS lacks sufficient instrumentation for displaying and recording the reactor pressure vessel water level.

(b) The CPS lacks sufficient instrumentation for detecting inadequate core cooling in case of an abnormal occurrence.

(c) Direct indication of safety relief valve position should be, but is not, provided for in the CPS instrumentation.

(d) A Safety Parameter Display System should be, but is not, provided for in the main control room.

(e) The CPS lacks adequate instrumentation for monitoring accident conditions.

(f) IP has not demonstrated its ability to comply with current NRC requirements for overall control room design standards.

(g) The CPS control room design and instrumentation has not been subjected to a comparative evaluation of the interaction of human factors and efficiency of operation.

(h) Not all CPS control panels are completely unobstructed and accessible. It is insufficient to have certain surveillance and monitoring actions on back row panels. Moreover, there has been no documentation of the criteria used to determine which instruments should be placed on back row panels.

(i) The FSAR contains no evaluation of the CPS control room layout and instrumentation in terms of the new criteria resulting from the accident at TMI Unit 2.

(j) The FSAR contains no documentation of how the power station can or will be modified to meet the new criteria imposed following the TMI accident.

7 (PA, # 10). The CPS nuclear system has not been demonstrated to meet the General Design Criteria requirements of 10 CFR Part 50, Appendix A. Specifically,

(a) In noncompliance with Criterion 2, the seismic qualification of the CPS design does not account for the worst case seismic activity now known to occur in the site region;

(b) In noncompliance with Criterion 4, the CPS containment is not, but should be, hardened to account for the impact of existing, and possibly increased, civilian aircraft traffic in the site vicinity; only one of four federal vector pathways near the site has been considered by IP in calculating the probability of aircraft impact of the CPS containment.

8 (PA, # 12). The CPS should not be licensed to operate until Applicants have demonstrated the capability to comply with NRC regulatory require-

ments (10 CFR Part 50, Appendix A) regarding detection of reactor coolant pressure boundary leaks. Specifically,

(a) In noncompliance with Criterion 13, sump flow monitoring calculations and indication devices are not, but should be, seismically qualified;

(b) In noncompliance with Criterion 14, the transmitters of sump flow monitoring instruments for drywell equipment and floor drains are not, but should be, readily accessible for operability and calibration during plant operation.

9 (PA, # 13). The CPS should not be licensed to operate until Applicants have demonstrated that radiation exposure levels will be maintained as-low-as-reasonably-achievable as required in 10 CFR 20.1. The FSAR does not adequately consider occupational radiation exposure to be expected from either the normal operation of CPS Units 1 and 2 or that which may occur during an abnormal occurrence or serious accident. Specifically,

(a) Applicants have failed to provide a sufficient number of monitors to continuously measure airborne radioactivity; additionally, the monitors provided are not sufficiently sensitive in that they require up to 10 hours to detect emissions;

(b) The area radiation monitoring equipment does not provide a reasonable assurance of accuracy in that it is only accurate within plus or minus 20%.

10 (PA, # 14). The CPS Emergency Core Cooling System (ECCS) has not been demonstrated to meet the requirements of 10 CFR Part 60.46 and 10 CFR Part 50, Appendix K. Specifically,

(a) In noncompliance with 10 CFR Part 50.46, the core spray distributing of CPS's ECCS is of unproven operating capability;

(b) In noncompliance with 10 CFR Part 50, Appendix K, the models used to predict ECCS performance of the CPS have not been proven accurate.

11 (PA, # 15). The effects of the low-level radiation to be released from Clinton Units 1 and 2 have not been adequately assessed and considered in the following respects:

(a) gaseous effluents anticipated to be released from Clinton Unit 2 are not, but should be, considered in calculations estimating population doses;

(b) the methods used to calculate atmospheric effluents of routine releases are inadequate in that conservative estimates were not, but should have been, used by IP;

(c) the residual risks of low-level radiation which will result from the release of radionuclides from Clinton Units 1 and 2 have not been, but should be, adequately assessed and factored into the NEPA cost-benefit analysis for Clinton Units 1 and 2.

12 (PA, #16). Applicants have failed to provide a procedure for preoperational testing of the functional capability of the spent fuel transfer system which provides a reasonable assurance of safety. The spent fuel transfer tube is of unproven design for the CPS design. In the absence of additional testing, the safe operation of the spent fuel transfer system is questionable. Additionally, there is no assurance that occupational exposure to personnel will be maintained as-low-as-reasonably-achievable for the operation and maintenance of the spent fuel transfer system.

Cite as 13 NRC 725 (1981) **DD-81-4**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT
Victor Stello, Jr., Director

In the Matter of **Docket Nos. 50-247**
50-286
(10 C.F.R. 2.206)

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
(Indian Point, Unit No. 2)

POWER AUTHORITY OF THE STATE OF NEW YORK
(Indian Point, Unit No. 3) **May 14, 1981**

The Director of the Office of Inspection and Enforcement denies a request under 10 C.F.R. 2.206 which asked the Commission to order the immediate suspension of operations at Indian Point Units 2 and 3 because of alleged noncompliance by the two facilities with the NRC's emergency planning rule.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By letter dated April 1, 1981, Mr. Donald Ross and Ms. Joan Holt, on behalf of the New York Public Interest Research Group, Inc. ("NYPIRG") requested the Commission to "order the immediate suspension of operations at Indian Point on the grounds of noncompliance with the NRC's Final Emergency Planning Rule". In a letter dated April 6, 1981, Ms. Holt emphasized that NYPIRG was seeking "*immediate* relief" (emphasis in original) and expressed concern that NYPRIG's request was being treated as a request for action pursuant to 10 C.F.R. 2.206 of the Commission's regulations. The Commission has determined that NYPIRG's request is most appropriately treated as a request for action pursuant to 10 C.F.R 2.206 and, accordingly, has referred the matter to me for action. After considering the request, I have concluded for the reasons which follow that the alleged noncompliance by the Indian Point facilities with the NRC's Final Emergency Rule does not warrant in itself the immediate suspension of a resumption of operations at the two Indian Point facilities. Accordingly, I have determined not to grant the requested relief.

DISCUSSION

NYPIRG has requested that there be an immediate suspension of operations at Indian Point Units 2 and 3 until such time as there is a workable and implemented emergency plan for the two facilities and until such time as an investigation by the Atomic Safety and Licensing Board that was ordered by the Commission has been completed. The Commission has had occasion recently to consider this very issue and has resolved the issue in favor of allowing continued operation of the facilities.

On May 30, 1980, the Commission ordered a discretionary adjudication to resolve safety issues at the two Indian Point facilities raised by a previous petition submitted by the Union of Concerned Scientists. On July 15, 1980, the Commission decided that the risk posed by operation of the two facilities did not warrant suspension of the facilities' operating licenses during the pendency of the adjudicatory proceeding. On January 8, 1981, the Commission, after having addressed a number of concerns including the lack of an emergency plan for the surrounding area, concluded that its earlier decision to permit continued operation of Indian Point 3 remained valid. With respect to Indian Point 2, however, the Commission declared that, prior to resumption of operations at the facility, it would reexamine its July 15, 1980 decision to permit continued operations at the facility in order to determine whether the July 15 decision remained valid.

Until April 1, 1981, the Commission's regulations did not require an emergency plan to be in effect for the service area. However, if after April 1, 1981, the NRC finds that the state of emergency preparedness does not provide reasonable assurance that appropriate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down or whether other enforcement action is appropriate. With respect to Indian Point 2, on April 7, 1981, the Commission, in a meeting attended by Ms. Holt, addressed the question of whether its decision to permit continued operations of the facility remained valid. For the reasons set forth orally in that meeting, I recommended that the Commission not disturb its previous decision to allow continued operation of the plants. In that meeting, the Commission spent considerable time discussing the emergency plans for the Indian Point facilities. As reflected in the Commission Secretary's memorandum dated April 10, 1981 summarizing the April 7, 1981 meeting "[t]he Commission, by a vote of 3-1, agreed that its decision of July 11, 1980, to permit continued operation of Indian Point 2 and 3 remains valid."[1] On April 22, 1981, the Commission was further briefed on the status of offsite emergency preparedness around the nuclear power facilities in New York State. On April 24, 1981, letters were sent to all holders of power reactor operating licenses in New York State, including the Indian Point units, requesting that certain deficiencies identified to the NRC by the Federal Emergency Management Agency be corrected within 120 days.

The Commission has already determined, in accordance with my prior recommendation, that the two Indian Point facilities can operate despite alleged problems with the emergency plan for the facilities. Therefore, NYPIRG's request for immediate suspension of operations at Indian Point Units 2 and 3 based on alleged noncompliance with the NRC's Final Emergency Planning Rule is denied. I believe it is important to add, however, that as the Commission itself noted, this is not the final judgment on the safety of the two facilities. I will continue to monitor the operations of the two facilities, particularly with a view to the status of the emergency plans for the two facilities and will take appropriate action to protect the public's health and safety as circumstances warrant.

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555 and in the local public document room for the two facilities, located at the White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Additionally, a copy of this decision will be filed with the Secretary of the Commission for review by the Commission in accordance with 10 C.F.R. Section 2.206(c) of the Commission's regulations. As provided in 10 C.F.R. 2.206(c), this decision will constitute final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

Victor Stello, Jr., Director
Office of
Inspection and Enforcement

Dated at Bethesda, Maryland
this 14th day of May 1981.

[1] In a footnote to this statement, the Secretary's memorandum noted that the July 15, 1980 date, as discussed in the Commission's order of January 8, 1981, was in error and should have read July 11, 1980.

Cite as 13 NRC 728 (1981) DD-81-5

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION
Harold R. Denton, Director

In the Matter of **Docket Nos. STN 50-454**
STN 50-455
(10 C.F.R. 2.206)

COMMONWEALTH EDISON COMPANY
(Byron Station, Units 1 and 2) **May 7, 1981**

The Director of Nuclear Reactor Regulation denies a petition under 10 C.F.R. 2.206 that requested institution of proceedings to suspend or revoke the Byron construction permits unless a number of safety problems are resolved.

ATOMIC ENERGY ACT: SAFETY STANDARDS

Investment made by a utility in constructing a facility is not a proper factor for consideration in whether an operating license should be issued. The ultimate safety standards may not be compromised by consideration of the cost or difficulty associated with implementing measures required to ensure safety.

ATOMIC ENERGY ACT: RESOLUTION OF SAFETY ISSUES

Institution of proceedings prior to consideration of an operating license is not mandated whenever an unresolved safety question is raised after issuance of a construction permit. Continued construction despite unresolved safety questions does not itself pose any danger to public health and safety.

RULES OF PRACTICE: SHOW-CAUSE PROCEEDING

In the absence of extraordinary circumstances, the Director of NRR will not upset the usual two-stage licensing process by instituting a proceeding in response to a 10 C.F.R. 2.206 petition to consider issues that are properly within the scope of the operating license review.

RULES OF PRACTICE: PETITIONS UNDER 10 C.F.R. 2.206

Petitions under 10 C.F.R. 2.206 may not be used as a device to avoid a forum in which the issues raised in the petition more logically should be presented.

NEPA: SCOPE OF REVIEW

Under the Commission's June 1980 policy statement, future environmental statements must include consideration of severe accidents, including those that result in inadequate core-cooling and melting of the core.

OPERATING LICENSE: SCOPE OF REVIEW

The NRC staff should measure operating license applications against the regulations and the TMI-related requirements found in NUREG-0737. Other items in the Commission's Action Plan should be addressed through the normal process for development and adoption of new requirements rather than through immediate imposition as pending applications.

REGULATORY GUIDES: STATUS

Regulatory Guides are not regulations but merely one means of satisfying a regulatory requirement, and thus conformance to Regulatory Guides is not a prerequisite to issuance of a license.

OPERATING LICENSES: CRITERIA

Explanations of why an operating license should be issued in spite of unresolved generic safety issues should appear in the staff's Safety Evaluation Report.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By letter dated November 21, 1980, the Rockford League of Women Voters (the "League") transmitted a petition pursuant to 10 CFR §2.206(a) requesting that the Director of Nuclear Reactor Regulation initiate a proceeding pursuant to 10 CFR §2.202 to:

(a) Modify the construction permits issued to Commonwealth Edison Company for the Byron Nuclear Power Station, Units 1 and 2, so that construction may not proceed without resolution of all outstanding safety problems presently applicable to the Byron Station;

(b) Suspend or revoke the authority earlier granted to Commonwealth Edison Company to construct the Byron Station until such time as Commonwealth Edison has formulated an acceptable and realistic plan for resolving all outstanding safety problems;

(c) Revoke the construction permits issued to Commonwealth Edison Company for the Byron Station, if the outstanding safety problems cannot be resolved prior to the completion of construction due to Edison's financial condition or for other reasons; and

(d) Pending full hearings and determinations on these requests, immediately halt further construction of the Byron Station.

By my letter of December 22, 1980 to the League,[1] I acknowledged receipt of the League's request of November 21, 1980 and denied request (d) for an immediate halt to further construction of the Byron Station. Pursuant to 10 CFR 2.206, this decision is my response to the League's remaining requests (a), (b) and (c). In developing this response, I have also considered a supplemental, though unsigned, affidavit which the League submitted by letter dated January 27, 1981.

Discussion

The construction permits for the Byron facility were issued by the Office of Nuclear Reactor Regulation on December 31, 1975. Consequently, the Byron facility is currently being constructed pursuant to valid construction

[1]Notice of receipt of the petition was published in the *Federal Register* on December 31, 1980 (45 FR 86584). Counsel for Commonwealth Edison submitted comments on the petition in a letter of February 13, 1981.

permits. The League's petition raises principally questions concerning the adequacy of plant design.[2] In the construction permit review, the staff primarily reviews design criteria and the plant's preliminary design. Information regarding the detailed design of the plant is not required for the issuance of a construction permit. Detailed design features of the plant are generally developed after issuance of the construction permit and are evaluated and approved during the course of the staff's review of the operating license application. In the interim, a licensee pursues construction work under a construction permit at its own risk pending approval of the final design of the plant. Investment made by a utility in constructing a facility is not a proper factor when considering the issuance of an operating license. *PRDC* v. *International Union of Electrical, Radio and Machine Workers,* 367 U.S. 396, 415 (1961). Prior to receiving an operating license, the applicant will be required to do anything necessary to ensure safe operation of the plant. The cost or difficulty associated with implementing actions needed to ensure safety are not relevant considerations to this agency. *Public Service Company of New Hampshire, et al.* (Seabrook Station, Units 1 and 2), ALAB-623, 12 NRC 670, 677-78 (Dec. 9, 1980). The safety standards which an applicant must meet at the operating license stage are unconditional. Consequently, the League's concern that continued construction of the Byron facility might bias safety decisions at the operating license stage is not well founded given the NRC's responsibility and the unconditional safety standards that this agency applies. *See Porter County Chapter of the Izaak Walton League, Inc.* v. *NRC,* 606 F.2d 1363, 1369-70 (D. C. Cir. 1979).

Reactor licensing is a two-stage process, the first stage being the consideration of issuance of a construction permit and the second being the consideration of an operating license for a facility. This two-stage process has been sanctioned by the Supreme Court in the *PRDC* case, and the courts have recognized that the institution of proceedings prior to the operating license stage is not mandated whenever an unresolved safety question is raised after issuance of a construction permit. *Porter County Chapter, supra,* 605 F.2d at 1367, 1369. In the words of the Court of Appeals for the District of Columbia Circuit, "permitting continued construction of the plant despite unresolved safety questions does not of itself pose any danger to the public health and safety". *Id.* at 1369. Thus, in the absence of extraordinary circumstances, I will not upset the Commission's usual two-

[2]The Byron facility is currently undergoing an operating license review. Notice of receipt of application and availability of opportunity for a hearing was published in the *Federal Register* on December 15, 1978 (43 FR 58659). An operating license hearing will be held for the Byron facility and the League has been admitted as an intervenor in that proceeding.

stage licensing process by instituting a proceeding in response to a 10 CFR 2.206 petition to consider issues that are properly within the scope of the operating license review. *Public Service Company of Indiana* (Marble Hill Nuclear Generating Station, Units 1 & 2), DD-79-21, 10 NRC 717, 720 (1979). This approach is in accord with the basic principle established by the Commission that 10 CFR 2.206 is not to be used as a device to avoid a forum in which issues "more logically should be presented". *Consolidated Edison Company* (Indian Point, Units 1-3), CLI-75-8, 2 NRC 173, 1977 (1975). The League has raised many of the same issues in its petition under 10 CFR 2.206 as it has posed as contentions before the Licensing Board. *See* LBP-80-30, 12 NRC 638 (Dec. 1980).

In this context, issues such as those raised by the League concerning detailed design features of the plant do not normally warrant suspension of construction, because such issues are resolved during the operating license review. As each of the remaining portions of the League's request which I am considering in this decision, namely subparagraphs (a), (b) and (c), essentially request consideration now of questions which are more properly examined at the operating license stage, the petition may be denied on that ground alone. Nevertheless, I have examined each of the six safety areas which are addressed in the petition to determine if there are any exceptional circumstances which would warrant consideration of any agency action at this time.

Systems Interaction (Issue 1)

The League's concern is that the "systems interaction issue" may not receive adequate consideration in connection with the Byron facility. The League in its petition identifies the systems interaction issue as dealing with related problems of accident and safety analysis. These include the lack of systems interaction analysis, the lack of multiple or "common-cause" failure analysis, and the tendency of the "single-failure criterion" to exclude a large number of potential accident causing events.

Sandia National Laboratories, under contract assistance to the NRC staff, began the first of an intended two-phase program in May 1978 to evaluate whether present review procedures and safety criteria ensure an acceptable level of redundancy and independence for systems required for plant safety. This study was organized to provide an independent investigation of safety functions, and systems required to perform these functions, in order to assess the adequacy of current review procedures. It was conducted by evaluating the potential for undesirable interactions between and among systems.

Phase I of the work was completed with the issuance of a report by Sandia National Laboratories, "Final Report - Phase I Systems Interaction Methodology Applications Program," NUREG/CR-1321, SAND 80-0384, (April 1980). The results of that report generally support continued use of NRC review procedures so long as system interaction assessments are not restricted to the safety systems identified in the Standard Review Plan.

A follow-on program for FY '81 and FY '82 was developed by the staff with technical support from Battelle and Brookhaven Laboratories to prepare interim regulatory guidance for systems interaction evaluation in Light Water Reactor (LWR) Plants. It is further intended to test this guidance during FY '82 on several pilot LWRs, for purposes of identifying and evaluating significant systems interactions. The results of this effort will be reflected in final regulatory guidance. Systems interactions that are determined to be adverse as a result of this effort will be evaluated for other plants not included in the pilot program.

In summary, the issue of systems interactions has been and is still under current study by the staff. Our contractor's findings indicate that with prudent consideration of interaction of systems identified as important to safety with non safety systems, the Standard Review Plan (SRP), which is being used to conduct the Byron review, ensures a substantive review of the safety aspects of systems interactions. To the extent continued investigations by the staff identify a need for modifications to the SRP or to the Byron facility, such modifications will be applied to Byron as needed and when identified.

Steam Generator Tube Integrity (Issue 2)

Another concern of the League is that technical issues related to the integrity of steam generator tubes may not receive adequate treatment in connection with the Byron facility. The technical issues center around whether or not the Westinghouse steam generator tubes have the capability to maintain their integrity during normal operation and postulated accident conditions. In addition, the League is concerned that requirements for increased steam generator tube inspections and repairs may result in significant increases in occupational exposures to workers.

Corrosion resulting in steam generator tube wall thinning has been observed in several Westinghouse and Combustion Engineering plants for a number of years. Major changes in the design of the secondary water treatment process essentially eliminated this form of degradation. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a buildup of support plate corrosion products in the annulus between the tubes and the support plates. This buildup eventually results in reduction of the cross sectional area of the openings in the tubes, called "denting", and deformation of the tube support plates. This phenomenon has led to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and cracking of U-bend sections of tubes which were highly stressed because of support plate deformation. Task A-3 in NUREG-0410[3] has been organized to provide resolution of the problem of tube degradation due to denting in Westinghouse steam generators.

The question of steam generator tube integrity is an issue which will be reviewed by my staff for Byron. In the recently licensed Sequoyah facility, which uses Westinghouse steam generators, the staff closely examined this very question. They determined that specific measures adopted there, such as changes to steam generator design features, providing for a secondary water chemistry control and monitoring program, adding condensate demineralization features and the careful selection of condenser tubing materials, would minimize the onset of steam generator tube problems.[4] In addition, inservice inspection provisions and Technical Specifications requirements for actions to be taken in the event that steam generator tube leakage occurs during plant operation provided a sufficient basis for a conclusion that Sequoyah Units 1 and 2 could be safely operated prior to the ultimate resolution of the steam generator tube integrity issue.

Task A-3 is expected to result in improvements in our current requirements for inservice inspection of steam generator tubes. These improvements will include a better statistical basis for establishing inservice inspection program requirements and consideration of the cost/benefit of increased inspections.

The staff has not yet completed its review of the steam generator design for the Byron facility. However, they will review design and associated operating provisions, inservice inspection provisions and Technical Specifications requirements. As for Sequoyah, the staff will determine whether there is a sufficient basis for a conclusion that Byron Station Units 1 and 2 can be operated prior to the ultimate resolution of the generic issue.

In summary, the issue of steam generator tube integrity has been handled on a case-by-case basis in the operating license review and is still under

[3]NUREG-0410 Appendix F, Task A-3 *Westinghouse Steam Generator Tube Integrity.*
[4]NUREG-0011 *Supplement No. 1, SER Related to Operation of Sequoyah Nuclear Plant, Units 1 and 2,* February 1980, pages C.9 and C.10.

study by the staff. Also, Westinghouse has introduced improvements in its steam generator designs. Some of these improvements have been incorporated in facilities for which operating licenses have been issued, and improvements introduced in the Byron steam generators will be reviewed. The petition on this issue raises no new concerns that are not already being pursued by the staff in Operating License reviews.

Equipment Qualification and Deterioration (Issue 3)

The League's concern here is whether there is sufficient assurance that safety-related equipment for the Byron facility will function in the manner intended when subjected to environmental service conditions under anticipated normal, abnormal and accident conditions. As a result of work completed under Task A-24, NUREG-0410,[5] the staff has issued its proposed resolution of this issue for reactors under licensing review in its report, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," NUREG-0588, dated December 1979. Also the Division of Operating Reactors issued "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines) in November 1979. On May 23, 1980, the Commission issued a Memorandum and Order that is applicable to the Byron facility. In that Memorandum and Order the Commission stated its findings that the current NRC requirements in the environmental qualification area, including compliance with the staff's position on equipment qualification (NUREG-0588), provided reasonable assurance that the public health and safety would be adequately protected. *Petition for Emergency and Remedial Action,* CLI-80-21, 11 NRC 707 (1980). The Commission stated: "Futhermore, pursuant to Section 161(b) of the Atomic Energy Act and based on the record in this proceeding the Commission is ordering today that these two documents (NUREG-0588 and DOR Guidelines) form the requirements which licensees and applicants must meet in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criteria (GDC)-4 which relate to environmental qualification of safety-related electrical equipment". This Memorandum and Order finalized the framework within which the issue will be reviewed. Given the absence of any new significant information on this issue in the petition, this issue is properly left for examination at the operating license review for the Byron facility.

[5]NUREG-0410 Appendix F. Task A-24 *Environmental Qualification of Safety Related Electrical Equipment.*

Evaluation of Potential Accidents and Corrective Measures (Issue 4)

The League's concern here is that the consequences of serious accidents, including those labelled "Class 9", may not receive adequate treatment in connection with the operating license review of the Byron facility. Specifically, the League calls for the development of liquid pathway interdiction systems for the Byron Station using models employed in a 1980 draft Sandia study.[6] This request is made by the League based on their understanding that the Sandia authors calculate large doses "in the case of plants with hydrogeologic features similar to the Byron site (i.e., high groundwater table and permeable rock)".

The League contends that "the potential radiation dose is approximately 2 to 5 x 10^7 person-rem (with uncertainties of an order of magnitude), which current studies translate into several thousand probable deaths from a major accident."

The draft Sandia report, January 1980, was developed as a step in a study performed by the NRC. As a draft report, it has been available for information but has not in any way been adopted by the staff or the Commission as representing a basis for regulatory decisions. Its availability for almost a year prior to issuance of the Summer DES did not result in the staff's reliance on it in the Summer analysis and the staff will not rely on it in its review of Byron. The staff will continue to review that draft report and subsequent reports by Sandia, as well as pertinent information from any other source, to determine whether the staff's approach in environmental reviews needs to be modified.

As I understand the League's petition, the League is asking that I initiate separate proceedings to consider Class 9 accidents, because the League does not believe that the environmental review for the Byron operating license will be subject to the Commission's new interim policy on consideration of severe accidents. A separate proceeding is unwarranted, since the Byron environmental review is indeed subject to the new policy.

In past NEPA reviews the staff's treatment of environmental consequences of postulated accidents has been guided by the Commission's proposed Annex to Appendix D of 10 CFR Part 50 (the "Annex") that was published for comment on December 1, 1971. *Consideration of Accidents in Implementation of the National Environmental Policy Act of 1969,* 36 FR 22851 (1971). The Commission issued a Statement of Interim Policy on June 13, 1980 in

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[6]Sandia Study (Draft) for U.S.N.R.C., "Effect of Liquid Pathways on Consequences of Core Melt Accidents", January 1980.

which it announced the withdrawal of the proposed Annex to Appendix D of 10 CFR Part 50.[7] It also announced its position that its Environmental Impact Statements shall include considerations of the site-specific environmental impacts attributable to accident sequences that lead to release of radioactive materials, including sequences that can result in inadequate cooling of the 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 environmental consequences of such releases. Under the Commission's guidance, "releases refer to radiation and/or radioactive materials entering environmental exposure pathways, including *air, water, and ground water.*" 45 FR at 40103 (emphasis added).

Although the environmental review for the operating license stage has not been started, the staff will follow the guidance of the interim policy statement of June 13, 1980. In accordance with the interim policy statement, the staff will undertake a more extensive analysis of severe accidents in the environmental review of the Byron operating licenses. Such analysis must include consideration of releases to air and liquid pathways, as the League requests. It should be noted that the League, as a party to the operating license proceeding, has raised contentions on Class 9 accidents before the Licensing Board. *See* LBP-80, 12 NRC 683, 692 (1880).

With regard to the liquid pathway analysis for the Byron facility, it is expected that implementation of the new policy will result in the application of a method of analysis similar to that used for the Virgil C. Summer Nuclear Station.[8] The approach in the Summer evaluation was one of determining whether or not the Summer site liquid pathway consequences would be unique when compared to land-based sites considered in the "Liquid Pathway Generic Study" (LPGS).[9] This approach is conservative and provides bounding calculations. The LPGS was completed by the staff using realistic values of site parameters throughout, and the staff did not take into account mitigative measures. The actual method used in applying this approach to the Summer Plant consisted of a direct scaling of the LPGS population doses based on the relative values of key parameters characterizing the LPGS "small river" site and the Summer site. The individual and population doses for the liquid pathway in the LPGS ranged from a substantial fraction to a small fraction of those that can arise from the airborne pathways. The staff conclusion

[7] *Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969,* 45 Fed. Reg. 40101 (June 13, 1980).
[8] NUREG-0534 Supplement, *Draft Environmental Statement related to the operation of Virgil C. Summer Nuclear Station, Unit No. 1,* Docket No. 50-395, November 1980.
[9] NUREG-0440, *Liquid Pathway Generic Study,* February 1978.

was that the Summer liquid pathway contribution to the population dose had been demonstrated to be the same order of magnitude as that predicted for the LPGS river site, which represents a "typical" river site. The staff then noted that there would be ample time to take measures which could be taken to minimize the impact of the liquid pathway. Such measures might include slurry walls and well-point dewatering systems to isolate the radioactive contaminants at the source. These are conventional means for controlling and limiting groundwater movement used in civil engineering and earthwork construction.

In connection with its request for analysis of Class 9 accidents, the League expresses concern that systems interaction evaluation and modifications and an Interim Reliability Evaluation Program (IREP) are not being implemented for the Byron Station. As explained in the NRC Action Plan, the IREP is a pilot program which utilizes a few typical plants to determine whether changes need to be made in the review of each plant.[10] There is no pressing reason why Byron should be selected in lieu of another plant as one of the typical plants for the pilot program.

The Commission has not mandated that the IREP be applied to plants like Byron which are under review for operating license. The Action Plan describes a gradual implementation of IREP and studies for operating reactors, but IREP is not part of the requirements for new operating licenses outlined in NUREG-0737, *Clarification of TMI Action Plan Requirements.* In this regard:

> "the Commission has concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0737 can provide a basis for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured by the NRC Staff against the regulations, as augmented by these requirements. In general, the remaining items of the Action Plan should be addressed through the normal process for development and adoption of new requirements rather than through immediate imposition on pending applicants." *Further Commission Guidance for Power Reactor Operating Licenses - Revised Statement of Policy,* 45 FR 85236, 85238 (Dec. 8, 1980).

In summary, the staff is reviewing the issue of potential accidents and potential need for liquid pathway interdiction, both in a generic sense and on a case specific basis for Byron. The staff will conduct its review of Byron

[10] *NRC Action Plan Developed as a Result of the TMI-2 Accident,* NUREG-0660, Vol. 1, at pp. II.C.-2 to II.C.5 (May 1980).

in accordance with the Commission's Statement of Interim Policy of June 13, 1980. If new facts from the Sandia study, the systems interaction study, the IREP program or from other sources are shown to pose questions as to the validity of the current approach, an assessment will be made and, as appropriate, the review requirements modified and applied to Byron.

Conformance to Current Regulatory Practices (Issue 5)

The League's concern here is that the staff assessment of the Byron facility at the operating license review will consist of less than a complete review of plant design against all current Regulatory Guides and safety standards. Also, the League is concerned with construction quality assurance and control based upon its perception of the findings of the Office of Inspection and Enforcement's Inspection Reports.

In regard to compliance with Regulatory Guides, the League cites exceptions to Regulatory Guides taken by the applicant in the Byron Final Safety Analysis Report (FSAR) as demonstration of non-compliance with the identified Regulatory Guides. However, as Regulatory Guides are not regulations but merely *one* means of satisfying a regulatory requirement, conformance with Regulatory Guides is not a prerequisite to the issuance of any Commission license. The applicant, by submitting the FSAR with these exceptions, must demonstrate that the Commission's regulations will be met in the appropriate area by means other than exact conformance with the applicable Regulatory Guides. Such proposed alternative means of complying with the Commission's regulations must be determined to be acceptable prior to a decision on issuance of an operating licenses for the Byron facility. In no event will my staff recommend issuance of an operating license with respect to the Byron facility unless all safety requirements are satisfactorily met.

In summary, the staff routinely reviews exceptions to Regulatory Guide positions to determine that the applicable regulations of the Commission are met by other means.

A second League concern relative to conformance to current regulatory practices is a concern as to whether the Byron Quality Assurance/Quality Control (QA/QC) programs are satisfactorily carried out in practice. Issues related to the implementation of QA/QC programs have been reviewed by the Office of Inspection and Enforcement (I&E). That Office has conducted an assessment of the Byron QA/QC program relative to the concerns raised by the League. That assessment was provided in a memorandum to the Commissioners on March 9, 1981 in response to a request by the

Commission following its review of my partial denial on December 22, 1980 of the League's petition. Copies of this memorandum were served on counsel for the League and Commonwealth Edison. A copy of that memorandum is attached to this denial.

In summary, the assessment by the Office of Inspection and Enforcement indicates that, from mid-1979 to date, the Byron QA/QC program generally has been effective and, in the staff's judgment, the information presented in the League's petition does not support the League's allegation that the Byron QA/QC program was not effective.

The Commission during its review under 10 CFR 2.206 of my partial denial had noted the subsequent issuance by the Director, Region III, of an Immediate Action Letter confirming suspension of work on electrical cable installation. The assessment by I&E was that the limited scope of that problem made limited stop-work the most appropriate technique to achieve corrective action and that there was no necessity for a total suspension of work at the Byron site. When the memorandum was forwarded to the Commission in March, the licensee had already met its commitments under the Immediate Action Letter and that constraint had been lifted.

In summary, the I&E staff is monitoring the implementation of the Byron QA/QC program for construction and is taking enforcement action appropriate to problems that occur. I have no reason to believe that I&E will not continue to take actions appropriate to any future problems that may arise including suspension of construction, if warranted.

Open Generic Issues (Issue 6)

The League in its affidavit cites the foregoing five issues as "...example[s] of a large number of such issues, all of which the NRC has identified (sometimes repeatedly) and a sizable portion of which NRC has repeatedly termed high-priority matters."[11] The League cites the November 1977 decision, *Gulf States Utilities Co.* (River Bend Station, Units 1 and 2, ALAB-444, 6 NRC 760, 774-75 (1977), by the NRC's Atomic Safety and Licensing Appeal Board as imposing an affirmative duty on the NRC staff to identify generic safety issues, and evaluate their impact on plant safety. Although not noted by the League, the Atomic Safety and Licensing Appeal Board in *Virginia Electric Power Company* (North Anna Nuclear Power Station Unit 1), ALAB-491, 8 NRC 245, 247-50 (1978) later specified that staff conclusions are required with respect to unresolved generic issues

[11]Petition, page 64.

for issuance of an operating license. The staff routinely reviews each facility prior to the issuance of an operating license to evaluate the safety significance of unresolved safety issues with respect to that facility. These unresolved safety issues have already been determined generically to not pose imminent public health and safety concerns so as to prohibit continued operation or continuation of licensing actions. In accordance with ALAB-491, the staff provides further explanation of the basis for licensing a particular facility in the absence of the long term generic resolution of these issues. The staff's findings are presented in the Safety Evaluation Report (SER) for each facility. Such SERs have been developed for issuance of operating licenses to applicants for stations generally similar to Byron that use Westinghouse-designed nuclear reactors, *e.g.*, North Anna Unit 2, Sequoyah Unit 1 and Salem Unit 2. The League presents no arguments that Byron would be substantially different from facilities already reviewed. Therefore, this issue is one which may be left for review at the operating license stage.

Finally, the League also expressed concerns about "institutional disincentives to safety including a concern that assertion by the NRC staff of safety concerns, particularly those that may be controversial, is most unlikely to advance one's career and is far more likely to result in stigmatization and 'career paralysis.' "[12] The Commission recognizes the potential problems in this area and has taken strong action to formulate a policy to preclude such institutional disincentives from affecting the NRC staff.[13] One of the prime objectives of that policy is to provide sanctions against employees who take retaliatory actions with respect to differing professional opinions within the NRC staff.

Conclusion

I have determined for the reasons set forth above that there exists no adequate basis for instituting a proceeding pursuant to 10 CFR 2.202 to inquire why the Byron Station, Units 1 and 2, construction permits should not be:

(a) Modified so that construction may not proceed without resolution of all outstanding safety problems applicable to the Byron Station;
(b) Suspended or revoked until such time as Commonwealth Edison has formulated an acceptable and realistic plan for resolving all outstanding safety problems; and

[12] Petition, pages 64-65.
[13] NRC Manual Chapter 4125 "Differing Professional Opinions."

(c) Revoked if the outstanding safety problems cannot be resolved prior to the completion of construction due to Edison's financial condition or for other reasons.

Accordingly, the November 21, 1980 request of the Rockford League of Women Voters is denied.

A copy of this decision will be filed with the Secretary of the Commission for its review in accordance with 10 CFR 2.206(c) of the Commission's regulations. In accordance with 10 CFR 2.206(c) of the Commission's Rules of Practice, this decision will constitute the final action of the Commission 25 days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 7th day of May, 1981

United States
Nuclear Regulatory Commission
Washington, D.C. 20555

MARCH 9, 1981
MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne

FROM: William J. Dircks,
Executive Director for Operations

SUBJECT: BYRON STATION UNITS 1 AND 2
QUALITY ASSURANCE PROGRAM
(SECY 81-56)

On November 21, 1980, the Rockford League of Women Voters filed a request pursuant to 10 CFR 2.206 and 2.202 (petition) seeking that the Director, Office of NRR, take certain actions with respect to the Byron Nuclear Power Station. On December 22, 1980, the Director partially denied that portion of the petition seeking an immediate halt to construction. The Commission, during its review under 10 CFR 2.206 of this partial denial by the Director, determined that the Director had failed to set forth his evaluation of one of the claims in the petition. The claim not addressed was that the Quality Assurance/Quality Control (QA/QC) program at Byron is not being effectively implemented. The Commission also noted the recent issuance by the Director, Region III, of an Immediate Action Letter confirming suspension of work on electrical cable installation.

The Commission directed that the staff provide "its reasoned evaluation of the petitioner's allegations" on the effectiveness of the licensee's QA/QC program at Byron. Attached is an assessment by Region III of the status and effectiveness of the QA/QC program currently in effect at the Byron site. In summary, the assessment states that from mid-1979 to date, the Byron QA/QC program generally has been effective, and that the recent Immediate Action Letter represents a specific action with respect to a limited scope of work (electrical cable and cable support installation) by a single contractor. The limited scope of the problem makes limited stop-work the most appropriate technique to achieve corrective action. In the opinion of the staff, there is no necessity for and no value to a total suspension of work at the Byron site. It should be noted that the licensee has already met its commitments under the Immediate Action Letter, and that constraint has now been lifted.

The staff has also examined the affidavit accompanying the League's petition which claimed that the Byron QA/QC program was not effective. Section 2 of the evaluation addresses the affidavit. In the staff's judgment, the information presented does not support the allegation made.

In summary, the various individual violations referred to in the affidavit represent relatively minor offenses against a QA/QC program which is basically sound. The situation addressed in the Immediate Action Letter dated January 13, 1981 does represent a breakdown in the QA/QC program for a single construction contractor and the associated inspection contractor. This can and should be addressed specifically without requiring suspension of unrelated work, and a halt to all construction at the site is not called for under the criteria set forth in the Interim Enforcement Policy (Section IV.C.2). Although the Immediate Action Letter constraint has now been lifted, further enforcement action is being considered by the staff consistent with the Interim Enforcement Policy, to emphasize to the licensee the need to take lasting corrective action.

William J. Dircks
Executive Director for Operations

Enclosure:
Evaluation of
Byron Nuclear Power Station
QA/QC Program

[Enclosure]

EVALUATION OF
BYRON NUCLEAR POWER STATION
QA/QC PROGRAM

1. Past and Current Performance

a. The Licensee's Quality Assurance Program was assessed in a Region III mid-term QA inspection of Byron in late 1979. This involved a thorough evaluation of the QA Program in the areas of site quality assurance, including auditing, trending, surveillance activities, and the control of nonconformances; corporate quality assurance, including auditing, trending, and interfaces; design and design change control; procurement, including procedures, purchasing, and auditing of suppliers; and control of purchased materials, including procedures, receipt inspection, storage and maintenance; and surveillance of site contractors. That evaluation did not produce any evidence of a breakdown of the QA Program in the above areas.

b. The Licensee's overall regulatory performance was recently reviewed in the Systematic Assessment of Licensee Performance (SALP) appraisal (mid-1979 - mid-1980). The assessment included a review of the Byron QA/QC program implementation. While some deficiencies have been identified with QA/QC, the number and nature of these problems were not viewed to be significant from a regulatory standpoint. In this regard, it should be noted that the number of noncompliances at Byron were fewer than at most construction projects within Region III. Furthermore, those deficiencies that have been identified were, for the most part, isolated deficiencies and not symptomatic of a broad problem.

c. In addition to the above, the Quality Assurance Program is continually assessed during the performance of the construction inspection program. Occasionally during such inspections, quality assurance problems in limited areas are found which are not indicative of an overall Quality Assurance Program breakdown, but are important enough to cause issuance of a stop work order in the area(s) involved. Although these cases are undesirable, they do not represent a breakdown in the overall Quality Assurance Program.

A recent example of such a case involved electrical work at Byron. Inspection indicated a problem with the QA/QC program in the electrical area. We concluded that it was necessary to stop work in this area, reexamine work performed to date, correct identified problems, and upgrade the QA/QC program for this work. We discussed our concerns with the licensee, Commonwealth Edison Company. The licensee agreed to suspend safety-related electrical work pending resolution of the problems. This agreement was confirmed by Region III's Immediate Action Letter.

More examples of this kind may occur before construction at Byron is complete; appropriate action will be taken.

2. Comments on QA/QC Concerns Contained in the League of Women Voter's Petition of November 21, 1980

a. The affidavit attached to the League's petition notes at p. 64 that IE inspectors expressed concern over "excessive rework" at Byron in mid-1980. It should be noted that extensive rework in itself is not a citable offense, as long as effective control of the rework is maintained by the licensee. Region III's concern was that extensive rework increases the difficulty of maintaining effective control of the construction activities. The in-depth evaluation performed by Commonwealth Edison in response to the Region III inspection findings indicates that the extensive rework has not resulted in (or from) a breakdown in the site QA Program.

b. Unresolved items are referred to as "problem areas" at p. 62 of the subject affidavit. The staff does not necessarily consider them as such. Unresolved items are matters about which more information is required to determine whether they have possible safety implications or are acceptable. In the case of Byron, unresolved items have been pursued by the staff and none of the items, either singularly or collectively, would warrant an order to stop all work at the Byron site.

c. The affidavit states at p. 62 that "inadequate welding has been a continuing problem" and refers to seven IE inspection reports during the August 1978 to June 1980 time period. Our review shows that three of the seven reports do not contain noncompliances related to welding, and that the other four reports contain a total of five noncompliances and one deviation which are welding-related. The five noncompliances involved procedure and record discrepancies, and the deviation involved an interpretation of ASME Code requirements regarding procedure qualification. No evidence of actual inadequate welding was noted. In view of the number and nature of the welding-related noncompliances cited, we do not consider that welding has been a significant problem at Byron.

d. The affidavit document refers at p. 62 to a May 1979 IE inspection report which cited as a "recurrent item" the inadequate protection of "important equipment...which has caused damage to equipment." Neither the noncompliance cited in May 1979, nor the previous noncompliance in April 1979 which made the second one a "recurrent item," involved actual damage to equipment, but rather the possibility of damage if the conditions were not corrected. Both noncompliances were closed out in subsequent IE inspection reports based on corrective actions taken by the licensee and the site contractors involved.

A later IE inspection at Byron in March 1980 (Reference 211 of the affidavit) did find installed mechanical snubber assemblies which had been damaged because of inadequate protection. The licensee at that time removed the damaged snubbers, voluntarily stopped the installation of safety-related snubber assemblies, and agreed to not reinitiate installation until the construction environment is such that the snubber assemblies can be adequately protected.

In view of the small number of noncompliances cited in this area, and the licensee's positive and prompt corrective actions, the staff does not consider that storage, cleaning and preservation of equipment has been a significant problem at Byron.

In conclusion, the information provided in the affidavit accompanying the League's petition presents nothing new. The staff is closely monitoring the construction activities at the Byron site and continues to be of the view that a halt in construction is not warranted.

Cite as 13 NRC 748 (1981) **DD-81-6**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

James H. Sniezek, Acting Director

In the Matter of **Docket No. 50-344**
(10 CFR 2.206)

PORTLAND GENERAL ELECTRIC COMPANY
(Trojan Nuclear Plant, Unit 1) **May 21, 1981**

The Acting Director of the Office of Inspection and Enforcement denies a request by the Trojan Decommissioning Alliance that the Trojan Nuclear Plant be shutdown immediately due to electrical problems experienced at the site.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By mailgram dated April 20, 1981, the Trojan Decommissioning Alliance requested that the U.S. Nuclear Regulatory Commission (NRC) order the immediate shutdown and launch an investigation of the Trojan Nuclear Plant of the Portland General Electric Company. This request has been considered under the provisions of 10 CFR 2.206 of the Commission's regulations.

The request by the Alliance for the shutdown and investigation of the Trojan Nuclear Plant was based on six events at the Trojan plant involving electrical equipment. All of these events have been investigated in accordance with normal NRC inspection procedures. In each case, the results of the investigations were documented in inspection reports. In one case, the inspection resulted in an enforcement action being taken against Portland General Electric Company by the NRC. The relevant details of each of these events are discussed below:

1. On April 20, 1981, an electrician attempted to use a multi-meter, mistakenly set to a milliamp scale, to measure the voltage across the 480-

volt terminals of a motor control center. The low voltage device applied to the high voltage terminals had the effect of a short circuit, which caused an electrical flash in a breaker and resulted in burns to the electrician's face and hands. This individual was taken to the local hospital where he was treated for first and second-degree burns of the left hand. Flash burns of the face, neck, and chest were also treated. The individual returned to work on May 5, 1981.

There was minor damage to the motor control center that caused a loss of normal feedwater control. This resulted in an automatic reactor shutdown when the steam generator water level decreased to a low level set point. The NRC resident inspector reviewed the event and summarized his findings in IE Inspection Report No. 50-344/81-09 as follows:

"On April 20, 1981, while troubleshooting starting problems of Service Water Booster Pump C, a plant electrician inadvertently shorted over the phases of a breaker associated with the pump. The worker was attempting to measure the voltage between the B phase and ground with a digital multimeter. The worker apparently had used the milliampere jack which would have caused a direct short through the meter. The arc drawn on the B phase stab of the breaker caused a flashover to the other phases, and, thereby, shorted the breaker out. The flash caused superficial 1st and 2nd degree burns on the workman's left hand and chest.

"The shorting out of the breaker caused a feeder breaker to Motor Control Center B25 to open. The loss of power to B25 interrupted power to instrument busses Y01 and Y03 which provide control power to nonvital loads. This caused the recirculation valves on the condensate pumps whose control power is supplied by Y01 and Y03 to open, this subsequently lowered flow of feedwater to the steam generators and the reactor tripped on receipt of a low low level signal in C steam generator.

"Power was subsequently restored to Y01 and Y03, meanwhile the plant responded normally to the trip with all safety-related equipment available. The licensee after determining the cause of the trip, and placing affected equipment back in service, brought the plant back to power operation approximately seven hours after the plant trip.

"No items of noncompliance or deviations were identified."

No safety-related equipment was affected by this event other than the service water booster pump, which was out of service for the trouble-shooting activities already described.

2. On February 26, 1981, a blown fuse, located in the central board annunciator test circuit, caused all of the alarm lights on the associated panels to illuminate. This test circuit is provided so that all of the alarms on the panel can be tested by pushing a single button. In this instance, the malfunction of the test circuit caused all of the panel alarms to remain energized. The operator, who was unable to reset the energized alarm lights, declared the control board annunciators out of service and initiated a local emergency. This was a prudent response under the circumstances and no unsafe actions were taken. As it turned out, no safety-related equipment had actually malfunctioned during the 15 minutes that the annunciators were declared inoperable. The subject alarm panel also remained functional because it is designed so that an actuated alarm will indicate subsequent alarms by flashing on and off. The NRC Inspector's review of this event was summarized in Inspection Report No. 50-344/81-09 as follows:

   "The inspectors examined the licensee action taken as a result of the control board annunciators failing to reset after testing on Thursday, February 26, 1981. The circumstances were initially classified as an unusual event as described in the licensee's emergency response plan. Testing performed by the licensee since the event has indicated that the alarm features of the control board annunciators were still functional and alarms would have been indicated in the control room even though the reset feature following a system test was not functional. The reset feature failed to function as a result of a blown fuse in the test/reset button circuit. The fuse was rated at 4 amps. Measurements of the test current made by the licensee indicate that undernormal test conditions the annunciator test/reset circuit draws approximately 4 amps. This situation was discussed with the manufacturer and a recommendation was made to install a 5 amp fuse. The 5 amp fuse was installed and no problems have occurred as a result of periodic annunciator testing since the February 26, 1981 incident. The licensee has initiated facility design change, RDC 81-042, which when completed will reorganize the annunciators into five separate zones, each zone having its own test/reset button circuit. This design change will significantly reduce the current through the test/reset circuit and at the same time prevent a single fuse failure from affecting all annunciator test/reset circuits.

"No items of noncompliance or deviations were identified."

3. During a surveillance test of a containment spray pump on January 22, 1981 the cooling fans associated with the pump failed to start automatically due to a tripped breaker (Licensee Event Report 81-04). The fans started when the breaker was reset. Subsequent electrical testing showed that the fans operated normally. The NRC resident inspector reviewed the event and summarized his findings in Inspection Report No. 50-344/81-08 as follows:

   "The licensee has examined the circuitry for the B containment spray pump fan cooling unit thermal overload protection. No excessive current or faults could be found, and the exact cause could not be determined. After resetting the thermal overloads, the fans started and functioned normally. Additional investigative effort by the licensee had included placing a recorder in the circuitry to detect any possible overloads; no indications were recorded during the period the recorder was in use.

   "No items of noncompliance or deviations were identified."

   This event did not have a significant effect on safety because the building ventilation was adequate to prevent the spray pump from overheating without the fans. In addition, a redundant containment spray pump remained fully functional.

4. On December 12, 1980, a blown fuse was found in the breaker control power circuit for the vent supply fan for the turbine-driven auxiliary feedwater pump room (Licensee Event Report 81-01). The ventilation fan operated satisfactorily when the fuse was replaced. The NRC resident inspector's review of this event was summarized in Inspection Report No. 50-344/81-05 as follows:

   "The control power circuit to the ventilation supply fan was checked by licensee personnel for shorts, grounds, or overcurrent conditions. No faults could be found that would have blown the fuse, and the fan tested satisfactorily after the fuse was replaced.

   "No items of noncompliance or deviations were identified."

   This ventilation fan is designed to start if the turbine-driven auxiliary feedwater pump is needed and is provided to prevent this feedwater pump from overheating.. However, the temporary loss of this fan is of minimal safety significance since adequate ventilation could have been supplied to the turbine-driven feedwater pump by opening the door to the pump room. A second auxiliary feedwater system was also available and remained fully funcitional.

5. On December 31, 1980, repeated cycling of worn contacts in a motor control center breaker caused electrical arcing that ingnited dust and a small plastic dust collector located near an electrical bus (Licensee Event Report 80-28). The fire, which was promptly extinguished, caused minor damage to the bus work. The motor control center that supplies power to certain safety-related equipment remained functional, but was removed from service for five and one-half hours while repairs were made. Removal of this equipment from service for limited time periods is permitted by the plant technical specifications because backup equipment is available for each of the affected safety components. The resident inspector reviewed the event and the corrective action. The results of the review documented in Inspection Report No. 50-344/81-09 are as follows:

   "LER 80-28 (Closed): The licensee completed an engineering evaluation of the circumstances and corrective action taken regarding the breaker fire in Motor Control Center B22. The evaluation concludes that the switchgear is properly sized and should safely handle the currents associated with operation of connected plant loads. The study recommends that preventive maintenance procedures be examined to provide for the inspection of the bus work in the vicinity of the stab connection when breaker maintenance is performed. Should the bus work tin plating be worn off, the resulting contact between the aluminum base metal and the stab connector would provide a higher than normal resistance contact. This condition in conjunction with the rapid cycling of the load such as repeated starting of a large electrical motor, could result in contact overheating. The study also recommends that operating procedures be reviewed to verify that they contain appropriate limitations on the number of starts large electrical motors are permitted in specified time intervals. This limitation is primarily for motor protection, but should also minimize the potential for bus work heating during starting current transients. The procedure reviews and any additional testing of the switchgear (if required) will be completed during the 1981 refueling outage.

   "No items of noncompliance or deviations were identified."

6. On October 3, 1980, the licensee discovered that the automatic start signal, which starts both trains of auxiliary feedwater pumps on low

steam generator water level, had been disabled by a wiring error (Licensee Event Report 80-20). The discovery was made when the steam generator water level decreased to the signal actuation point following a reactor trip. (The water level drop resulted from water shrinkage from cooldown and is expected in this type of reactor.) The auxiliary feedwater pumps were started manually by the operators as part of their routine procedures, although in this case they were not needed.

The error occurred when the electrical leads transmitting the signal were connected to the wrong terminals following the 1980 refueling outage. Although the problem was caused by a human error, the main concern in this instance was that the error was not detected by the testing that is required to be conducted on all safety-related systems following maintenance. This event was reviewed in considerable detail by NRC. The site inspector's review was summarized in the Inspection Report No. 50-344/80-29 as follows:

"The inspectors examined the long term corrective action taken by the licensee to preclude recurrence of the incident described in the LER. The basic cause of the LER was the personnel wiring error which was not detected by an appropriate test when the safety-related automatic start of the auxiliary feedwater pumps was reconnected following the completion of maintenance in the steam generators. Facility procedures require the testing of all safety-related systems, structures and components upon return to service following maintenance. In actual implementation of these procedures, the testing of safety-related equipment upon return to service was limited to the systems, components and instrumentation specified in the technical specifications. The low level automatic start feature of the auxiliary feedwater pumps was not a function specified by the facility technical specifications. The licensee's quality assurance program is committed to ANSI N18.7-1976 which in section 5.2.6 requires that when safety related equipment is returned to service, operating personnel shall place the equipment in operation and verify and document its functional acceptability. The weakness in the implementation of procedure (AO-3-14) which limited the safety-related equipment to that specified in the technical specification is in noncompliance with the procedure itself in view of the Quality Assurance Program commitment to ANSI N18.7-1976.

"The specific correction taken by the licensee to correct the situation described above has been to emphasize and revise, as appropriate, Administrative Order No. 3-14, Safety-Related Equipment Outages, and Administrative Order No. 6-2, Bypass of Safety Functions, to clearly specify that all safety related equipment must be verified functional upon return to service following maintenance, not just safety-related equipment, components or instrumentation required by the technical specifications. As applicable, the methods for verifying functionability include, operation of the component in accordance with an approved test procedure, performance of an installation check or an independent verification check. In addition to the above, the licensee's technical specifications have been amended to require that all automatic starting features of the auxiliary feedwater system be operable and tested at a specified surveillance frequency during modes 1, 2 and 3.

"One item of noncompliance was identified by the licensee as described above. No deviations were identified."

Although this event revealed a deficiency in the licensee's procedures, it did not have a significant effect on public safety. The auxiliary feedwater system continued to be capable of performing its intended function. In this instance, the main feedwater system did not shut down, so there was no need for the auxiliary feedwater system. However, if a loss of main feedwater had occurred, an automatic start feature for the loss of the main feedwater pumps would have started the auxiliary feedwater system. In addition, the manual auxiliary feedwater start can be considered to be a reliable backup for this system because the steam generators will continue to provide adequate cooling for 30 minutes following a reactor trip (FSAR Fig. 15.2-31) without the addition of any feedwater. In this case, the control operator, following the normal post trip procedures, accomplished a manual start of the auxiliary feedwater in about one minute.

The number of electrical equipment problems reported by Portland General Electric over the time period covered by these events is not considered unusual for a plant such as Trojan with its thousands of electrical components. None of these events represented a significant reduction in the level of protection provided for public health and safety. In each case, backup systems and measures were available to provide the functions of the affected components. In each case, the licensee has provided appropriate corrective action.

In my judgment, the inspections already performed by the NRC staff, as well as the corrective actions taken by the licensee, described in the previously referenced documents have adequately addressed the events identified in the Trojan Decommissioning Alliance's communication. On that basis, I deny the petition.

A copy of this Decision and its enclosures will be placed in the Commission's public document room at 1717 H Street, NW, Washington, DC 20555 and in the local public document room for the Trojan facility, located at Multnomah County Library, Social Science & Science Department, 801 SW 10th Avenue, Portland, Oregon 97205. A copy will also be filed with the Secretary of the Commission for review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

As provided in 10 CFR 2.206(c) of the Commission's regulations, this Decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission, on its own motion, institutes a review of this Decision within that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

James H. Sniezek, Acting Director
Office of Inspection and Enforcement

Dated at Bethesda, Maryland
this 21 day of May 1981.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

Victor Stello, Jr., Director

In the Matter of **Docket Nos. 50-458**
50-459
(10 CFR 2.206)

GULF STATES UTILITIES COMPANY
(River Bend Station, Units 1 and 2) **May 26, 1981**

The Director of the Office of Inspection and Enforcement denies a request by the Union of Concerned Scientists that construction at the River Bend Station, Units 1 and 2 be halted because of alleged dangerous practices that had arisen on site as the result of an accelerated construction schedule.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

By petition dated July 21, 1980, the Union of Concerned Scientists (UCS) "on behalf of an individual who wishes to remain anonymous,"* requested pursuant to 10 CFR 2.206 of the Commission's regulations that the U.S. Nuclear Regulatory Commission (NRC) halt the construction of the River Bend Station Units 1 and 2 of the Gulf States Utilities (GSU) Company. This request has been considered under the provisions of 10 CFR 2.206 of the Commission's regulations. Notice of receipt of the petition was published in the *Federal Register* on August 25, 1980 (45 FR 56476). An initial decision denying a request for immediate action to halt construction activities at the River Bend units was forwarded to the UCS on August 18, 1980.

According to the UCS petition, the individual who wishes to remain anonymous (hereinafter referred to as the alleger) identified what were

*UCS followup letter of October 6, 1980 to Victor Stello.

defined as "dangerous practices that arise from the fact that the plant is being built on an accelerated schedule, and, under pressure from that schedule, the project engineering management has taken certain shortcuts that would lead to dangerous conditions if not checked before plant cutover." The alleger then provided UCS a number of examples of these "dangerous practices." These examples addressed problems in the qualification of various electrical cables and cable trays, the use of certain specifications prior to prescribed approval, the use of standards and guides that are still in the review process, and the use of two dissimilar cables in a run to a specific power supply.

A special inspection was conducted by the Office of Inspection and Enforcement (IE) on July 30-31, 1980 to investigate the examples provided by the alleger. The findings of the inspection were documented in the enclosed IE Inspection Report No. 50-458/80-08, dated August 19, 1980 (a copy of which is appended to this decision). Each of the examples identified by the alleger was addressed in that inspection report.

The findings in the inspection report revealed that the alleged problems or deficiencies did, indeed, exist. However, each had been previously identified by either the licensee or the contractor and, again in each instance, proper disposition had been initiated as required by the Quality Assurance Program of GSU. The IE inspection confirmed the fact that the Quality Assurance Program was functioning properly, nonconformances were identified, and proper dispositions thereof were undertaken. Consequently, there is no basis for NRC citations of noncompliance or deviation on these matters.

IE Inspection Report No. 50-458/80-08 and a request for additional information related to the allegations were forwarded in a letter to UCS by the Office of Inspection and Enforcement on September 5, 1980.

In a response dated October 6, 1980, UCS forwarded additional information to the Office of Inspection and Enforcement. The response included a September 27, 1980 letter from the alleger to UCS commenting on the inspection findings, as well as additional comments by UCS on these findings.

UCS questioned the NRC basis for declining to suspend construction immediately upon receipt of the UCS petition. As was stated in the August 18, 1980 letter to UCS declining to immediately halt construction, the significant fact in this case was the very early stage of plant construction. Construction at the River Bend Station was in such an early stage that none of the equipment in question had been installed but rather was still in the procurement and delivery process. Furthermore, prior to the August 18, 1980 letter, IE had completed a preliminary investigation of the quality assurance issues raised by UCS that did not indicate an inadequate quality assurance program at River Bend. In fact, all the allegations addressed situations that had already been identified and properly dealt with by the licensee or its contractor. Thus, the quality assurance program was functioning properly by identifying deficiencies or engineering problems and tracking them for adequate resolution.

The UCS letter of October 6, 1980 also noted that the findings of IE Inspection Report No. 50-458/80-08 did not address the concern that scheduling pressures might be contributing to lax practices. This point was further discussed during a telephone conference call on October 21, 1980 between NRC, UCS, and the alleger.

As a result of the conference call, NRC agreed to perform an additional investigation to assess the effect of scheduling pressures. This investigation was performed on October 29-31, 1980. The investigation included interviews with eleven members of the engineering staff, and the findings confirmed that pressure on electrical/drawing engineers to meet schedules did exist, but not to the extent that it would cause engineers to sacrifice or compromise quality. Details of the areas covered by the investigation are documented in the enclosed IE Inspection Report No. 50-458/80-11, dated November 18, 1980 (a copy of which is appended to this decision).

The final UCS comment in its October 6, 1980 letter raised the question whether Gulf States Utilities would or would not accept Regulatory Guide 1.131 "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants." This Regulatory Guide has been issued for comment only. The River Bend PSAR commits GSU to IEEE Standard 383 "Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations." The PSAR does not commit GSU to Regulatory Guide 1.131. A licensee is not required to commit to a Regulatory Guide that has been issued only for comment nor is he required to meet the positions of a final Regulatory Guide unless he voluntarily committed to meet them or the positions have been incorporated into a regulation. It is the understanding of my staff that GSU will review the final version of the Regulatory Guide for adoption when issued; in the interim, the current practice of meeting the requirements of IEEE Standard 383 is acceptable.

Finally, NRC staff reviewed the health and safety items identified in the September 27, 1980 letter from the alleger that was attached to the UCS response of October 6, 1980. The staff's analysis is contained in the Appendix to this Decision. The analysis finds no merit in any of the health and safety items.

The results of the investigations performed by the NRC staff, as described in the documents referenced above, demonstrate that no adequate basis exists to suspend construction of River Bend Station Units 1 and 2. Consequently, the UCS petition is hereby denied.

A copy of this Decision and its enclosures will be placed in the Commission's public document room at 1717 H Street, NW, Washington, DC 20555 and in the local public document rooms at the Audubon Library, West Feliciana Branch, Ferdinand Street, St. Francisville, Louisiana 70775 and at Louisiana State University, Government Documents Department, Baton Rouge, Louisiana 70803.

A copy of this Decision will also be filed with the Secretary of the Commission for review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

As provided in 10 CFR 2.206(c) of the Commission's regulations, this Decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance unless the Commission, on its own motion, institutes a review of this Decision within that time.

FOR THE NUCLEAR
REGULATORY COMMISSION

Victor Stello, Jr.
Director
Office of Inspection and Enforcement

Dated at Bethesda, Maryland
this 26th day of May 1981.

Enclosures;
1. Appendix
2. IE Inspection Report No. 50-458/80-08
3. IE Inspection Report No. 50-458/80-11

[IE Inspection Report Nos. 50-458/80-08 and 50-458/80-11 have been deleted from this publication but are available at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.]

Appendix

NRC Staff Analysis of Questions Raised with Respect to the River Bend Station by an Anonymous Alleger in a Letter Dated September 27, 1980 to the Union of Concerned Scientists

The following include responses to the individual health and safety questions from the alleger's letter dated September 27, 1980. The alleger's questions dealing with excessive costs incurred by GSU are not proper considerations of this Commission when examining a nuclear facility with regard to its effect on public health and safety.* In some cases, the responses are reiterations of findings in IE Inspection Report Nos. 50-458/80-08 and 50-458/80-11. In all cases, the responses reflect the most current information as of March 5, 1981. The format of the analysis follows that of the September 27, 1980 letter with the allegation or question stated first and the staff response that follows:

Question
Will Okonite be permitted to ship power cable before satisfactory test results are available?

Answer
Qualification test results have been compiled and submitted by Okonite to Stone & Webster Engineering Company (SWEC) for review and approval. SWEC and Gulf States Utilities (GSU) concur that the results demonstrate the qualified life of the cable. Thus, these results are approved and Okonite is about to start cable production. No cable will be shipped prior to approval by SWEC. It may be noted that there is no statement in the GSU PSAR that qualification test results must be acceptable prior to shipment of the cable. Cable must be demonstrated to be qualified prior to use.

Question
In IE Report No. 50-458/80-08, the staff stated that site activities only involve splice identification and not qualification on cable splices. The alleger then asks how the site personnel will go about qualifying these splices.

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*Public Service Company of New Hampshire, et al. (Seabrook Station, Units 1 and 2) ALAB-623, 12 NRC 670, 677-678 (1980).

Answer
Since the referenced report was issued, Okonite has committed to furnishing "rework free" cable. The applicable purchase specifications (241.234 and 241.240) have been revised to incorporate this requirement as follows:

1. Finished cable shall not contain conductor-to-conductor splices.

2. After the insulation is extruded onto the conductor, there shall be no repairs made to the insulation.

3. Cosmetic repairs, such as buffing, to improve the outer surface of the jacket is permitted. Removal/replacement of a section of jacketing is not permissible.

Question
If SWEC, indeed, accepts with no exceptions Regulatory Guide 1.131, which was issued some years after the PSAR was issued, are we to believe that GSU and its agents accept with no objection ex post facto rule-making?

Answer
As noted in the body of the Decision, there is no requirement that GSU commit to or meet the conditions of a draft Regulatory Guide. The current practices of GSU in this area are acceptable.

Allegation
The alleger claims that a specification violation occurred with respect to thermocouple extension wire and that SWEC revised the specification to accommodate the vendor.

Answer
The specification was not revised to accommodate the vendor, but rather to clarify the necessary traceability requirements. The revised specification now requires the tests and documentation to provide that traceability. The revision was made in accordance with good engineering practice. The vendor must certify that its conductor meets the specification requirements. Certified Mill Test Reports (CMTRs) on raw copper that is 99.999% pure, including conductor resistance tests, are to be received with each cable shipment. The vendor further attests to this traceability by certificates of conformance in accordance with ASTM Standard B33.

Question
Is the procedure of specification revision referred to above acceptable to the NRC?

Answer
Revising the specifications, precedures, instructions, etc., is an acceptable procedure to NRC. The revisions must be properly reviewed and controlled, proper engineering judgment must be exercised, and quality and safety are not to be compromised.

Question
The alleger states that a number of "Nonconformance and Disposition" (N&D) reports have been issued against cable trays delivered to the site. The alleger notes that one such report had not been dispositioned over two months after issuance and asks, "Does SWEC routinely take over two months to disposition an N&D?"

Answer
The normal length of time to obtain an engineering disposition to N&Ds is one to two weeks. Procedures have been established that provide for monitoring the status of N&Ds by the Quality Systems Division. If an N&D has been awaiting disposition for a period of thirty days or more, immediate action is required to provide a status of the N&D and either to complete the disposition or describe why it is not practical to provide a disposition at that time.

In the case of the N&D in question, it was impractical to provide a disposition until seismic acceptability of the proposed repairs was evaluated. Under no circumstances should QA approve an N&D disposition that would be adverse to quality for the sake of expediency, regardless of the age of the N&D.

Question
How can the licensee assure that these nonconforming cable trays will not be used as is, and how do they propose to deal with post installation damage?

Answer
Present project procedures, specifications, and inspection plans provide for both pre- and post-installation inspection by Field Quality Control (FQC).

In addition, each individual damaged section of cable tray must have a reject tag affixed. The tray cannot be removed from storage until this tag has been removed by FQC, indicating that the cable tray is acceptable for installation.

Question
The alleger notes that certain cable trays were shipped to the site before the vendor received a sign-off from a professional engineer (PE). The alleger's question on this activity relates to the waiver of PE sign-off of seismic calculations prior to shipment of cable tray: "Why isn't the operation monitored closely enough to cover this kind of sloppy activity and are there other items on site that are similarly uncovered by PE sign-off, despite specification requirements?"

Answer
An engineering "Release for Shipment" was given to the vendor prior to shipment of the cable tray. SWEC Procurement Quality Assurance inspectors had placed a "Hold on Shipment" because a professional engineer had not signed off on the seismic calculations from the vendor. The matter was referred to SWEC engineering and evaluated by the cognizant SWEC engineer. In his judgment, the calculations had already been reviewed and were considered adequate by the Stone & Webster Engineering Department. In addition, the vendor advised the cognizant engineer that the professional engineer's signature was forthcoming. The cognizant engineer then revised the specification requirement based on sound engineering judgment.

It is likely that there are other items on site that have similarly received an engineering evaluation prior to issuing a "Release for Shipment." This type of controlled process is good engineering practice and is subject to the controls established in the Quality Assurance program.

Allegation
The alleger disagrees with the NRC staff statement in Inspection Report No. 50-458/80-08 that the use of two different types of cable in the same circuit at River Bend's run to the makeup water structure does not violate good engineering practice. The alleger suggests that the NRC will be hard pressed to find a competent cable engineer to endorse such a practice because, for one thing, in the case in question, the ground braids used on the two cables are made of different materials, which is definitely not a recommended practice. The alleger claims that the main rationale behind

the decision to use two different sizes of cable for the run to the makeup water structure was to save money, and also suggests that it is highly doubtful that this design had the endorsement of the SWEC cable specialist

Answer

As documented in the enclosed IE Inspection Report No. 50-458/80-11 dated November 18, 1980, the use of direct bury cable does not violate good engineering practice. The use of direct bury distribution feeders is a widely accepted engineering practice. The design and cable specification requirements for this installation have been reviewed and endorsed by the responsible SWEC cable specialist. The copper versus bronze shields on the interface between the cable used in the plant and cable duct and the direct bury cable should not cause a problem if normal splicing and grounding techniques are applied as required by specifications and procedures.

Question

How will a cable tray with a rung removed, a permissible configuration for installation, be qualified for seismic considerations? The vendor, Husky Products, can prove the seismic capabilities of their tray with a rung removed either by analysis or by test, but, if they do it by analysis, then that too should be signed off by a PE. Also, if they have seismically proven their straight cable trays, how do we know that the fittings would meet the same criteria? Was a change to the PSAR submitted to NRC?

Answer

The vendor has submitted supplemental seismic calculations for cable tray rung removal to SWEC. These calculations have been reviewed and approved by Stone and Webster, the architect-engineer. All cable tray including straight runs, fixtures and accessories will be seismically qualified

There is no need to change the PSAR since every applicant for a construction permit is required to include in the PSAR a description of the quality assurance program to be applied to the design and construction of the facility. The River Bend Quality Assurance Program, as discussed in Chapter 17 of the PSAR, is required to meet the requirements of Appendix B to 10 CFR 50. Criterion VII of Appendix B requires documentary evidence that material and equipment shall conform to the procurement requirements and shall be available at the nuclear power plant prior to installation or use. Thus, the site specification requirement is in accordance with the PSAR.

Allegation

Any comparison between what is called out in calculation E46H and what is now ordered from Anaconda in the specification and all addenda, as well as tables of reel assignments, will show that E46H is totally obsolete and that the cables have been ordered to conform to E120. E46H was performed several years ago and was correct when it was done, but since that time the circuit length estimates have changed, the loads have changed, and, indeed, even the cable impedance tables have changed. Thus, the SWEC contention that they are using an obsolete calculation to order 5 kV and 15 kV power cable is totally in error.

Answer

The approved calculation E46H was used by the licensee as a basis to initiate the purchase order for cable. It was anticipated that design changes could impact cable requirement; therefore, the licensee developed calculation E120 to address this impact. Since design changes are still taking place, calculation E120 is still not approved. The calculation will be approved when the final design changes are made. What NRC requires is that the applicant perform an analysis of the complete system from the switchyard down to the lowest voltage of the Class IE systems for the worst-case conditions; that is, for the lowest grid voltage and the highest loading in plant. NRC also requires the applicant to determine by analysis that, given the worst-case condition, including a design basis accident and starting of large motor loads, the power quality at all the Class IE busses is within the normal range.

After that analysis is completed and the plant is in its preoperational stage, then the applicant is required to perform testing to measure the loads and the voltages at all the safety busses, and then enter the load measurements into his computerized analysis to compare the actual measured voltages against the voltages determined by analysis results and verify the validity of his model.

Question

How do they propose to meet the 400,000 Btu/hr flame test requirements, and when?

Answer

There is no known requirement to meet a 400,000 Btu/hr flame test. Neither Regulatory Guide 1.131 nor IEEE Standard 383, 1974 requires anything close to a 400,000 Btu/hr heat rate for flame testing.

Finally, the alleger states that NRC acknowledges eight items of noncompliance in IE Inspection Report No. 50-458/80-08 and yet a statement is made in the report that "no items of noncompliance or deviations have been identifed." The response to this statement is provided in the body of the Decision.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of **Emergency Core Cooling Systems**

PETITION TO SUSPEND ALL OPERATING LICENSES FOR PRESSURIZED WATER REACTORS **May 29, 1981**

The Director of the Office of Nuclear Reactor Regulation denies a request by the Environmental Coalition on Nuclear Power for suspension of the operating licenses for all pressurized water reactors because the accident at TMI-2 demonstrated that the evaluation of the plants' Emergency Core Cooling Systems did not meet the requirements of 10 C.F.R. 50.46.

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

By petition dated March 29, 1979 the Environmental Coalition on Nuclear Power (ECNP) requested that the Nuclear Regulatory Commission (NRC) suspend all operating licenses for pressurized water reactors (PWRs). This petition has been considered under the provisions of 10 C.F.R. 2.206 of the Commission's regulations. Notice of receipt of the petition was published in the *Federal Register* December 6, 1979 (44 FR 70241).

The petition contends that safety evaluations for all operating PWRs are invalid and thus licenses for all PWRs should be suspended or revoked. Petitioner asserts that the consequences of the accident at Three Mile Island Unit 2 (TMI-2), at least some of the fuel melted, was in excess of the performance required for the Emergency Core Cooling System (ECCS) under 10 C.F.R. 50.46. Yet, the petitioner also contends, the accident which initiated the TMI-2 fuel damage was less severe than accidents specifically analyzed to demonstrate acceptable performance by the ECCS. Thus, petitioner contends, the analyses used to predict performance under the provisions of 10 C.F.R. 50.46 must be invalid and hence the basis for granting all PWRs licenses is invalid and these licenses should be suspended or revoked.

I have reviewed the information submitted by the Environmental Coalition on Nuclear Power and the issues addressed in the petition. For the reasons set forth below, petitioner's request that all operating licenses for PWRs should be suspended or revoked is denied.

Section 50.46 of the Commission's regulations requires that each boiling and pressurized light water nuclear power reactor must be provided with an emergency core cooling system designed in such a way that its calculated cooling performance following postulated loss-of-coolant accidents conforms to a set of criteria. Included in that set of criteria [10 C.F.R. 50.46(b)] is a requirement that the calculated maximum fuel element cladding temperature shall not exceed 2200°F. 10 C.F.R. 50.46 further requires that ECCS cooling performance is to be calculated: 1) in accordance with an acceptable evaluation model and 2) for a number of postulated loss-of-coolant accidents sufficient to provide assurance that the entire spectrum of loss-of-coolant accidents is covered. The spectrum of accidents examined incudes a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system. (10 C.F.R. Part 50, Appendix K, I.C.1.)

On March 28, 1979, TMI-2 experienced a feedwater transient that through a particular sequence of failures, led to a small break loss-of-coolant accident and resulted in significant core damage. The failures that were experienced occurred in the general areas of design, equipment malfunction, and human performance. This TMI-2 sequence of events and failures had not been previously analyzed and the fuel damage was beyond that predicted by 10 C.F.R. 50.46 analyses. Therefore, a question could be raised as to whether the analyses performed to meet 10 C.F.R. 50.46 were adequate, specifically: 1) whether the evaluation model used for compliance with 50.46 to evaluate the behavior of the reactor system during a postulated loss-of-coolant accident was adequate; and 2) whether there is sufficient assurance that a proper set of loss-of-coolant accidents has been analyzed to determine that the ECCS will perform as required.

In the NRC's Office of Inspection and Enforcement investigation of the TMI-2 accident (NUREG-0600, "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement"), it was stated that the TMI-2 accident could have been prevented in spite of any known or postulated inadequacies in transient and accident analyses. The forward to NUREG-0600 states:

"The design of the plant, the equipment that was installed, the various accident and transient analyses, and the emergency procedures were adequate to have prevented the serious consequences of the accident, if they had been permitted to function or be carried out as planned. For example, had the operators allowed the emergency core cooling system to perform its intended function, damage to the core would most likely have been prevented."

NUREG-0600 estimates that during the initial $3\frac{1}{2}$ hours of the accident the average ECCS flow was only about 25 gpm, because the operators had reduced the flow. As part of the TMI Inquiry, Battelle Columbus Laboratories explored alternative accident sequences. (NUREG/CR-1219, "Analysis of the Three Mile Island Accident and Alternative Sequences"). They concluded that if the high pressure injection (HPI) ECCS flow had not been throttled, full ECCS flow through the pumps would have remained above 800 gpm. As a result, the core would have remained covered, the fuel cladding temperature would not have increased at all, and the requirements of 10 CFR 50.46 would not have been violated.

Babcock and Wilcox has analyzed small break accidents similar in size to the TMI stuck open PORV. For these analyses ("Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant", May 7, 1979) B&W used methods which comply with the requirements of 50.46 and Appendix K to 10 CFR 50. The required single failure for these small breaks would mean that one of the HPI pumps did not work. The B&W analyses showed that the core remained covered for these small breaks even with half the total possible HPI flow. Both the BMI and B&W analyses were benchmarked successfully against the reduced HPI flow data from the TMI accident.

Clearly the TMI ECCS system was designed to cope with a TMI type accident, and would have, had the system not been overridden by the operators. 10 CFR 50.46 and Appendix K require that the ECCS be capable of mitigating the effects of an accident, assuming the most limiting single failure. However, it is recognized that the occurrence of multiple equipment failures and/or operator errors could result in conditions which exceed the core thermal limits of Appendix K. Such was the case at TMI. Contrary to the petitioners contention, the events which occurred at TMI removed the plant from its design envelope, and placed it in a more severe condition than that required to be analyzed by Appendix K and 50.46. It is not reasonable to require protection from the effects of every conceivable combination of errors which could occur, without limiting the number of errors, because the number of such combinations is limitless.

One of the tasks of the NRR Bulletins and Order Task Force (B&OTF) formed in May 1979 was to generically evaluate feedwater transients, small breaks LOCAs, and other TMI-2 related events in operating plants to confirm or establish the basis for their continued safe operation. In order to fulfill this charter, B&OTF investigated a large spectrum of small breaks and transients to assure that the installed system for all modern operating light water reactors could adequately cope with these events. Reactor vendors, NRC consultants, and the NRC staff were required to analyze hundreds of cases in pursuit of this goal. As a result of this review, some parts of the analytical models were targeted for future review and possible improvement. However none of these sub-models were judged to have a substantial impact on ECCS system design. The analysis also aided in assessing operator guidelines for recognition and mitigation of small break LOCA (see NUREG-0645, "Report of the Bulletins and Orders Task Force" January 1980). Another major charter of B&OTF was to assure that all licensees were well trained in the recognition and mitigation of small break LOCAs and would not prematurely throttle or terminate the ECCS during such an event.

To implement the recommendations of all the internal and external TMI inquiries, a plan of action was devised (NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, published May 1980 and revised August 1980) in the form of a set of findings and requirements for safe operation of all reactors. These findings and requirements have been further elaborated in NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980. Included in this action plan is the small break model re-assessment. This activity has already begun and includes the comparison of affected analytical models to a large variety of experiments. To date the ability of these analysis tools has been encouraging. I see no evidence that they are not up to the task. The NRR staff constantly encourages small break evaluation model holders to make improvements.

In view of the above actions taken since the TMI-2 accident, I find no basis to conclude either from the assertions in the subject petition or from our current knowledge of loss-of-coolant accident analysis methods, that the analyses performed in compliance with 10 C.F.R. 50.46 are not valid.

In addition, as part of the TMI-Action Plan Requirements, a program to evaluate the uncertainties which may exist in small-break ECCS performance calculations has been proposed. Holders of approved ECCS evaluation models will evaluate these uncertainties; the Office of Nuclear Reactor Regulation will evaluate their results. If changes are needed in the analysis methods to properly account for these uncertainties, recommendations will be made to the Commission to adopt such changes. (See NUREG-0660, Task II.E.2)

On the basis of my conclusion that the analyses performed in compliance with 10 C.F.R. 50.46 are valid and in view of the many changes which have been imposed on PWRs, I find that continued operation of PWR's poses no undue risk to the public health and safety.

CONCLUSION

Based on the foregoing discussion and the provisions of 10 C.F.R. 2.206, I have determined that there are no adequate bases for suspension of PWR operating licenses. The request by the Environmental Coalition on Nuclear Power is, therefore, *denied.*

A copy of this decision will be placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C. 20555 and each local public document room for all PWRs. A copy of this decision will also be filed with the Secretary for the Commission's review in accordance with 10 C.F.R. 2.206(c), of the Commission's regulations.

As provided in 10 C.F.R. 2.206(c), this decision will constitute the final action of the Commission twenty-five (25) days after the date of issuance, unless the Commission, on its own motion, institutes a review of this decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 29th day of May, 1981

Cite as 13 NRC 773 (1981) CLI-81-10

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter Bradford
John F. Ahearne

In the Matter of

FEDERAL TORT CLAIM OF GENERAL PUBLIC UTILITIES CORP., *et al.* **June 8, 1981**

The Commission denies a claim submitted by the owners of Three Mile Island, Unit 2 under the Federal Tort Claims Act to recover property damages assertedly sustained as a result of the accident at Three Mile Island.

ATOMIC ENERGY ACT: ROLE OF NRC

Under the regulatory framework derived from the Atomic Energy Act, it is the regulated industry (*i.e.,* the licensees and their suppliers and consultants) that bears the primary responsibility for the proper construction and safe operation of licensed nuclear facilities. The NRC has the statutory responsibility for prescribing licensing standards to protect public health and safety and for inspecting industry's activities against these standards; it does not thereby certify to the industry that the industry's design and procedures are adequate to protect its equipment or operations.

MEMORANDUM AND ORDER

On December 8, 1980, the licensees authorized to operate Three Mile Island Unit No. 2 and their parent company submitted an administrative claim to NRC under the Federal Tort Claims Act (28 U.S.C. § 2671 *et seq.*)

to recover $4,010,000,000.00 in property damages which they assert they have sustained as a result of the March 28, 1979 accident at TMI-2.[1] The claimants are the General Public Utilities Corporation ("GPU") and its operating subsidiaries, Jersey Central Power & Light Company ("JCP&L"), Metropolitan Edison Company ("Met-Ed"), and Pennsylvania Electric Company ("Penelec"). The operating subsidiaries are co-owners and co-licensees of TMI-2. Met-Ed is the operator of TMI-2.

The claimants assert that NRC was negligent in the performance of its regulatory duties respecting TMI-2 and that such negligence was a proximate cause of the accident. More particularly, they claim that NRC failed to review with due care certain equipment, analyses, procedures, and training before licensing TMI-2 on February 8, 1978, and failed to warn them of defects affecting TMI-2 of which NRC was, or should have been, aware.

1. The GPU claim rests on two general assertions. First, the claim asserts that NRC negligently failed to warn GPU or Met-Ed of defects in the equipment, analyses, procedures, and training supplied for TMI-2 and negligently failed to direct Met-Ed to implement new operating requirements to correct these deficiencies. Claimants contend that NRC maintains a comprehensive system to collect, analyze, and disseminate data derived from the operating experience of all nuclear reactors in the United States. They claim that they relied on NRC to warn them of any adverse condition that might require corrective action at TMI-2. They contend that NRC failed to fulfill its obligation by negligently failing to investigate, analyze, and warn them of the "Davis-Besse Incident," an "accident that closely paralleled the events which occurred 18 months later at TMI-2."

On September 24, 1977, while operating at less than 10% of full power, the Davis-Besse I nuclear plant experienced a loss of feedwater and turbine trip. Claimants assert that the sequence of events that followed was a precursor to TMI-2: The pilot-operated relief valve ("PORV") on the pressurizer automatically opened and subsequently failed to close, leading to a loss of reactor coolant; high-pressure injection ("HPI") of new coolant activated automatically, but was terminated by operators who, unaware of the open PORV, secured HPI based on pressurizer water level indications

[1]An account of the accident's events and consequences can be found in any of the several major investigations of it. See, for example, *Three Mile Island, A Report to the Commissioners and to the Public*, January 1980; *Report of the President's Commission on the Accident at Three Mile Island*, October 1979; *Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement* (Investigative Rept No. 50-320/79-10), August 1979; *Nuclear Accident and Recovery at Three Mile Island: A Report Prepared by the Subcommittee on Nuclear Regulation for the Committee on Environment and Public Works of the U.S. Senate*, June 1980; *TMI-2 Lessons Learned Task Force Final Report*, August 1979; *TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations*, July 1979.

alone. Davis-Besse officials discovered the open PORV approximately 22 minutes into the incident and immediately shut the PORV block valve. Following other actions including the manual restarting of HPI, the plant resumed a stable condition without damage to the reactor.[2]

Claimants maintain that, as a result of the Davis-Besse incident, NRC knew or should have known of defects in (i) equipment application and instrumentation relating to the PORV, (ii) analyses of potential small coolant breaks and openings at the top of the pressurizer, (iii) procedures and training for plant operators, and (iv) operating and emergency procedures regarding the HPI system. The failure of NRC to notify Met-Ed adequately of these "generic problems" was, they claim, a proximate cause of the accident at TMI-2.

The second general assertion of the claim is that NRC negligently performed its regulatory review of equipment, analyses, procedures, and training supplied for TMI-2 when it licensed the plant's operation. Claimants contend that they relied on NRC to perform with due care the regulatory review required by statute of the safety and safeguards of all facilities, materials, and activities associated with nuclear power plant construction and operation. They argue that NRC negligently reviewed and approved (i) transient analyses relating to small-break loss-of-coolant accidents ("LOCA") and loss of normal feedwater which were inadequate as a basis for plant design and for development of operating procedures and operator training programs, (ii) procedures for operating TMI-2 which were later used by operators during the accident and which incorrectly proscribed filling the pressurizer "solid" with water and risked uncovering the core during a small-break LOCA, (iii) equipment, analyses, and procedures which relied on repeated, correct operation of the PORV which NRC knew, or should have known, incurred prior failures, and (iv) the licensing of operators who were not properly trained to respond to the events that occurred at TMI-2 on March 28, 1979.

2. The claim is without merit. The claim is at odds with the regulatory framework flowing from the Atomic Energy Act of 1954, as amended. Within that framework, the regulated industry (*i.e.,* the licensees and their suppliers and consultants) bears the primary responsibility for the proper construction and safe operation of licensed nuclear facilities. The Nuclear Regulatory Commission has the statutory responsibility for prescribing licensing standards to protect public health and safety and for inspecting

[2]At TMI-2, the PORV was stuck in the open position for more than two hours. Med-Ed officials failed to realize that the valve had not shut. Reactor operators turned off one HPI pump and reduced the flow from a second pump early in the accident sequence. HPI was not restored until almost an hour after the PORV block valve was closed. Substantial damage was done to the reactor.

industry's activities against these standards. The Commission does not thereby certify to the industry that the industry's designs and procedures are adequate to protect its equipment or operations.

This is the understanding that prevailed when NRC issued the license to operate TMI-2, as it had for more than 20 years of commercial nuclear plant licensing and as it continues to prevail today. Therefore the claim is denied.

It is so ORDERED.

Commissioner Ahearne's additional views are attached.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 8th day of June, 1981.

Commissioner Ahearne's Additional Views

I concur in the result reached by the Commission. However, I find the description of our reasons unfortunately brief. We rejected the claim because it is inconsistent with the NRC regulatory philosophy as well as the law.

Within the regulatory framework flowing from the Atomic Energy Act and other applicable statutes, the regulated industry (i.e., the licensee, the vendor, and the architect-engineer) bears the primary responsibility for protecting the general public from the health, safety, and environmental risks posed by the generation of electricity from nuclear power. The industry must take the initiative to develop safe nuclear plants, to monitor them for sufficiency, and to evaluate the need for change. It is best equipped with the resources and detailed knowledge of particular equipment, systems, and procedures to accomplish this task. The Federal government cannot invest enough resources into the review, inspection, and operation of each nuclear power plant to develop the level of knowledge of individual plants possessed by the licensees.

The Nuclear Regulatory Commission has a statutory responsibility for prescribing the minimum standards for assuring the adequate protection of public health and safety. Through licensing and inspection, the Commission's function is to ensure that the industry meets these threshold standards. However, NRC's approval of a licensee as meeting these requirements at one time does not absolve the industry of its independent obligation to operate its equipment in a manner to protect the public. NRC licensing and inspection reviews cannot be and are not intended to be all-encompassing. As is well known to NRC licensees, NRC programs are based on a sampling and do not supplant reviews by the regulated sector. When violations of regulations occur, the NRC imposes penalties. But this is after the violation has occurred and been found.[1] However, the Commission expects nuclear power plant licensees, and the suppliers and architect-engineers with whom they contract, through their own comprehensive reviews to assure or verify independently the adequacy of a plant's design, construction, and operation, and to monitor data respecting the plant's operation to detect the need for corrective measures.

Chairman Hendrie agrees with these views.

[1]It may be noted that compliance with NRC requirements could have prevented the accident's serious consequences. Following the review of the accident, Metropolitan Edison was cited for and chose not to contest violations of NRC requirements. In particular, Metropolitan Edison operating personnel had become accustomed to a leaking pilot-operated relief valve prior to the accident. During the accident this led them to disbelieve indications that the pilot-operated relief valve was stuck open and a loss of coolant accident was in progress.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:
Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **45 Fed. Reg. 76602 (November 19, 1980)**

FIRE PROTECTION FOR OPERATING NUCLEAR POWER PLANTS (10 CFR 50.48) **June 12, 1981**

The Commission denies a motion seeking, *inter alia*, a stay of the effectiveness of various items included in the Commission's Final Rule on Fire Protection for Operating Nuclear Power Plants (45 Fed. Reg. 76602 (November 19, 1980)).

REGULATORY GUIDES: STATUS

Rather than serving as inflexible, legal requirements that must be followed by licensees, regulatory guides (and branch technical positions) are meant to give guidance to licensees concerning those methods the staff finds acceptable for implementing the general criteria embodied in the NRC's rules.

RULES OF PRACTICE: STAY REQUESTS

In deciding whether to grant a stay pending judicial review of an agency action, the Commission considers four factors: (a) whether movants are likely to prevail on the merits; (b) whether movants will be irreparably injured without a stay; (c) whether other interested parties would be harmed by a stay; and (d) whether the public interest supports a stay.

Virginia Petroleum Jobbers Association v. *FPC*, 259 F.2d 921, 925 (D.C. Cir. 1950).

RULEMAKING: RECORD OF PROCEEDING

The record in an informal, notice and comment rulemaking that must be compiled and filed for the purpose of judicial review differs from the record compiled in a formal, "on the record" rulemaking with its more elaborate procedural requirements; the record in an informal rulemaking consists of copies of the notice of rulemaking, any comments on the proposed rule, the final rule supported by a general statement of basis and purpose, 5 U.S.C. 553(b)-(c), and those background materials that indicate adequate support for the rule's factual premises. *Portland Cement Association* v. *Ruckelshaus*, 486 F.2d 375, 393 (D.C. Cir. 1973), *cert. denied*, 417 U.S. 921 (1974).

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

It is well established that an agency's decision to use rulemaking or adjudication in dealing with a problem is a matter of discretion. *NAACP* v. *FPC*, 425 U.S. 662, 668 (1976); *NLRB* v. *Bell Aerospace Corp.*, 416 U.S. 267, 294 (1974); *SEC* v. *Chenery Corp.*, 332 U.S. 194, 201-03 (1947).

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

Because there is no bright line test to assess an agency's exercise of discretion in determining whether to proceed with a rulemaking or an adjudication, courts have focused on the nature of the decision to be reached. Rulemaking is usually prospective in scope and non-accusatory in form, directed to the implementation of general policy concerns through legal standards; adjudication is individual in impact and often condemnatory in purpose, directed to the determination of the legal status of particular persons or practices through the application of preexisting legal standards. *FTC* v. *Brigadier Industries Corp.*, 613 F.2d 1110 (D.C. Cir. 1979).

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

Where a rulemaking proceeding addresses generic concerns applicable to all nuclear plants, the fact that the standards adopted may actually affect a few licensees (or even only one licensee) does not make the agency's utilization of rulemaking improper. *Hercules Inc.* v. *EPA*, 598 F.2d 91, 118

(D.C. Cir. 1978); *Southern Terminal Corp.* v. *EPA,* 504 F.2d 646, 661 & n. 13 (1st Cir. 1974).

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

It is within the purview of the agency's authority to settle factual issues of a generic nature by means of rulemaking. *Minnesota* v. *NRC,* 602 F.2d 412, 416-17 (D.C. Cir. 1979); *Ethyl Corp.* v. *EPA,* 541 F.2d 1, 28-29 & n. 58 (D.C. Cir.) (*en banc*), *cert. denied,* 420 U.S. 941 (1976); *Ecology Action* v. *AEC,* 492 F.2d 998, 1002 (2d Cir. 1974).

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

The NRC has the authority to modify operating licenses by rule; plant-by-plant adjudicatory hearings are not necessary. Atomic Energy Act, Section 187, 42 U.S.C. § 2237.

NRC: CHOICE OF RULEMAKING OR ADJUDICATION

An agency's previous initiation of a case-by-case method of resolving a problem cannot be raised as a bar to any later efforts to resolve generic issues by rulemaking. *Pacific Coast European Conference* v. *United States,* 350 F.2d 197, 205-06 (9th Cir.), *cert. denied,* 382 U.S. 958 (1965).

REGULATIONS: RETROACTIVITY

In determining whether a rule or regulation may be properly made retroactive, the standard to be applied is reasonableness under the circumstances. Relevant factors include: the rule's degree of retroactivity as measured by whether it is an abrupt departure from established practice or an attempt to resolve unsettled questions; the complaining party's reliance upon the agency's former policies; the burden imposed by the retroactive rule; and the need for administrative flexibility in light of changing circumstances. *New York Telephone Co.* v. *FCC,* 631 F.2d 1059, 1068 (2d. Cir. 1980); *Tennessee Gas Pipeline Co.* v. *FERC,* 606 F.2d 1094, 1116 n. 77 (D.C. Cir. 1979), *cert. denied,* 445 U.S. 920 (1980).

REGULATIONS: RETROACTIVITY

The NRC has the authority to apply new standards to already licensed plants, consistent with evolving concepts of what measures are necessary to protect the public health and safety. Atomic Energy Act, Section 186(a), 42

U.S.C. § 2236; *Ft. Pierce Utilities Authority* v. *United States*, 606 F.2d 986, 996 (D.C. Cir.), *cert. denied*, 444 U.S. 842 (1979).

TECHNICAL ISSUES DISCUSSED:

Electrical fires;
Fire barriers;
Safe shutdown capability;
Safe shutdown redundancy and alternatives;
Reactor coolant pump lubricating oil collection system;
Lubricating oil fire hazard;
Fire suppression system.

MEMORANDUM AND ORDER

Several licensees[1] have jointly moved the Commission to stay three items in the Final Rule on Fire Protection for Operating Nuclear Power Plants, pending judicial review. 45 Fed. Reg. 76602 (November 19, 1980). The items are: Item G (Fire Protection for Safe Shutdown Capability); Item L (Alternative and Dedicated Shutdown Capability); and Item O (Oil Collection System for Reactor Coolant Pump) of Appendix R. III to 10 CFR Part 50. These licensees have also requested a stay of the effective and compliance dates for those items. Subsequently, these licensees and several others petitioned the Commission for exemptions from schedule requirements in the rule pursuant to 10 CFR 50.12. In addition, some licensees also petitioned for exemptions from various substantive fire protection features required by Appendix R including Items III.G, III.L and III.O. For the reasons discussed below, the motion is denied.

I. Introduction

Before addressing the motion for a stay, it would be useful to review the events leading to the promulgation of 10 CFR 50.48 and Appendix R to Part 50. The NRC's current concern with fire protection was initiated by a fire at the Browns Ferry Nuclear Station in March 1975. The fire damaged over 1600 electrical cables and caused the temporary unavailability of some core cooling systems. Because this fire did substantial damage, the NRC established a Special Review Group which initiated an evaluation of the

[1]The movants are: Northeast Utilities Service Company, the Connecticut Light and Power Company, the Hartford Electric Light Company, Western Massachusetts Electric Company, Northeast Nuclear Energy Company, Boston Edison Company, Florida Power and Light Company, and Arkansas Power and Light Company.

need for improving the fire protection programs at all nuclear power plants. The group found serious design inadequacies regarding fire protection at Browns Ferry, and its report, "Recommendations Related to Browns Ferry Fire" (NUREG-0050, February 1976), contained over fifty recommendations regarding improvements in fire prevention and control in existing facilities. The report also called for the development of specific guidance for implementing fire protection regulations, and for a comparison of that guidance with the fire protection program at each operating plant.

NRC developed technical guidance from the technical recommendations in the Special Group's report, and issued those guidelines as Branch Technical Position Auxiliary Power Conversion Systems Branch 9.5-1 (BTP 9.5-1),[2] "Guidelines for Fire Protection for Nuclear Power Plants." This guidance did not apply to plants operating at that time. Guidance to operating plants was provided later in Appendix A[3] to BTP 9.5-1 which, to the extent practicable, relies on BTP 9.5-1. The guidance in these documents was also published for public comment as Regulatory Guide 1.120, "Fire Protection for Nuclear Power Plants" (June 1976). In response to public comment, the NRC issued an extensively revised version of Regulatory Guide 1.120 for further public comment.

In May 1976, the NRC asked licensees to compare operating reactors with BTP 9.5-1, and in September 1976, those licensees were informed that the guidelines in Appendix A would be used to analyze the consequences of a fire in each plant area. Early in 1977 each licensee responded with a Fire Protection Program Evaluation which included a Fire Hazard Analysis. These evaluations and analyses identified aspects of licensees' fire protection programs that did not conform to the NRC guidelines. Thereafter, the staff initiated discussions with all licensees aimed at achieving implementation of fire protection guidelines by October 1980. The staff held many meetings with licensees, conducted extensive correspondence with them, and visited every operating reactor. As a result, many fire protection items were resolved, and agreements were memorialized in Fire Protection Safety Evaluation Reports issued by the NRC. Other fire protection issues remained unresolved.

By early 1980, most operating plants had implemented most of the guidelines in Appendix A. However, as the Commission noted in its Order

[2]Rather than serving as inflexible, legal requirements that must be followed by licensees, issuances such as regulatory guides and branch technical positions are meant to give guidance to licensees concerning those methods the staff finds acceptable for implementing the general criteria embodied in the NRC's rules. *See, e.g., Petition for Emergency & Remedial Action*, CLI-78-6, 7 NRC 400, 406 (1978); *Gulf States Utilities Company* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760, 772 (1977).

[3]Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976.

of May 23, 1980, the fire protection program has had some significant problems with implementation.[4] Despite the staff's efforts, several licensees had expressed continuing disagreement with, and refused to adopt, recommendations relating to several generic issues, including the requirements for fire brigade size and training, water supplies for fire suppression systems, alternate and dedicated shutdown capability, emergency lighting, the qualifications of seals used to enclose places where cables penetrated fire barriers, and the prevention of reactor coolant pump lubrication system fires. To establish a definitive resolution of these contested subjects in a manner consistent with the general guidelines in Appendix A and to assure timely compliance by licensees, the Commission issued a proposed fire protection rule and its Appendix R, which was described as setting out minimum fire protection requirements. 45 Fed. Reg. 36082 (May 29, 1980). The fire protection features addressed included protection of safe shutdown capability, emergency lighting, fire barriers, associated circuits, reactor coolant pump lubrication system, and alternate shutdown systems. The Commission stated that it expected all modifications (except for alternate and dedicated shutdown capability) to be implemented by November 1, 1980.[5]

As originally proposed, Appendix R would have applied to all plants including those for which the staff had previously accepted other fire protection modifications. After analyzing comments on the rule, the Commission determined that only three of the fifteen items in Appendix R were of such safety significance that they should apply to all plants, including those for which alternative fire protection actions had been approved previously by the staff. These items are protection of safe shutdown capability (including alternate shutdown systems), emergency lighting, and the reactor coolant pump lubrication system. Accordingly, the final rule requires all reactors licensed to operate before January 1, 1979, to comply with these three items even if the NRC had previously approved alternative fire protection features in these areas. 45 Fed. Reg. 76602 (Nov. 19, 1980). However, the final rule is more flexible than the proposed rule because Item III.G now provides three alternative fire protection features which do not require analysis to demonstrate the protection of redundant safe shutdown equipment, and reduces the acceptable distance in the physical separation alternative from fifty feet to twenty feet. In addition, the rule now also provides an exemption procedure which can be initiated by a licensee's assertion that any required fire protection feature will not enhance fire protection safety in the facility or that such modifications may

[4] 11 NRC 707, 718 (1980).
[5] *Id.* at 719.

be detrimental to overall facility safety. 10 CFR 50.48(c)(6). If the Director, Nuclear Reactor Regulation determines that a licensee has made a *prima facie* showing of a sound technical basis for such an assertion, then the implementation dates of the rule are tolled until final Commission action on the exemption request.

Movants have requested a stay of Item III.G, Protection of Safe Shutdown Capability, the related Item III.L, Alternate Shutdown System, and Item III.O, Reactor Coolant Pump Lubrication System.

II. Stay Request

In deciding whether to grant a stay pending judicial review, the Commission considers four factors: (A) whether movants are likely to prevail on the merits; (B) whether movants will be irreparably injured without a stay; (C) whether other interested parties would be harmed by a stay; and (D) whether the public interest supports a stay. *Virginia Petroleum Jobbers Association* v. *FPC*, 259 F.2d 921, 925 (D.C. Cir. 1950); *Washington Metropolitan Area Transit Commission* v. *Holiday Tours, Inc.*, 559 F.2d 841, 843 (D.C. Cir. 1977); see also 10 CFR 2.788. For the reasons discussed below, we find that movants have utterly failed to sustain their burden. Accordingly, their motion is denied.

A. Probability of Success on the Merits

Movants contend that the rule is defective because it is not supported by an adequate statement of basis and purpose or by an adequate record. They also assert that there were a number of procedural flaws in the process used in the promulgation of the rule. They allege that the fire protection rule is an unlawful attempt to avoid adjudication, and contend that plant-by-plant adjudications were required because there are material factual disputes regarding each affected facility. Movants also believe that the fire protection rule is really a license amendment and, thus, believe they are entitled to a hearing under Section 189 of the Atomic Energy Act of 1954 42 U.S.C. § 2239, and 10 CFR 2.204. Movants contend that the NRC abused its discretion by undertaking the rulemaking when the agency previously had sought to deal with fire protection problems as they concerned each individual nuclear facility. Movants also challenge the agency's decision to "backfit" Items III.G, III.L, and III.O, for alleged failure to comply with the terms of 10 CFR § 50.109(a) and as an impermissible retroactive rulemaking.

1. Basis and Purpose

Movants challenge the adequacy of the record supporting Items III.G, III.L, and III.O. For the past several years, the NRC staff and licensees for all operating reactors have engaged in extensive discussions and exchanges of correspondence regarding the many aspects of fire protection, including Items III.G, III.L, and III.O. These interactions have led to the creation of a substantial volume of publicly available material containing the factual premises and policy considerations underlying the various portions of this fire protection rule.[6] These documents must be considered part of the record for this rulemaking and they provide a more than adequate basis to support the final rule.[7] In particular, a review of the record shows that the movants are unlikely to succeed on the merits of their arguments regarding Items III.G, III.L and Item III.O. The specific details of the record for these Items are discussed in order below.

a. III.G. Protection of Safe Shutdown Capability

All reactors are designed to include redundant independent equipment in safety systems so that the failure of any component or subsystem will not prevent the function of a safety system. A fundamental purpose of the fire protection requirements is maintenance of the ability of safety systems to shut down a nuclear power plant in the event of failure caused by a fire.[8] Item III.G of the final rule is intended to accomplish this goal by providing licensees a choice from among three specific fire protection features that would ensure that redundant safety equipment would be protected from damage by a single fire in a fire area. These features are: (1) separation by a fire barrier having a three-hour rating; (2) enclosure of one redundant train by a fire barrier having a one-hour fire rating in combination with

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[6]Additionally, congressional hearings on the Browns Ferry fire contain information about the fire protection issues which was available to the Commission when it adopted the rule. *See* Browns Ferry Nuclear Plant Fire: Hearings Before the Joint Committee on Atomic Energy, 94th Cong., 1st Sess. (1975).

[7]Although Federal Rules of Appellate Procedure 16 and 17 speak of a "record" that must be compiled and filed for the purpose of judicial review, this record for informal, notice and comment rulemaking is to be distinguished from the "record" compiled in a formal, "on the record" rulemaking with its more elaborate procedural requirements. *See National Nutritional Foods Ass'n* v. *Weinberger,* 512 F.2d 688, 701 (2d Cir.), *cert. denied,* 423 U.S. 827 (1975). The record in informal rulemaking consists of copies of the notice of rulemaking, any comments on the proposed rule, the final rule supported by a general statement of basis and purpose, 5 U.S.C. 553(b)-(c), and those background materials that indicate adequate support for the rule's factual premises, *see Portland Cement Ass'n* v. *Ruckelshaus,* 486 F.2d 375, 393 (D.C. Cir. 1973), *cert. denied,* 417 U.S. 921 (1974). *See generally* 1 K. Davis *Administrative Law Treatise,* §§ 6:4, :10, :13 (2d ed. 1978).

[8]BTP 9.5-1, App. A., pg. 2, A.2 (1976).

automatic fire detection and suppression systems; or (3) physical separation of redundant systems by twenty feet of space free of intervening combustible material and automatic fire detection and suppression systems. If none of these features can be achieved in a specific fire area, then an alternative or dedicated shutdown system should be installed. Licensees could also request an exemption from these requirements based on a fire hazards analysis for a specific plant.

Electrical cables which carry power to this equipment must also be redundant and independent to preserve the function of safety systems. Thus, for example, cables which connect to one set of safety equipment must be separated from the cables which connect to the redundant safety equipment so that the same event will not simultaneously disable both sets of cables and, thus, their connected safety equipment. Most cables are grouped together in metal cable trays; and trays containing cables connected to redundant safety equipment are usually physically separated from each other. Nevertheless, the fire at Browns Ferry showed that some redundant cables were not adequately separated and could be simultaneously disabled by the same fire. For example, cables connected to one division of safety equipment were close to other cables connected to the redundant safety equipment. Even though the cables from one division were in electric conduit (lightweight metal pipes) and the cables from the other division were in metal trays, the burning cables in the trays caused the cable in the conduit to fail.[9]

Moreover, redundant systems were found not to be independent as previously believed because non-safety cables connected to safety cables were not recognized as being capable of causing the failure of safety equipment. At Browns Ferry, these non-safety circuits, which are also called "associated circuits," were not separated into divisions. As a result, the fire simultaneously damaged both sets of non-safety cables and caused the failure of redundant safety equipment.[10]

As a result of the experience at Browns Ferry, and in accordance with national and international publications on fire protection at nuclear power plants[11] the NRC, beginning in 1976, established a guideline that required

[9]NUREG-0050 at 36.

[10]For example, redundant control circuits providing power to operate valves necessary for safe shutdown were simultaneously disabled by fire damage to associated non-safety circuits which connected the safety circuits to indicator lights in the control room. NUREG-0050 at 35-36.

[11]Among the documents considered by the NRC are "The International Guidelines for the Fire Protection of Nuclear Power Plants," 1974 Edition, 2nd Reprint, published on behalf of the National Nuclear Risks Insurance Pools and Association and "Specifications for Fire Protection of New Plants" published by Nuclear Energy Liability and Property Insurance Association; *see* BTP 9.5.1-16.

redundant safety related cable divisions in new plants should be separated by fire barriers having a minimum fire resistance rating of three hours.[12] Three-hour barriers were also established as separations between many other safety related systems.[13] These guidelines are consistent with the well-recognized principle that walls having a three-hour fire rating should be used in significant commercial, residential, and industrial buildings wherever essential structural features must be protected from fire damage.[14]

For plants already built or operating, staff recognized that the separation of redundant safety systems by three-hour fire barriers might not always be possible. Accordingly, the guidelines in Appendix A provided that equivalent protection could be provided by a combination of alternative fire protection features including physical separation, fire detection and suppression systems, fire barriers or enclosures having less than a three-hour fire rating, and fire retardant coatings.[15] If a suitable combination of features could not be achieved in a plant already built and operating, an alternative or dedicated shutdown system was to be considered.[16] The guidelines also addressed the treatment of associated circuits. Associated circuits were either to be treated as safety circuits regarding separation and independence or were to be adequately isolated from safety circuits.[17]

These guidelines were used by each licensee to prepare a Fire Hazard Analysis which evaluated the potential effects of a fire at each operating utility. The NRC staff reviewed each Fire Hazard Analysis, visited each plant, and prepared a Fire Safety Evaluation Report (Report) for each plant. Each report described the various fire protection features which would implement the guidelines.[18] All operating plants were found to require modifications to protect redundant safe shutdown equipment.[19]

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[12]BTP 9.5-1, pp 9.5.1-24 and 9.5.1-52 (1976).

[13]*See,* for example, BTP 9.5-1, pg. 9.5.1-54 which requires three-hour barriers between redundant switch gear safety divisions, battery rooms, and each battery room and the rest of the plant.

[14]*See,* for example, the "Fire Protection Handbook," Fourteenth Edition, National Fire Protection Association, 6-37 (1976).

[15]*See,* for example, BTP 9.5-1, App. A at pg. 12, D.1(a)(2) and pg. 15, D.1(j) (1976).

[16]BTP 9.5-1, App. A at pp. 2, 12, 39 (1976).

[17]*See,* for example, Reg. Guide 1.75, Rev. 2, pp. 1.75-2 and 1.75-3 (1978).

[18]*See,* for example, the NRC Fire Protection Evaluation of February 22, 1980 for Haddam Neck, a facility operated by one of the movants. Staff informed the licensee that an alternate shutdown system would be required because the inadequate separation of electrical cables and equipment was not adequately compensated for by flame retardant coatings and fire suppression systems. Pp. 4-1 and 4-9.

[19]*See,* for example, the Fire Protection Safety Evaluation Report for Pilgrim Nuclear Station, Unit No. 1, in which the staff found that additional analysis of cable separation was required because neither the existing separation between cables nor the asbestos sheets interposed between cables provided adequate fire barriers. P. 4-13 (1978). Inadequate separation of redundant cable trays was also identified in several other plants including Millstone Nuclear Power Station, Unit No. 2, Fire Protection Safety Evaluation Report at pp. 4-7 to 4-8 (1978).

Subsequently, several licensees installed the fire protection features de-scribed in the Reports, including the rerouting of associated circuits or installation of relays to isolate associated circuits from safety circuits.[20] Others installed alternate or dedicated safe shutdown systems.[21] However, some licensees questioned whether all the modifications proposed by the staff to protect safe shutdown systems were needed; and in spite of extensive explanations by the staff, these licensees refused to voluntarily implement all of those modifications. Because the staff believed that implementation of those fire protection features was necessary to assure that a single fire would not disable redundant safe shutdown systems, the staff included proposed Item II.A.2.f, III.G, III.M, II.E, III.Q, and II.A.3 in the proposed rule issued on May 29, 1980. 45 Fed. Reg. 36082.

Item II.A.2.f provided that fire retardant, heat shields, or local fire barriers shall be provided between redundant safe shutdown systems and components or between such systems where their physical separation is inadequate to ensure that automatic and manual fire suppression can limit fire damage to one division of shutdown systems.

Item III.G of the proposed rule provided that the protective features to protect safe shutdown capability could be a combination of several measures already familiar to licensees, i.e., fire barriers, physical separation, automatic and manual fire suppression capability, fire retardant coatings, and alternative shutdown capability. Item III.G also contained a detailed discussion of the combination of fire hazards, susceptibility activities, fire suppression means available, and availability of alternative shutdown capability that would have to be considered in evaluating the effectiveness of fire protection in areas containing safe shutdown equipment.

Item III.M of the proposed rule required that fire barriers separating fire areas, or equipment, or components of redundant systems important to safe shutdown in an area should have a fire rating of three hours unless a lower rating is justified by a fire hazard analysis. As discussed above, such three-hour barriers are a well-recognized fire protection feature.[22] Moreover, in 1977, the staff explained its choice of a three-hour barrier in a publicly available memorandum to the Advisory Committee on Reactor Safeguards Enclosure A. Staff gave five reasons for choosing three-hour barriers. First there is no precise quantitative relation between the amount of combustible material in a fire area (the fire load) and the rating of a fire barrier that wil

[20]*See*, for example, Fire Protection Safety Evaluation Report for Arkansas Nuclear One, Uni 1, pg. 3-3 (1978).

[21]*See*, for example, Memorandum from Edson Case to Commission which reported that five plants had installed alternate or dedicated shutdown systems and that twelve more plants required such modification (July 6, 1979).

[22]*See* Regulatory Guide 1.120, Sections C.4.a.(1) and C.4.c.(2) (1977).

not be breached by a fire in that area. Thus, an assessment of the fire load in a fire area will provide only a general estimate of the appropriate fire rating for a barrier that would isolate a fire in that area. Second, a three-hour barrier provides the plant fire brigade an extra margin of time in fighting a fire. The extra time is especially important for nuclear power plants located in remote locations. For such plants, fire fighters in the vicinity would require additional time to respond to a call for help from the plant. Third, a three-hour barrier is the minimum fire related barrier that is constructed entirely of non-combustible material. Fourth, three-hour barriers are commonly used to protect buildings which are "high value risks." Finally, three-hour barriers provide a safety margin to compensate for the uncertainties over the level of transient combustibles which may be introduced into the plant over its 40-year life.

Item II.E of the proposed rule provided that no additional fire hazard analysis would be needed for redundant systems and components separated by either a three-hour fire barrier or fifty feet both horizontal and vertical of clear air space. Lesser fire barriers or separations would require justification.

Item III.Q of the proposed rule required either that associated circuits be isolated from safety circuits, or if they could not be isolated, be treated as part of the safe shutdown circuit with which they were associated. Reference was made to Regulatory Guide 1.75, which introduced the concept of associated circuits in 1974, and IEEE 384-1974.

Item II.A.3 of the proposed rule required the provision of alternative shutdown capability if safe shutdown could not be ensured by a combination of fire protection features including barriers and detection and suppression systems.

Commentors objected to the level of detail in proposed Item III.G. They also objected to the fifty-foot separation criterion in Item II.E; Alabama Power and Light noted that a twenty-foot separation is accepted practice in the nuclear industry and the Edison Electric Institute stated that in the past the NRC considered adequate a twenty-foot separation. Some commentors requested a definition of associated circuits. Several commentors contended that they could not identify all associated circuits and take the required actions in the time period provided by the rule.

In response, much of the detail was deleted from Item III.G as proposed and the five proposed items discussed above were consolidated into Item III.G of the final rule, which provided three acceptable alternatives for ensuring maintenance of the redundant shutdown capability in the event of fire. They are: (1) separation of redundant safety trains by a three-hour fire barrier; or (2) separation of redundant trains by twenty feet free of intervening combustibles coupled with fire detectors and an automatic fire suppression system; or (3) enclosure of one redundant safety train by a one-hour fire barrier coupled with fire detectors and an automatic fire suppression system. Thus, the final rule relaxes some of the requirements in the proposed rule and provides licensees with greater flexibility. Instead of a fifty-foot separation between redundant systems, a twenty-foot separation is now considered adequate without analysis. An additional alternative, enclosure of one redundant safety train by a one-hour fire barrier coupled with fire detectors and an automatic fire suppression system, is available to demonstrate the protection of redundant safe shutdown equipment without further analysis. In addition, associated circuits that can adversely affect redundant trains of safety equipment are to be treated in the same manner as redundant safety circuits. If these features cannot be provided, then an alternate or dedicated independent safe shutdown capability is required.

The statement of supplemental information (Statement) accompanying the final rule explained that the rule specifies design basis protective features instead of a design basis fire because it is not possible to predict the specific conditions under which fires may occur and propagate. That Statement also discussed the bases supporting the well-known choice of a three-hour fire barrier. Such barriers are inherently reliable because they are passive and thus not subject to mechanical failure. Moreover, the choice of three-hour barriers is consistent with the potentially serious consequences of a fire in a nuclear power plant and is in the range of values used for barriers in comparable industrial properties. The Statement goes on to explain that if specific plant conditions preclude the installation of a three-hour barrier, then licensees may rely on alternate methods that provide equivalent fire protection. The example analyzed in the Statement is the combination of a one-hour barrier with an automatic suppression system. The Statement explains that because a fire may not immediately actuate an automatic suppression system, a one-hour barrier is also required to protect the redundant safety train from fire until the suppression system goes on.

Regarding associated circuits, the Statement notes that the definition of associated circuits was provided in Reg. Guide 1.75, and IEEE Standard 384-1974, a commonly used industry standard. Staff also noted that the objections by some commentors clearly indicated that they understood the proposed requirements regarding associated circuits. Since the rule has been issued, the NRC staff has met with some licensees to discuss associated circuits and provided a modified definition of associated circuits in a February 20, 1981 generic letter sent to all power reactor licensees with plants licensed prior to January 1, 1979. Enclosure B. The new definition of associated circuits removes any uncertainty over their identification. Although the new definition includes more types of circuits, it does not appear to materially affect the number of circuits which will be identified as associated circuits because the definition now excludes circuits which are protected by coordinated circuit breakers, fuses, or other such devices. Credit previously given for such devices is uncertain. No licensee has requested an exemption on the basis of inability to identify associated circuits.

In addition to the Statement of Consideration, other publicly available information long known to licensees supports the rule. For example, there were an extensive series of fire protection discussions between the staff and licensees preceding promulgation of this rule. As discussed above, the requirement of a three-hour fire barrier was well-known even before the proposed rule was issued for comment. Similarly, NRC documents have long-recognized the principle of substitution, which permits licensees to use protection features that provide equivalent protection to redundant safe shutdown that cannot be separated by a three-hour barrier.[23]

The Division of Safety Systems, Nuclear Reactor Regulation, for several years has used a twenty-foot separation criterion in reviewing license applications for new plants. The need for such physical separation is also supported by the results of cable fire tests reported by Sandia Laboratories in 1977 and by a draft report prepared by Gage-Babcock and issued on August 8, 1979. The Sandia test results showed that under some circumstances a fire in a cable tray could damage redundant cables in a separate metal tray five feet away. These results were widely distributed, and provided one basis for a "Petition for Emergency and Remedial Relief" filed by the Union of Concerned Scientists on November 4, 1977. Many NRC licensees participated in the Commission's proceeding on the petition. In response to that petition, the Commission noted that the Sandia tests confirmed earlier conclusions based on the review of the Browns Ferry fire that additional fire protection measures were required over and above the cable separation distances previously accepted by the NRC.[24] The Gage-

[23] *See*, for example, Reg. Guide 1.120, pg. 1.120-12 (1977), and the Fire Protection Handbook, Fourteenth Edition, p. 6-37 (1976).

[24] 7 NRC at 420-21 (1978). In reaching this conclusion, the Commission had before it several comments from utilities who criticized as unrealistic certain aspects of the test procedures used in SAND-77-1424. They contended that the exposure fires used could not be considered credible, and that the fire was more severe than could be expected in a nuclear plant. These comments have been available in the NRC's Public Document Room since late 1977. *See*, for example, Response of Commonwealth Edison Company of November 23, 1977; letter of November 23, 1977 from Consumers Power Company; letter of November 25, 1977 from Debevoise and Liberman on behalf of the Duke Power Company; and Comments of the Connecticut Light and Power Company, the Hartford Electric Company, Western Massachusetts Electric Company, Northeast Nuclear Energy Company, and Connecticut Yankee Atomic Power Company, November 27, 1977. The staff responded to these criticisms in a December 15, 1977 memorandum from E. Case, Acting Director, Office of Nuclear Reactor Regulation, to the Commission which is also part of the administrative record on the UCS petition and has been readily available in the Public Document Room.

Babcock report had been available in the NRC Public Document Room for approximately one year before publication of the proposed rule.[25]

Substantial publicly available information also supports the decision in the final rule not to consider fire retardant coatings on electrical cables to be an acceptable alternative to satisfying Item III.G. Although staff has never treated coatings as fire barriers,[26] it has previously accepted coatings as an alternative to the physical separation of redundant cables. However, beginning in 1977, test results from the Sandia Laboratories showed that fire retardant coatings did not protect electrical cables under certain fire conditions.[27] Fire caused coated cables to lose electric integrity (short circuit to cable trays) even if the cables did not burn.[28] Those test results have been broadly disseminated among licensees. Licensees were also informed in November 1978, by IE Circular 78-18 that some fire retardant coatings are consumed by fire. The Union of Concerned Scientists raised that issue in its Petition for Reconsideration of its Petition for Emergency and Remedial Action; and the Commission expressed concern over the efficacy of fire retardant coatings in its Order of May 27, 1980, denying that petition.[29] Finally, the staff informed some licensees, including some of these movants, of the inadequacy of fire retardant coatings before the proposed rule was issued for comment.[30] Thus, several years of test results

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[25]Mr. M. Bender, a fire protection expert and member of the Advisory Committee for Reactor Safeguards also had the report for at least one year before publication of the proposed rule. Transcript of the ACRS meeting of July 9, 1980 at 102. At that meeting, in a discussion with representatives of the NRC staff and reactor operators, Mr. Bender also noted that "For some reasons or other these kinds of reports which the industry ought to be just as interested in as the regulatory staff don't seem to be of any interest to the industry. Why is that?"

[26]*See*, for example, Memorandum from Moore to Vassallo (June 26, 1979).

[27]*See*, for example, "A Preliminary Report on Fire Protection Research Program Fire Retardant Coatings Test" in which it was concluded that although fire retardant coatings offer some measure of fire protection, there is a wide range in the relative effectiveness of various coatings. SAND-78-0518 (1978).

[28]SAND-78-1456 (1978).

[29]CLI-80-21, 11 NRC at 717-18 (1980).

[30]*See*, for example, Staff Positions on the Turkey Point Fire Protection Program (April 3, 1980). The staff states:

> The fire retardant coating used at Turkey Point has also been tested at Sandia Laboratories. The Sandia tests subjected coated cables to a larger exposure fire than that used in the licensee tests. The cable tray directly exposed to the fire in the Sandia tests suffered considerable damage and burned for approximately 42 minutes. Coated cables in a tray 10-½ inches above the exposed tray were also damaged. Although the cables in the upper tray did not burn, propagation of flames to the upper tray would have occurred if a larger exposure fire had been used, if the cables had been energized at rated current or if the coating had been applied and the cables arranged to simulate more closely a field installation.
>
> It is therefore concluded that the combination of fire retardant coating and the minimum separation permitted between redundant divisions of cables at Turkey

have created a clear basis for doubting the efficacy of fire retardant coatings. The NRC's response to those results has clearly been developing in the direction of reduced reliance on fire retardant coatings, and the uncertainties regarding fire hazards led the Commission to finally endorse the position taken in this rule. Under these circumstances, we find no merit in any contention regarding the adequacy of the basis for that part of the rule that gives no credit to fire protection retardant coatings as a fire barrier.

This review of the record on the fire protection features in Item III.G.2 clearly shows that the requirement of a three-hour barrier to protect redundant safe shutdown systems has long been accepted as appropriate for protection against the potential hazard posed by a fire in a nuclear power plant. Moreover, in recognition of the limits imposed by the circumstances that the affected plants are already built, the rule provides alternatives for achieving the same level of protection as provided by a three-hour barrier and also provides licensees an opportunity to seek an exemption if they can demonstrate that any required fire protection feature will not enhance fire protection safety in the facility or that such modifications may be detrimental to overall facility safety. 10 CFR 50.48(c)(6).

b. III.L. Alternative and Dedicated Shutdown Capability

Movants also challenge the adequacy of the statement of basis and purpose supporting Item III.L which establishes criteria for acceptable alternative or dedicated shutdown capability. A review of the development of these criteria shows that movants' contentions are without merit.

As previously discussed with regard to Item III.G, the Fire Hazard Analyses conducted by licensees after the Browns Ferry fire showed that some plants required alternative or dedicated shutdown systems. Some utilities promptly installed such systems. Staff's experience from reviewing these systems led it to formulate general guidelines for acceptable or dedicated shutdown systems. These guidelines were set out in a document entitled "Staff Position - Safe Shutdown Capability." Copies of staff's

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Point is not acceptable fire protection to assure safe shutdown capability. It has been shown repeatedly that coated cables directly exposed to a flame (for several minutes) will be damaged and will ignite.

Although flame propagation is retarded by the coating, coated cables will burn and add their heat of combustion to the heat input to a compartment in a fire. The lack of adequate separation between redundant divisions of safe shutdown cables at Turkey Point could result in damage to both divisions from direct flame impingement from an exposure fire.

position were sent on November 6, 1979 to all power reactor licensees with plants licensed to operate prior to January 1, 1979.

Staff's position statement provided that in the event of fire the alternate shutdown capability shall be able to achieve and maintain subcritical conditions in the reactor, maintain reactor coolant inventory, achieve and maintain hot standby conditions (for a Pressurized Water Reactor) or hot shutdown (for a Boiling Water Reactor) as defined in the Technical Specifications, achieve cold shutdown within 72 hours and maintain cold shutdown conditions thereafter. These criteria were to be satisified under post-fire conditions where either off-site power is available or is not available for 72 hours. Moreover, during the post-fire shutdown, the reactor coolant system process variables shall be maintained within the range predicted for a loss of normal alternating current power and fission product boundary integrity shall not be affected; i.e. there shall be no damage to fuel cladding, rupture of any primary coolant boundary, or rupture of the containment boundary.

Staff's position statement also established performance goals. These included provision of: (1) a reactivity control function capable of achieving and maintaining cold shutdown reactivity conditions; (2) a reactor coolant makeup function capable of maintaining the reactor coolant level above the top of the core for a BWR and in the pressurizer in a PWR; (3) a reactor heat removal function capable of achieving and maintaining decay heat removal; (4) a process monitoring function capable of providing direct readings of the process variables necessary to perform and control the above functions; and (5) a supporting function capable of providing the process cooling, lubrication and other services necessary to permit the operation of the equipment used for safe shutdown by the systems identified above. In addition, the equipment and systems used to achieve and maintain hot standby conditions at a PWR or hot shutdown at a BWR should be free of fire damage, capable of maintaining such conditions for longer than 72 hours if the equipment required to achieve and maintain cold shutdown is unavailable due to fire damage, and capable of being powered by an on-site emergency power system. Equipment and systems used to maintain cold shutdown conditions should be either free of fire damage or reparable within 72 hours; and equipment and systems used prior to 72 hours after the fire should be capable of being powered by an onsite emergency power system. Equipment and systems used after 72 hours may be powered by off-site power only.

Finally, staff's position statement listed the equipment generally required for hot shutdown of a BWR, hot standby at a PWR, and cold shutdown at both types of reactors. The statement also listed information required for staff review including description and design of the alternate or dedicated shutdown systems and a demonstration that wiring for these systems is independent of equipment wiring in the fire area for which the alternate shutdown system is required.

Item III.L of the proposed rule addressed the criteria for an acceptable alternate shutdown capability. The proposed requirements were derived entirely from the staff's earlier statement of position and essentially just reiterated the guidelines and performance goals described above. Thus, proposed Item III.L was familiar to all licensees. Moreover, in restating the staff's position, the rule clarified the requirements for equipment and systems used to achieve and maintain hot standby conditions at a PWR or hot shutdown at a BWR in the event that fire disabled the equipment needed to achieve and maintain cold shutdown. Instead of requiring this equipment to be capable of maintaining these conditions for longer than 72 hours, the equipment was required to be capable of maintaining such conditions until cold shutdown could be achieved. This clarification provided consistency with the requirement that equipment needed to attain cold shutdown should be reparable within 72 hours.

Commentors contended that the proposed Item III.L was too detailed regarding alternatives to cold shutdown, did not provide a basis for requiring the attainment of cold shutdown within 72 hours, did not provide adequate time for the installation of an alternative or dedicated shutdown system, might result in plant modifications which could be inconsistent with other modifications which may be required pursuant to the Systematic Evaluation Program (SEP),[31] and would require substantial expenditures.

The comments were addressed in the Statement of Consideration accompanying the final rule. Staff explained that it was appropriate to require detailed alternatives to shutdown systems where a single fire could disable systems required to achieve either cold or hot shutdown. Specification of the minimum capability and time requirements for each shutdown condition ensured that safe shutdown would be achieved after a fire. Moreover, staff believes that adequate safety margins would be provided by requiring the reparability of cold shutdown equipment within 72 hours.[32]

[31]The Systematic Evaluation Program is an NRC review of the design of eleven older operating plants licensed before many of the current design features were required. Upon completion of this review, staff may require licensees to install additional safety equipment at these plants.

[32]Similar or shorter time limits have been imposed on the achievement of cold shutdown in the event of other accidents. For example, the Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors require cold shutdown within 36 hours if a Limiting Condition for Operation is not met and corrective actions cannot be taken within 1 hour. NUREG-0212, Revision 2, pp. 3/4 0-1, 4-9, 6-7F. Some deviations from normal operation are so serious that an even quicker response is required. For example, cold shutdown within 21 hours is required if no emergency core cooling system subsystem is operable. *Id.* at 5-7. *See* also, Standardized Technical Specifications for Westinghouse PWRS, NUREG-0452, 3/4 0-1, 5-11, 6-1A, 7-15, 7-17, 7-19 (1978).

Regarding potential conflict with modifications that may be required pursuant to the SEP, staff found that the current need to provide adequate fire protection was not consistent with deferral of the installation of an alternate or dedicated shutdown system pending completion of the SEP review. In addition, staff believes that licensees can design an alternate or dedicated shutdown system which can accommodate anticipated SEP requirements. In response to the contention that the proposed rule did not provide adequate time to install an alternative or dedicated shutdown system, staff modified the rule to provide additional time of up to 30 months.[33]

For the most part, Item III.L of the final rule is a slightly edited and rearranged version of the corresponding Item III.L in the proposed rule. The substantive requirements of this Item are consistent with staff's resolution of the comments on the proposed rule. In addition, the final version of Item III.L contains a specific provision, Item III.L.7, which applies the associated circuit concept of proposed Item III.Q to this particular fire protection feature. The Statement of Consideration explains that this provision was added because most plants currently operating were designed before associated circuits were required to be identified. However, unless these circuits are identified and properly treated, an alternative shutdown system might not function in the event of fire damage to the associated circuits. Therefore, the final rule provides that alternate safe shutdown equipment for a particular fire area shall be known to be isolated

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Similar Technical Specifications apply to other types of plants, including those operated by the petitioners. For example, the Technical Specifications for Millstone Nuclear Power Station, Unit No. 1 require cold shutdown within 24 hours if both core spray subsystems are not operational and cannot be repaired in the specified time when irradiated fuel is in the reactor vessel, both containment cooling systems are not operable and cannot be repaired in the specified time when irradiated fuel is in the reactor vessel, the automatic pressure release subsystem is not operable and cannot be repaired in the specified time when the reactor pressure exceeds 90 pounds per square inch, the isolation condensor is not operable and cannot be repaired in the specified time when the reactor pressure exceeds 90 pounds per square inch, and both emergency power sources are not operable and cannot be repaired in the specified time when irradiated fuel is in the reactor. Appendix A to License No. DPR-21, 3/4 5-1 to 5-9 (December 1977). *See* also Technical Specifications for H.R. Robinson, Unit No. 2 which, among other things require cold shutdown within 72 hours if certain deviations from the normal operation of the Chemical and Volume Control System, Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, Post-Accident Containment Venting Systems and Isolation Seal Water System cannot be restored in 24 hours. Appendix A to Facility Operating License DPR-23, 3.2-1 to 3.3-9 (1973).

Fire damage to plant systems could be as serious as the deviations from operation described above. Thus, the time specified in the rule to achieve the capability to attain cold shutdown is consistent with other license conditions.

[33]Staff did not respond to the comments regarding cost because other utilities had installed such systems.

from associated non-safety circuits in that fire area. The rule also provides that an acceptable method of compliance would be to meet Regulatory Guide 1.75, position 4 which essentially provided that associated circuits should be treated as safety circuits.

Subsequent to publication of the final rule, some licensees requested further clarification of the associated circuit requirement for an alternative shutdown system. On February 20, 1981, the NRC staff provided additional guidance in a letter to all power reactor licensees affected by Appendix R. The letter defines associated circuits related to an alternative shutdown system as circuit whose damage by fire could cause failure or mal-operation of the alternative safe shutdown system. This definition of associated circuits includes safety related as well as non-safety cables which are associated with the alternative or dedicated method of shutdown, are separated from the fire area they are alternate to by less than the requirements in Item III.G.2 and have either (1) electrically unprotected power source in common with the alternate shutdown equipment; (2) a connection to circuits or equipment whose spurious operation will adversely affect the shutdown capability; or (3) a common enclosure with alternative shutdown cables and which are not electrically protected from the post-fire shutdown circuits. This definition is different from the guidance referenced in the rule in that it includes some safety-related cables as well as non-safety-related cables but excludes cables which are electrically protected from shutdown circuits by circuit breakers, fuses, or similar devices. This exclusion of certain cables provides licensees additional flexibility in satisfying the requirements of the rule.

c. III.O Reactor Coolant Pump Lubrication System

Movants have also challenged the adequacy of the statement of basis and purpose supporting Item III.O of the rule. A review of events leading up to promulgation of this aspect of the rule shows there is no merit in movants' contentions.

The lubrication system for reactor coolant pumps has long been recognized as the single largest fire hazard inside the reactor containment. Details of fires caused by leaks or drips of lubricating oil onto nearby hot surfaces were described as early as 1977 in a publicly available report.[34] Additional analyses of the fire hazards associated with the reactor coolant pump lubrication system are contained in several Fire Protection Safety Evaluation Reports prepared by the NRC in 1978. Some of these reports

[34] "Fire Damage Data Analysis As Related To Current Testing Practice For Nuclear Power Application." BNL-NUREG-23364 (October 1977).

were prepared for facilities operated by the utilities seeking a stay of this rule.[35]

Because the fire hazard associated with the reactor pump lubrication system is a generic issue not resolved at all plants, Item P of the proposed rule would have required either an oil collection system or an automatic fire suppression system. 45 Fed. Reg. at 36090. Either system was required to be designed to withstand a Safe Shutdown Earthquake (SSE) to protect nearby safety related equipment from fires resulting from earthquake induced oil leaks. This seismic qualification requirement was contained in the staff's early draft of the proposed rule.[36] Shortly after staff prepared that draft, some licensees, including some of the present movants, were requested to comply with the seismic qualification requirement.[37] Thus, several months before the proposed rule was published, licensees were aware that seismic qualification would be required features which would protect reactors from fire hazards associated with the reactor pump lubrication system. And before the final rule was issued, at least one of these movants committed to the installation of a seismically qualified oil collection system.[38]

In view of this background, it is not surprising that, for the most part, commentors did not challenge the need for a system to protect against fires which could be caused by oil leaking from the reactor cooling pump lubrication system. Some commentors noted that the seismic qualification requirement for either protection system had not been included in previous NRC regulatory guidance documents and contended that compliance with the requirement could not be achieved within the time provided by the rule. Comments were also provided on technical aspects of the proposed seismic design requirements for an oil collection system.[39]

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[35]*See*, for example, the Fire Protection SER for the Haddam Neck Plant, pp. 5-9 and 5-10 (1978).

[36]SECY-80-88 (February 13, 1980).

[37]*See*, for example, letter from D. L. Ziemann to W. G. Council, Vice President, Connecticut Yankee Atomic Power Company (February 22, 1980); and Letter from A. Schwencer to R. Uhrig regarding Turkey Point, Units 3 and 4 (April 3, 1980).

[38]Turkey Point, Units 3 and 4.

[39]The principal technical comments were:

1. An exception should be provided for containments with inerted atmospheres (i.e. atmospheres having substantially reduced levels of oxygen);

2. Because reactor coolant pumps are not necessary for safe shutdown, the seismic design of a lubricating oil collection system need only be able to maintain structural integrity after an SSE, not operability; and

3. Although a seismically qualified oil collection system is good in principle, the only parts of the system which should be designed are those whose failure could expose safety-related equipment to fire.

The final rule on this issue is now contained in Item III.O, which requires a seismically qualified system to collect oil spills on leaks from reactor recirculation pumps. The bases for this requirement are extensively described in the accompanying Statement, which reiterates the well-known, publicly available information in the documents discussed above. The Statement explains that each reactor coolant pump motor contains up to 225 gallons of lubricating oil and that oil leaking from some portions of the lubricating oil system may come in contact with surfaces hot enough to ignite the oil. Such an oil spill could spread outside the coolant pump area and result in an unmitigated fire that could damage redundant divisions of safety related systems required for safe shutdown. Accordingly, the collection system must be capable of collecting lubricating oil from all potential leak sites and draining that oil to a vented closed container that can hold the entire oil inventory. Moreover, the oil collection system must be seismically qualified[40] because an earthquake could cause an oil spill by damaging piping for lubricating oil systems. Any previously installed oil collection system which was not designed to withstand a design basis event must be modified to meet the new seismic qualification criteria.

In addition, the statement responds to the technical comments on the seismic design requirements for the oil collection system. The Statement explains that seismic qualification requirements were already implicitly contained in General Design Criterion 2 - "Design Bases for Protection Against Natural Phenomena," which requires structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. Regulatory Guide 1.29, "Seismic Design Classification," describes an acceptable method for identifying equipment that should be designed to withstand a Safe Shutdown Earthquake. That guidance provides for the seismic qualification of equipment whose earthquake induced failure could reduce the functioning of safe shutdown systems. The reactor coolant pump oil collection system is in the category of such equipment because an earthquake induced failure in that system could cause a fire which would prevent a safety related system from performing its function. Thus, the

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One comment was received regarding the alternative of a seismically qualified fire suppression system. The commentor questioned the ability of such a system to function after an SSE if its water supply system is not designed to withstand an SSE.

[40]Item III.O requires the oil collection system to be qualified to a level which assures that it will withstand a Safe Shutdown Earthquake. This means that the oil collection system should be designed, engineered, and installed, so that an earthquake will not lead to a fire affecting safety related equipment. The scope of this requirement was discussed at the Commission's public meeting of October 27, 1980. Staff stated that oil collection systems would be required to be seismically resistant, but would not be required to satisfy the higher levels of seismic qualification applicable to safe shutdown equipment. Transcript of October 27, 1980 at 19.

seismic qualification of the reactor coolant pump oil collection system is not a departure from previous NRC practice.

The final rule also responds to other comments on the proposed rule. The final rule now provides an exception for reactors with inerted containments, and the implementation schedule has been extended.

Finally, the alternative of an automatic fire suppression system was deleted from the final rule. The Statement of consideration explains this change. Staff realized that a suppression system may not prevent unacceptable fire damage to safety related systems because: (1) a suppression system may not be actuated soon enough after initiation of an oil fire; and (2) as noted by a commentor, the suppression system's water supply is not designed to withstand seismic events. Moreover, in the event of seismically induced or other system malfunctions in the fire suppression system, it is self-evident that timely action by the fire brigade would be difficult because the reactor coolant pumps are located in a relatively inaccessible area within the biological shield which is located inside the containment. This location also would prevent a fire patrol from either fixing a fire suppression system that failed during operation or periodically inspecting the area as is normally required. For these reasons, the alternative of a fire suppression system was deleted from the final rule. However, the final rule provides licensees an opportunity to request an exemption from the requirement of an oil collection system.[41]

Under these circumstances, the Commission believes that the record provides a substantial statement of basis and purpose supporting this aspect of the rule.

2. Agency Choice of Procedure

It is well-established that an agency's decision to use rulemaking or adjudication is in its discretion. *E.g., NAACP* v. *FPC,* 425 U.S. 662, 668 (1976); *NLRB* v. *Bell Aerospace Co.,* 416 U.S. 267, 294 (1974); *SEC* v. *Chenery Corp.,* 332 U.S. 194, 201-03 (1947). Because there is no bright line test to assess an agency's exercise of discretion, courts have focused on the nature of the decision reached in the proceeding. *FTC* v. *Brigadier Industries Corp.,* 613 F.2d 1110, 1117 (D.C. Cir. 1979); *see, United States* v. *Florida East Coast Railway,* 410 U.S. 224, 245 (1973). Rulemaking is usually prospective in scope and non-accusatory in form, directed to the implementation of general policy concerns through legal standards. *FTC* v. *Brigadier*

[41] Only one movant has requested an exemption to be allowed to use a fire suppression system. Staff has accepted that exemption request for evaluation and has tolled the compliance date for this item pursuant to 10 CFR 50.48(c)(6).

Industries Corp., supra, 613 F.2d at 1117. Rulemaking is particularly appropriate for the resolution of generic issues. *Minnesota* v. *NRC,* 602 F.2d 412 (D.C. Cir. 1979). Adjudication, on the other hand, is individual in impact and often condemnatory in purpose, directed to the determination of the legal status of particular persons or practices through the application of preexisting legal standards. *FTC* v. *Brigadier Industries Corp., supra.*

In this instance, although the efforts of the Commission regarding fire protection may have been triggered by the pendancy of unresolved issues involving certain plants, without question the rule addressed only generic concerns applicable to all plants, including those for which the staff previously accepted different features. Indeed, Items III.G and III.O, and by implication Item III.L, merely seek to establish for all licensees those safety measures that some have already instituted voluntarily. But, even if these general standards may actually affect only a few, or even one licensee, that circumstance does not make the agency's utilization of rulemaking improper. *See Hercules, Inc.* v. *EPA,* 598 F.2d 91, 118 (D.C. Cir. 1978); *Southern Terminal Corp.* v. *EPA,* 504 F.2d 646, 661 & n.13 (1st Cir. 1974).[42] In any event, these movants are not the only persons affected by various aspects of the Fire Protection Rule. The interest in seeing that nuclear power plants are protected from the potential destruction that fire can cause is a broad one affecting the public as a whole. Accordingly, rulemaking was an appropriate mechanism for the Commission to resolve fire protection issues.

Petitioners also challenge rulemaking as improper because of alleged material factual disputes regarding the requirements of Appendix R that could only be resolved by individual adjudications. However, movants have not provided a single, specific instance of a disputed fact in their request for a stay. Even if they had provided such examples, this contention is unlikely to succeed because it does not recognize that it is within the purview of an agency's authority to settle factual issues within the context of a rulemaking. *Ethyl Corp.* v. *EPA,* 541 F.2d 1, 28-29 & n.58 (D.C. Cir.) *(en banc), cert. denied,* 420 U.S. 941 (1976).[43] To the extent factual questions relating to

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[42]In *Hercules, Inc.* v. *EPA,* the District of Columbia Circuit held that even though a rule limiting the discharge of toxic substances into waterways might apply to only two manufacturers, the use of rulemaking rather than adjudication to achieve that limitation was proper. Likewise, the unique effect upon a parking lot of an air pollution control rule restricting the number of parking spaces on a regional basis was found not to require an adjudicative hearing in *Southern Terminal Corp.* v. *EPA.*

[43]In *Ethyl Corp.* v. *EPA,* the District of Columbia Circuit recognized that in determining whether gasoline lead additives would endanger the public so as to authorize EPA to limit their use, the agency possessed the same fact-finding powers in rulemaking as those of a jury, allowing it to draw conclusions from uncertain or even conflicting scientific evidence to form the proper basis for the regulation.

generic fire protection issues existed, their resolution by means of rulemaking undoubtedly was preferable to relitigating such questions in individual adjudicatory proceedings. *See Minnesota* v. *NRC*, 602 F.2d 412, 416-17 (D.C. Cir. 1979); *Ecology Action* v. *AEC*, 492 F.2d 998, 1002 (2d Cir. 1974).

Likewise without substance is the petitioners' claim that Section 189 of the Atomic Energy Act of 1954 and 10 CFR 2.204 require the Commission to hold plant-by-plant adjudicative hearings. Movants' arguments are incorrect because Section 187 of the Atomic Energy Act, 42 U.S.C. § 2237, provides for license modifications by rule. Certainly, to the extent that a rule is based on "the general characteristics of an industry, rational decision is not furthered by requiring the agency to lose itself in an excursion into detail that too often obscures fundamental issues rather than clarifies them." *WBEN, Inc.* v. *United States*, 396 F.2d 601, 618 (D.C. Cir.), *cert. denied*, 393 U.S. 914 (1968); *see California Citizens Band Association* v. *United States*, 375 F.2d 43, 50-53 (9th Cir.), *cert. denied*, 389 U.S. 844 (1967).[44] Thus, reactor license modification by means of notice and comment rulemaking is well within the Commission's authority.

It also is apparent that the NRC did not abuse its discretion by deciding to initiate a rulemaking to consider generic fire protection issues despite its earlier attempts to promote fire protection by dealing with individual licensees. An agency's previous initiation of a case-by-case method of resolving a problem cannot be raised as a bar to any later efforts to resolve generic issues by rulemaking. *Pacific Coast European Conference* v. *United States*, 350 F.2d 197, 205-06 (9th Cir.), *cert. denied*, 382 U.S. 958 (1965); *see* also *Siegel* v. *AEC*, 400 F.2d 778 (D.C. Cir. 1968).[45] Indeed, in this instance it was not the whim of the Commisssion that caused the change in procedure. Rather, the more informal method of consultation with individual licensees previously used had to be abandoned because of an inability to achieve voluntary implementation of the necessary fire protection guidelines.

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[44]In both *WBEN, Inc* v. *United States* and *California Citizens Band Ass'n* v. *United States*, the courts rejected the argument that Section 316 of the Communications Act of 1934, which provided for a hearing when the FCC seeks by order to amend a license, was applicable when a Commission rule relating to generic issues had the effect of modifying a license.

[45]In *Pacific Coast European Conference* v. *United States*, the Ninth Circuit found that the Federal Maritime Commission could abandon its case-by-case consideration of the propriety of certain shipping contracts and revert to rulemaking to establish a general regulation. This Commission itself has indicated previously that a pending adjudication need not bar rulemaking on a contested subject. Citing a pending proposed rule, the Atomic Energy Commission in 1967 ordered that the issue of the protection of nuclear plants from enemy attack not be considered in an ongoing adjudication. The final rule excluding that question from licensing proceedings was upheld by the District of Columbia Circuit in *Siegel* v. *AEC*.

3. Backfit and Retroactivity

Turning next to the questions of supposed backfitting and retroactivity under the fire protection rule, it is apparent that movants' assertions regarding the applicability of 10 CFR § 50.109(a) and the agency's purported failure to comply with the regulations' requirement that the Commission find that a proposal "backfit" will provide "substantial, additional protection which is required for the public health and safety" are without merit. First, their contention must fail because it does not account for the language of subsection (b) of Section 50.109, which provides that "[n]othing in this Section shall be deemed to relieve a holder of a ...license from compliance with the rules, regulations, or orders of the Commission." A rule that requires a licensee to take some action to modify an existing facility after issuance of a construction permit does not require any Commission consideration under Section 50.109(a). Moreover, even if Section 50.109(a) did apply, it is evident that the Commission has complied with the rule's requirement that the agency find a plant design change will provide additional protection that is required for the public health and safety. Section 50.109 is directed at ensuring that plant modifications will be carefully appraised before being imposed; certainly, the Commission's clarification of the proposed rule to require the uniform application of only three of its twenty items is a reflection of such careful concern.[46] Further, the importance of Item III.G, associated Item III.L, and Item III.O as fire protection measures providing added protection to the public health and safety is made clear by the explanation accompanying the rule. The former two are designed to ensure that a safe reactor shutdown can always be achieved and maintained in the event of a fire, 45 Fed. Reg. at 76605, while the latter is aimed at preventing several large, smoky oil fires in the containment area, which would not be readily accessible because of the high radiation levels and could cause the breakdown of safety-related equipment, *id.* at 76608-09. Based as they are on the Commission's "increased knowledge and experience developed on fire protection matters over the last several years," *id.* at 76603, the imposition of these requirements upon all licensees supported by an expressed rationale is more than sufficient to satisfy Section 50.109(a).

So too, the question of whether the Commission has required impermissible retroactive modifications through its rulemaking is one which is without substance. To whatever extent the Commission's rulemaking efforts may be

[46]Besides Items III.G and III.O, all licensees will be required to comply with Item III.J, relating to eight-hour emergency lighting, regardless of any efforts already undertaken to follow the guidance of Appendix A concerning such lighting. The petitioners do not challenge the Item III.J requirements.

considered retroactive, it is apparent that in this instance the regulations are fully in accord with the recognized standard for denoting a proper retroactive rule: one that is reasonable under the circumstances. *See California* v. *Simon,* 504 F.2d 430, 438-39 (Temp. Emer. Ct. App.), *cert. denied,* 419 U.S. 1021 (1974); 2 K. Davis, *Administrative Law Treatise* § 7:23, at 109 (2d ed. 1978). In determining whether a retroactive rule is reasonable, the relevant factors include the rule's degree of retroactivity as measured by whether it is an abrupt departure from established practice or an attempt to resolve unsettled questions, the complaining party's reliance upon the agency's former policies, the burden imposed by the retroactive rule, and the need for administrative flexibility in light of changing circumstances. *See, e.g., New York Telephone Co.* v. *FCC,* 631 F.2d 1059, 1068 (2d Cir. 1980); *Tennessee Gas Pipeline Co.* v. *FERC,* 606 F.2d 1094, 1116 n.77 (D.C. Cir. 1979), *cert. denied,* 445 U.S. 920 (1980).

Movants object to the requirements in Item III.G to the extent they go beyond fire protection modifications previously approved by the NRC staff. The Commission, on the basis of public discussions with the staff, determined protection of the public health and safety would require implementation of Item III.G at all plants to ensure a uniform level of safety. In view of the continuing growth of knowledge regarding fire safety, licensees cannot reasonably expect that staff's acceptance of certain fire protection features will forever bar the imposition of additional features when the facts show them to be required to protect against fire. Futher, with regard to Item III.O, there is no specific showing by any movant regarding either reliance on a former staff position or the burden of compliance.

In addition, the other factors in the balance strongly support the reasonableness of the rule's purported retroactive effects. The degree of retroactivity of the rule is not substantial. The rule is the culmination of an ongoing administrative process and, as such, is intended to resolve a few remaining generic fire protection issues. Moreover, the rule is consistent with the recognized need for administrative flexibility in the regulation of nuclear plants. Under Section 186(a) of the Atomic Energy Act, 42 U.S.C. § 2236, the NRC has the authority to apply new standards to already licensed plants, consistent with evolving concepts of what measures are necessary to protect the public health and safety.

> Congress, when it enacted Section 186(a) in 1954, must have envisioned that licensing standards, especially in the areas of health and safety regulations, would vary over time as more was learned about the hazards of generating nuclear energy. Insofar as those standards become more demanding, Congress surely would have wanted the new

standards, if the Commission deemed it appropriate, to apply to those nuclear facilities already licensed.

Ft. Pierce Utilities Authority v. *United States*, 606 F.2d 986, 996 (D.C. Cir.) (footnote omitted), *cert. denied*, 444 U.S. 842 (1979); *see* also *General Telephone Co.* v. *United States*, 449 F.2d 846, 863-64 (5th Cir. 1971). Under the circumstances, and in light of the rule's expressed purpose of protecting the public health and safety by ensuring that licensees take all prudent measures to prevent fires and are equipped adequately to fight any fire that may start while still maintaining control over reactor processes, we are unconvinced that movants can succeed in showing that most of the requirements in Items III.G, III.L, and III.O violate the precept against unreasonable retroactive rulemaking.

B. Irreparable Injury

Movants contend they will be substantially and irreparably harmed because they will be required to make substantial investment in these fire protection features required by Items III.G and III.L which they believe unwarranted. In particular, on the basis of their interpretation of the term "associated circuits" and their belief that the Commission will strictly adhere to the separation criteria in Section III.G, movants contend the rule offers them no alternative but to install a dedicated shutdown system.[47]

Movants' claims are without merit. Item III.G of the final rule provides three alternative fire protection features which do not require analysis to demonstrate the protection of redundant safe shutdown equipment and establishes the acceptable distance in the physical separation distance at twenty feet instead of the fifty feet in the proposed rule.[48] Moreover, the rule now also provides an exemption procedure which can be initiated by a licensee's assertion that any required fire protection feature will not enhance safety in the facility or that such modifications may be detrimental to overall facility safety. 10 CFR 50.48(c)(6). If the Director, Nuclear Reactor Regulation determines that a licensee has made a *prima facie* showing of a sound technical basis for such an assertion, then the implementation dates of the rule are tolled until final Commission action on the exemption request. We understand that movants have filed such exemption requests with the staff. Under these circumstances, the Commis-

[47]We also note that because the definition of associated circuits has been clarified there is no longer any need to consider a claim based on movants' definition.

[48]We also note that even this reduced twenty-foot separation criterion will not be applied rigidly. *See* Transcript of Commission Meeting of October 21, 1980 at 17-18; Transcript of Commission Meeting of October 27, 1980 at 20-2.

sion cannot now conclude that movants' only method of compliance with this rule would be the installation of an alternate or dedicated shutdown system. Moreover, even if an alternate or dedicated shutdown system is ultimately required, movants have not provided details of cost estimates which can be reviewed and corroborated by the staff. Thus, the Commission finds that movants have not satisfied their burden of demonstrating irreparable injury if they are required to comply with Items III.G and III.L.

Finally, petitioners' request for a stay of Item III.O does not present any reasons why this item will result in irreparable injury. The bald request for a stay of this item provided the Commision with no basis for a reasoned analysis of this request. Thus, it fails totally to meet petitioners' burden of persuasion in this regard. Moreover, as noted above, petitioners who believe that Item III.O will cause irreparable injury have pursued the exemption procedure in 10 CFR 50.48(c)(6). Until that procedure is concluded, it would be premature for the Commission to grant a stay of this item.

C. Harm To Others And The Public Interest

In addition to the above considerations supporting the merits of the rule, the Commission finds unpersuasive movants' other arguments for a stay. As the movants themselves state, the appropriate inquiry is whether a stay would serve the public interest. The Commission believes it would not.

Since the Browns Ferry fire of 1975, the Commission and its licensees have been involved in the development of an adequate fire protection program at each facility. During that time, an evolving program of fire protection has led to the installation of many fire protection features. However, not all facilities have implemented all the fire protection features which the NRC believes are warranted on the basis of extensive fire hazard analysis. Thus, in order to complete the fire protection program which the Commission believes is necessary to protect the public health and safety, a rule has been promulgated specifying acceptable ways of satisfying certain unresolved generic fire safety requirements. The Commission believes that the extensive review of safety at nuclear power plants demonstrates that all aspects of the rule are necessary to assure public health and safety, and does not believe that a stay of Items III.G, III.L, and III.O is consistent with that goal.

For all the above reasons, the motion to stay Items III.G, III.L, and III.O of this rule is denied.

It is so ORDERED.

The separate views of Commissioner Ahearne and concurring opinion of Commissioner Bradford are attached.*

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 12th day of June, 1981

*[The separate views of Commissioner Ahearne and Concurring opinion of Commissioner Bradford can be found after the enclosures to the Commission Order, starting on page 827.]

Enclosure A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

June 20, 1977

MEMORANDUM FOR: Raymond F. Fraley, ACRS

FROM: Guy A. Arlotto, Director, ES, SD

SUBJECT: PROPOSED DRAFT 2, REVISION 1 OF REGULATORY GUIDE 1.120, "FIRE PROTECTION GUIDELINES FOR NUCLEAR POWER PLANTS"

Enclosed for the use of the Regulatory Activities Subcommittee and the Fire Protection Subcommittee are ten copies of Draft 2 of the proposed revision of Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants," dated June 17, 1977.

This guide was considered by the Regulatory Guide Subcommittee in closed session in June 1976. The subcommittee withheld comment at that time, and the guide was issued for public comment in June 1976. The ACRS Fire Protection Work Group held two open meetings on the guide in July and August 1976. An additional open meeting was held with the ACRS Fire Protection Work Group on May 4, 1977, and additional public comments were solicited at that time.

Subsequent to the April distribution of Draft 1, Rev. 1 to Regulatory Guide 1.120 the staff has received 19 letters commenting on the Guide. These comment letters, representing 12 different sources, can be categorized in three groups; a) procedural comments directed at the guide issuance, b) general comments addressing the staff philosophy on fire barriers, the use of water and the fire hazards analysis and c) recommendations for specific changes to the guide.

A review of all the comments received to date has resulted in a few minor changes which have been addressed in the Resolution of Public Comments. It should be noted that an overwhelming majority of the comments received duplicated comments previously received and considered by the staff in the development of Rev. 1 of the Regulatory Guide. The fundamental staff philosophy regarding fire protection of nuclear power plants remains unchanged.

This draft of the guide also reflects a number of changes made by our technical editing staff. These changes, since they do not affect the staff position, are not discussed. Comparative text is based on Revision O issued June 30, 1976. The revised guide has been reviewed by all the relevant offices and divisions within NRC.

Also enclosed are ten copies of the public comment letters and a discussion of the resolution of public comments.

ACRS concurrence in the regulatory position is requested.

Guy A. Arlotto, Director
Division of Engineering Standards
Office of Standards Development

Enclosures:
1. R.G. 1.120, Rev. 1, Dråft 2, June 17, 1977
2. Statement of Resolution of Public Comments on R.G. 1.120
3. Ltrs. of Public Comment on R.G. 1.120

Resolution of Public Comments Received on Draft 1, Revision 1 to R. G. 1.120, "Fire Protection Guidelines for Nuclear Power Plants" (Received Prior to June 1, 1977)

As evidenced by the letters of comment there appear to be five general areas that received frequent comment and warrant discussion or clarification.

1. Scope of R. G. 1.120

The scope of R. G. 1.120 encompasses all safety-related structures, systems and components and all plant areas that either contain or could present a fire exposure hazard to safety-related structures, systems and components. It is the protection of all the safety-related aspects of the power plant that provides the necessary defense-in-depth capability to achieve and maintain safe (cold) shutdown and to minimize releases of radioactivity to the environment.

2. Use of Water to Fight Fires

The position taken by a number of commenters is that the staff has overemphasized the use of water in general and particularly the use of automatic water suppression systems to combat fires. This comment was made before, on Revision 0 of the guide, and was carefully considered by the staff and its consultants in the development of Revision 1. As a result, Revision 1 of the guide is significantly different than the previous revision in this area. Safety-related cable trays in most cases can now be protected by automatic water suppression systems providing *area* protection rather than protection *directed at each cable tray* as previously recommended in Revision 0. In some situations (as detailed in the guide) manual protection; i.e., hose and standpipe systems will suffice. Furthermore, in the 18 plant areas listed in Section C.6 of the Reg. Guide, an automatic water suppression system is specifically recommended in only one area; namely the cable spreading room. The staff has carefully weighed the advantages and disadvantages of water as a fire suppression agent and has concluded that a proper balance has been achieved and no further changes are required.

3. Three Hour Rated Fire Barriers

Many commenters are of the opinion that the ratings of fire barriers separating redundant safety-divisions and fire barriers separating safety-related structures, systems and components from non-safety-related fire hazards should be determined by a fire hazards analysis rather than using a fixed three hour fire rating as recommended in the Regulatory Guide. This comment is not new and has been considered in the development of Revision 1. The staff position remains unchanged and is based on the following:

1. A precise quantitative relationship between the combustibles in a given fire area and the rating of a fire barrier sufficient to allow the consumption of all combustibles in the area without breaching the barrier does not exist.

2. The three hour barrier rating gives the plant fire brigade an extra margin of time in which to combat a fire. This is particularly important considering the remote location of nuclear plants and the potential time delay in the arrival of additional fire fighting capability in the unlikely event that such backup to the plant fire brigade is needed.

3. A three hour rated fire barrier is the minimum rated fire barrier that is constructed entirely of non-combustible material.

4. Three hour rated barriers, in industrial usage, are not uncommon in the protection of "high value risks." In some commercial applications four hour rated fire barriers are constructed to define the "maximum foreseeable loss."

5. Three hour barriers provide some conservatism to compensate for the uncertainties introduced by unforeseen transient combustibles over the forty year operating life of the plant.

4. Reliance on Cable Qualified to IEEE-383

Cable qualified to the fire test in IEEE-383 is still thought of by many commenters as being sufficiently fire retardant to require little if any fire protection. The staff regards the IEEE-383 fire test as strictly a screening test, i.e., the test will eliminate the notably poor performing cable insulation, but passing the test is not in itself a necessary and sufficient condition for fire retardancy. Two very important variables which are standardized in the test are cable tray fill and cable spacing. *Passage of the 383 test is only indicative of fire retardancy in the standardized test configuration.* It is not possible to extrapolate these test results to other cable loading configurations. The staff recommends, as a minimum, the use of cable qualified to IEEE-383 but, in recognition of the above, additional fire protection is also recommended. The staff position remains unchanged.

5. Fire Hazards Analysis

A strongly voiced concern by a number of commenters, is that the Reg. Guide is overly specific and that all fire protection should be based solely on the fire hazards analysis. In addition, many feel that the fire hazards analysis has no other purpose but to verify the specific recommendations given in the guide. The balance between general guidelines and specific guidelines is a difficult one to achieve in any standards document. Should the scale swing too much to either side, the result would be either a document that is so general that it becomes useless or so specific that it restricts the creativity of the plant designer. The staff has given specific guidance in areas where we felt specific guidance was needed; however, in many of the eighteen areas detailed in Section c.6 of the guide the choice of the suppression agent is left to the designer and will be determined by a hazards analysis. The choice of detector type and specific location as well as the ratings of fire barriers that may be necessary within a given safety division will all be determined by the fire hazards analysis. The fire hazards analysis will be the principal design document determining the fire protection in a large portion of the plant.The staff believes that a reasonable balance between specificity and generality has been achieved in the guide. This belief is supported by the fact that comments on both sides of this argument have been received.

Changes Resulting From Public Comment

Page 5 - B.2

1. Added "Cable" to heading of Section 2 (IEEE)

2. Added "vital" in place of "paramount" (GE)

Page 22 - C.4.a(1)(a)

1. Added "in non-safety-related areas" (GE, PSE&G, IEEE, S&L). Clarification made to avoid the interpretation by some that equipment needed to be separated from itself by 3-hr rated fire barriers.

Page 25-C.4.a(9)

1. Guidance added regarding sizing of gas suppression systems to compensate for leakage through floor drains as an alternative to providing drain seals. (IEEE)

Page 26 - C.4.b(3)

1. Changed "highest" to "high" and "lowest" to "low". (IEEE).

Page 27

1. **C.4.c(1)** The guidance regarding electrical conduit has been revised, providing additional design alternatives. The previous recommendation in the guide was directed to the use of only rigid steel tubing as electrical conduit. This guidance has been broadened to include the use of any metallic tubing except for thin walled metallic tubing. Guidance has also been provided for the use of flexible conduit to connect to equipment. (Bechtel, GE, Yankee Atomic, IEEE).

   Metallic conduit, although not considered a rated fire barrier, does provide a degree of protection for electrical cables from the effects of exposure fires and electrically initiated cable fires. Experience has shown that thin walled metallic tubing provides significantly less protection for the cables, particularly when considering the exposure fire, than other types of metallic conduit.

2. **C.4.c(2)** Line 2 added "in non-safety-related areas".

3. **C.4.c(2)** The previous recommendation for smoke detectors in all areas through which safety-related cables pass has been revised. Area smoke detectors are now recommended as back-up to the continuous-line-detectors only when manual suppression is relied upon to protect cable tray configurations of six or less as detailed in the guide. (Yankee Atomic)

   After additional discussion it was decided that the recommendation for area smoke detectors in all areas containing safety-related cables was excessive when automatic water suppression systems and line-type-detectors are provided and redundant cable divisions are separated by barriers having three hour ratings. The recommendation for continuous line-type-detectors has been retained since these detectors will provide the quickest indication of a fire.

Page 29 C.4.c(3)

1. The last sentence was changed to correct an interpretation, made by some, that fire testing was required after the penetration seals were installed in the plant. (Yankee Atomic, IEEE)

Page 34 C.5.b(3)(b)

1. Added "Class 1E" (Yankee Atomic, Sargent & Lundy, IEEE)

   Any electrical equipment that is "permanently" connected to the Class 1E electrical system must be Seismic Category I and Class 1E.

Page 47 C.6.f

1. Additional clarification has been provided regarding the separation of "Remote Safety-Related-Panels". Separation of redundant safety-related panels by 3 hour rated barriers is recommended. (CE)

   This guidance is consistent with that given in Section C.4a(1)(b) and does not represent a change in staff position.

Enclosure B

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

February 20, 1981

TO ALL POWER REACTOR LICENSEES
WITH PLANTS LICENSED PRIOR TO JANUARY 1, 1979

SUBJECT: FIRE PROTECTION RULE
(45 FR 76602, NOVEMBER 19, 1980)
- Generic Letter 81-12

Paragraph 50.48(b) of 10 CFR Part 50, which became effective on February 17, 1981, requires all nuclear plants licensed to operate prior to January 1, 1979 to meet the requirements of Sections III.G, III.J and III.O of Appendix R to 10 CFR Part 50 regardless of any previous approvals by the Nuclear Regulatory Commission (NRC) for alternative design features for those items. This would require each licensee to reassess all those areas of the plant "... where cables or equipment, including associated non-safety circuits, that could prevent operation or cause maloperation due to hot shorts, open circuits or shorts to ground or (sic) redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment ..."[1] to determine whether the requirements of Section III.G.2 of Appendix R are satisfied. If not, the licensee must provide alternative shutdown capability in conformance with Section III.G.3 or request an exemption if there is some justifiable basis.

Paragraph 50.48(c)(5) requires that any modifications that the licensee plans in order to meet the requirements of Section III.G.3 of Appendix R must be reviewed and approved by the NRC. This paragraph also requires that the plans, schedules and design descriptions of such modifications must be submitted by March 19, 1981. To expedite our review process and

[1]Quoted from Section III.G.2 of Appendix R to 10 CFR Part 50. Note that the "or" preceding "redundant trains" is a typographical error and should read "of redundant trains".

reduce the number of requests for additional information with regard to this review, we are enclosing two documents which specify the information that we will require to complete our reviews of alternative safe shutdown capability. Enclosure 1 is *"Staff Position Safe Shutdown Capability". This document was originally sent to you in late 1979.* Section 8 specifies the information required for staff review. If you have already submitted any of the information required, you need only reference that previous submittal. Enclosure 2 indicates the additional information needed to ensure that associated circuits for alternative safe shutdown equipment is included in your reassessment and in our review. If you made no modifications that were required to provide alternative safe shutdown capability and if your reassessment concludes that alternative safe shutdown capability in accordance with the provisions of Section III.G.3 is not necessary, you do not have to provide the information requested by these Enclosures.

Finally, we request that as part of your submittal of plans and schedules for meeting the provisions of Paragraphs (c)(2), (c)(3) and (c)(4) of 10 CFR 50.48 as required by Paragraph 50.48(c)(5), you include the results of your reassessment of the design features at your plant for meeting the requirements of Sections III.G, III.J and III.O of Appendix R to 10 CFR Part 50.

This detailed information need not accompany the design description that must be submitted by March 19, 1981. However, we request that it be submitted as soon as possible, but no later than May 19, 1981.

This request for information was approved by GAO under a blanket clearance number R0071 which expires September 30, 1981. Comments on burden and duplication may be directed to the U. S. General Accounting Office, Regulatory Reports Review, Room 5106, 441 G Street, N. W., Washington, D. C. 20548.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
1. Staff Position
2. Request for Additional Information

Enclosure 1

STAFF POSITION
SAFE SHUTDOWN CAPABILITY

Staff Concern

During the staff's evaluation of fire protection programs at operating plants, one or more specific plant areas may be identified in which the staff does not have adequate assurance that a postulated fire will not damage both redundant divisions of shutdown systems. This lack of assurance in safe shutdown capability has resulted from one or both of the following situations:

* Case A: The licensee has not adequately identified the systems and components required for safe shutdown and their location in specific fire areas.

* Case B: The licensee has not demonstrated that the fire protection for specific plant areas will prevent damage to both redundant divisions of safe shutdown components identified in these areas.

For Case A, the staff has required that an adequate safe shutdown analysis be performed. This evaluation includes the identification of the systems required for safe shutdown and the location of the system components in the plant. Where it is determined by this evaluation that safe shutdown components of both redundant divisions are located in the same fire area, the licensee is required to demonstrate that a postulated fire will not damage both divisions or provide alternate shutdown capability as in Case B.

For Case B, the staff may have required that an alternate shutdown capability be provided with is independent of the area of concern or the licensee may have proposed such a capability in lieu of certain additional fire protection modifications in the area. The specific modifications associated with the area of concern along with other systems and equipment already independent of the area form the alternate shutdown capability. For each plant, the modifications needed and the combinations of systems which provide the shutdown functions may be unique for each critical area; however, the shutdown functions provided should maintain plant parameters within the bounds of the limiting safety consequences deemed acceptable for the design basis event.

Staff Position

Staff shutdown capability should be demonstrated (Case A) or alternate shutdown capability provided (Case B) in accordance with the guidelines provided below:

1. **Design Basis Event**

   The design basis event for considering the need for alternate shutdown is a postulated fire in a specific fire area containing redundant safe shutdown cables/equipment in close proximity where it has been determined that fire protection means cannot assure that safe shutdown capability will be preserved. Two cases should be considered: (1) offsite power is available; and (2) offsite power is not available.

2. **Limiting Safety Consequencess and Required Shutdown Functions**

   2.1 No fission product boundary integrity shall be affected:

   a. No fuel clad damage;
   b. No rupture of any primary coolant boundary;
   c. No rupture of the containment boundary.

   2.2 The reactor coolant system process variables shall be within those predicted for a loss of normal ac power.

   2.3 The alternate shutdown capability shall be able to achieve and maintain subcritical conditions in the reactor, maintain reactor coolant inventory, acheive and maintain hot standby* conditions (hot shutdown* for a BWR) for an extended period of time, achieve cold shutdown* conditions within 72 hours and maintain cold shutdown conditions thereafter.

   * As defined in the Standard Technical Specifications.

3. **Performance Goals**

   3.1 The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

   3.2 The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWR's and in the pressurizer for PWR's.

   3.3 The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

3.4 The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

3.5 The supporting function shall be capable of providing the process cooling, lubrication, etc. necessary to permit the operation of the equipment used for safe shutdown by the systems identified in 3.1 - 3.4.

3.6 The equipment and systems used to achieve and maintain hot standby conditions (hot shutdown for a BWR) should be (1) free of fire damage; (2) capable of maintaining such conditions for an extended time period longer than 72 hours if the equipment required to achieve and maintain cold shutdown is not available due to fire damage; and (3) capable of being powered by an onsite emergency power system.

3.7 The equipment and systems used to achieve and maintain cold shutdown conditions should be either free of fire damage or the fire damage to such systems should be limited such that repairs can be made and cold shutdown conditions achieved within 72 hours. Equipment and systems used prior to 72 hours after the fire should be capable of being powered by an onsite emergency power system; those used after 72 hours may be powered by offsite power.

3.8 These systems need not be designed to (1) seismic category I criteria; (2) single failure criteria; or (3) cope with other plant accidents such as pipe breaks or stuck valves (Appendix A BTP 9.5-1), except those portions of these systems which interface with or impact existing safety systems.

4. PWR Equipment Generally Necessary For Hot Standby

(1) Reactivity Control
Reactor trip capability (scram). Boration capability e.g., charging pump, makeup pump or high pressure injection pump taking suction from concentrated borated water supplies, and letdown system if required.

(2) Reactor Coolant Makeup
Reactor coolant makeup capability, e.g., charging pumps or the high pressure injection pumps. Power operated relief valves may be required to reduce pressure to allow use of the high pressure injection pumps.

(3) **Reactor Coolant System Pressure Control**

Reactor pressure control capability, e.g., charging pumps or pressurizer heaters and use of the letdown systems if required.

(4) **Decay Heat Removal**

Decay heat removal capability, e.g., power operated relief valves (steam generator) or safety relief valves for heat removal with a water supply and emergency or auxiliary feedwater pumps for makeup to the steam generator. Service water or other pumps may be required to provide water for auxiliary feed pump suction if the condensate storage tank capacity is not adequate for 72 hours.

(5) **Process Monitoring Instrumentation**

Process monitoring capability e.g., pressurizer pressure and level, steam generator level.

(6) **Support**

The equipment required to support operation of the above described shutdown equipment e.g., component cooling water service water, etc. and onsite power sources (AC, DC) with their associated electrical distribution system.

5. **PWR Equipment Generally Necessary For Cold Shutdown***

(1) **Reactor Coolant System Pressure Reduction to Residual Heat Removal System (RHR) Capability**

Reactor coolant system pressure reduction by cooldown using steam generator power operated relief valves or atmospheric dump valves.

(2) **Decay Heat Removal**

Decay heat removal capability e.g., residual heat removal system, component cooling water system and service water system to removal heat and maintain cold shutdown.

(3) **Support**

Support capability e.g., onsite power sources (AC & DC) or offsite after 72 hours and the associated electrical distribution system to supply the above equipment.

* Equipment necessary in addition to that already provided to maintain hot standby.

6. **BWR Equipment Generally Necessary For Hot Shutdown**

(1) **Reactivity Control**

Reactor trip capability (scram).

(2) **Reactor Coolant Makeup**

Reactor coolant inventory makeup capability e.g., reactor core isolation cooling system (RCIC) or the high pressure coolant injection system (HPCI).

(3) **Reactor Pressure Control and Decay Heat Removal**

Depressurization system valves or safety relief valves for dump to the suppression pool. The residual heat removal system in steam condensing mode, and service water system may also be used for heat removal to the ultimate heat sink.

(4) **Suppression Pool Cooling**

Residual heat removal system (in suppression pool cooling mode) service water system to maintain hot shutdown.

(5) **Process Monitoring**

Process monitoring capability e.g., reactor vessel level and pressure and suppression pool temperature.

(6) **Support**

Support capability e.g., onsite power source (AC & DC) and their associated distribution systems to provide for the shutdown equipment.

7. **BWR Equipment Generally Necessary For Cold Shutdown***

At this point the equipment necessary for hot shutdown has reduced the primary system pressure and temperature to where the RHR system may be placed in service in RHR cooling mode.

(1) **Decay Heat Removal**

Residual heat removal system in the RHR cooling mode, service water system.

(2) **Support**

Onsite sources (AC & DC) or offsite after 72 hours and their associated distribution systems to provide for shutdown equipment.

* Equipment provided in addition to that for achieving hot shutdown.

8. **Information Required For Staff Review**

(a) Description of the systems or portions thereof used to provide the shutdown capability and modifications required to achieve the alternate shutdown capability if required.

(b) System design by drawings which show normal and alternate shutdown control and power circuits, location of components, and that wiring which is in the area and the wiring which is out of the area that required the alternate system.

(c) Demonstrate that changes to safety systems will not degrade safety systems. (e.g., new isolation switches and control switches should meet design criteria and standards in FSAR for electrical equipment in the system that the switch is to be installed; cabinets that the switches are to be mounted in should also meet the same criteria (FSAR) as other safety related cabinets and panels; to avoid inadvertent isolation from the control room; the isolation switches should be keylocked, or alarmed in the control room if in the "local" or "isolated" position; periodic checks should be made to verify switch is in the proper position for normal operation; and a single transfer switch or other new device should not be a source for a single failure to cause loss of redundant safety systems).

(d) Demonstrate that wiring, including power sources for the control circuit and equipment operation for the alternate shutdown method, is independent of equipment wiring in the area to be avoided.

(e) Demonstrate that alternate shutdown power sources, including all breakers, have isolation devices on control circuits that are routed, through the area to be avoided, even if the breaker is to be operated manually.

(f) Demonstrate that licensee procedure(s) have been developed which describe the tasks to be performed to effect the shutdown method. A summary of these procedures should be submitted.

(g) Demonstrate that spare fuses are available for control circuits where these fuses may be required in supplying power to control circuits used for the shutdown method and may be blown by the effects of a cable spreading room fire. The spare fuses should be located convenient to the existing fuses. The shutdown procedure should inform the operator to check these fuses.

(h) Demonstrate that the manpower required to perform the shutdown functions using the procedures of (f) as well as to provide fire brigade members to fight the fire is available as required by the fire brigade technical specifications.

(i) Demonstrate that adequate acceptance tests are performed. These should verify that: equipment operates from the local control station when the transfer or isolation switch is placed in the "local" position and that the equipment cannot be operated from the control room; and that equipment operates from the control room but cannot be operated at the local control station when the transfer or isolation switch is in the "remote" position.

(j) Technical Specifications of the surveillance requirements and limiting conditions for operation for that equipment not already covered by existing Tech. Specs. For example, if new isolation and control switches are added to a service water system, the existing Tech. Spec. surveillance requirements on the service water system should add a statement similar to the following:

"Every third pump test should also verify that the pump starts from the alternate shutdown station after moving all service water system isolation switches to the local control position."

(k) Demonstrate that the systems available are adequate to perform the necessary shutdown functions. The functions required should be based on previous analyses, if possible (e.g., in the FSAR), such as a loss of normal a.c. power or shutdown on a Group I isolation (BWR). The equipment required for the alternate capability should be the same or equivalent to that relied on in the above analysis.

(1) Demonstrate that repair procedures for cold shutdown systems are *developed and material for repairs is maintained on site.*

Enclosure 2

REQUEST FOR ADDITIONAL INFORMATION

1. Section III.G of Appendix R to 10 CFR Part 50 requires cabling for or associated with redundant safe shutdown systems necessary to achieve and maintain hot shutdown conditions be separated by fire barriers having a three-hour fire rating or equivalent protection (See Section III.G.2 of Appendix R). Therefore, if option III.G.3 is chosen for the protection of shutdown capability, cabling required for or associated with the alternative method of hot shutdown for each fire area, must be physically separated by the equivalent of a three-hour rated fire barrier from the fire area.

   In evaluating alternative shutdown methods, associated circuits are circuits that could prevent operation or cause maloperation of the alternative train which is used to achieve and maintain hot shutdown condition due to fire induced hot shorts, open circuits or shorts to ground.

   Safety related and non-safety related cables that are associated with the equipment and cables of the alternative, or dedicated method of shutdown are those that have a separation from the fire area less than that required by Section III.G.2 of Appendix R to 10 CFR 50 and have either (1) a common power source with the alternate shutdown equipment and the power source is not electrically protected from the post-fire shutdown circuit of concern by coordinated circuit breakers, fuses or similar devices, (2) a connection to circuits of equipment whose spurious operation will adversely affect the shutdown capability, e.g., RHR/RCS Isolation Valves, or (3) a common enclosure, e.g., raceway, panel, junction box, with alternative shutdown cables and are not electrically protected from the post-fire shutdown circuits of concern by circuit breakers, fuses or similar devices.

   For each area where an alternative or dedicated shutdown method, in accordance with Section III.G.3 of Appendix R to 10 CFR Part 50, is provided by proposed modifications, the following information is required to demonstrate that associated circuits will not prevent operation or cause maloperation of the alternative or dedicated shutdown method:

A. Provide a table that lists all equipment including instrumentation and support system equipment that are required by the alternative or dedicated method of achieving and maintaining hot shutdown.

B. For each alternative shutdown equipment listed in 1.A above, provide a table that lists the essential cables (instrumentation, control and power) that are located in the fire area.

C. Provide a table that lists safety related and non-safety related cables associated with the equipment and cables constituting the alternative or dedicated method of shutdown that are located in the fire area.

D. Show that fire-induced failures of the cables listed in B and C above will not prevent operation or cause maloperation of the alternative or dedicated shutdown method.

E. For each cable listed in 1.B above, provide detailed electrical schematic drawings that show how each cable is isolated from the fire area.

2. The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

A. Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

B. Identify the device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

C. Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

D. For the areas identified in item 2.C above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

Commissioner Ahearne's Separate Views

I vote to deny the petition, for the following reasons.

This rule has a clear rationale.

On March 22, 1975 a fire occurred at the Browns Ferry nuclear power plant:

> "The fire started in an electrical cable penetration between the cable spreading room and the reactor building; the cable spreading room is located beneath the common control room for Units 1 and 2. The fire burned for about seven hours, spreading horizontally and vertically to all 10 cable trays within the penetration, into the cable spreading room for several feet, and along the cables through the penetration about 40 feet into the reactor building. The fire damage, confined to an area roughly 40 feet by 20 feet in the Unit 1 secondary containment building, affected about 1,600 electrical power and control cables.
>
> While both units were shut down safely, normally used shutdown cooling systems and other components which comprise the emergency core cooling system (ECCS) for Unit 1 were inoperable for several hours.
>
> . . .
>
> The cause of the fire was the ignition of cable penetration sealing material by a candle flame, being used by a construction worker checking for air leaks. The flexible polyurethane foam sealing material being used had not been specifically approved by the licensee's design department, nor had it been tested for this kind of application. The dangers involved in using flammable material in this manner were evidently not recognized by plant management, even though several small fires had occurred during similar testing activities at the plant. Personnel inspecting, sealing and testing the cable penetrations had not been provided with an adequate written procedural guide. Another contributing factor may have been the plant's fire-fighting techniques and equipment."[1]

The Nuclear Regulatory Commission learned a very straightforward lesson:

> "While the Browns Ferry fire will be under scrutiny for some time to come — in all its complexity of causal factors, contributing factors,

[1] U.S. Nuclear Regulatory Commission, Annual Report 1975, pp. 93-94.

real and possible consequences and implications for all nuclear facilities — the event has already demonstrated both the importance and the effectiveness of multiple, mutually reinforcing back-up safety systems, or defense-in-depth. The potential vulnerabilities revealed by the fire will be the subject of intense analysis and will probably result in new requirements both within the industry and the agency."[2]

Unfortunately, final resolution of the fire protection issue was not straightforward. Six years later the industry and the agency are still arguing over what protection is required.

The precise hazards posed by fires are difficult to define: where will fires start, how big will they be, how long will it take to put them out, and what human errors will occur. However, clearly there are problems that must be dealt with. Furthermore, regulators must continuously make decisions even though faced with uncertainty.

Over the past six years the NRC staff did its best. (I would note the industry was of little help.) It has attempted to develop precise criteria, while recognizing it is always difficult to determine that a particular number is *the* correct one. In many cases judgment played a major role. The Commission recognized that there might be room for argument on some of the decisions. For this reason licensees were *invited* to make the case that an alternate configuration would achieve the level of protection that the Commission desired.[3]

The case made by the movants indicates they believe NRC requirements to be invalid unless the NRC can demonstrate unequivocably that its numbers are correct based on reports, studies, or analyses. Their argument seems to be that the NRC has failed to sustain its burden, not that they have presented a better basis for a different number which the NRC has improperly rejected. The regulatory philosophy implicit in this approach is untenable. Certainly the Commission has an obligation to make rational decisions and provide an explanation of how those decisions were reached. However, given the uncertainties, it must be permitted to exercise judgment when that is necessary. Absent a constructive alternative or a major flaw in the underlying rationale, that judgment must be allowed to stand. If not, the result will be large gaps in the regulatory scheme for the many areas in which problems are clear but precise solutions are not.

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[2]*Id.* at 94.

[3]The Commission specifically inserted an exemption request procedure in 50.48(c)(6). If the Director of Nuclear Reactor Regulation determines that the licensee has provided a sound technical basis, then the clock stops on the time limit(s) while the NRC evaluates the merits of the licensee's proposal. This amounts to a *direct* invitation, because Section 50.12(a) of the regulations already provides for exemptions from the requirements of Part 50.

CONCURRING OPINION OF COMMISSIONER BRADFORD

I concur in the Commission's decision and opinion against staying the fire protection rule. However, I want also to set forth a more detailed history of NRC fire protection efforts as of this, the seventh year since the Browns Ferry fire. The motion for stay decided today and the related lawsuit are the latest effort by a minority segment of the nuclear power industry to resist the enhanced fire protection safety standards found necessary by the NRC.

This resistance should not obscure two important facts. First, a potentially disastrous fire in a nuclear power plant is a real threat. It is not hypothetical. It happened in 1975 at Browns Ferry, and less serious fires are commonplace.[1] Second, in many plants an unmitigated fire could disable redundant systems necessary for the safe shutdown of the plant because the licensees have not installed adequate barriers, automatic suppression systems and alternate shutdown systems. It is the hazard posed by these two facts which the NRC has finally attempted to cure by Appendix R. Further delay will further subject the public to an unacceptable and avoidable risk.

The Browns Ferry fire in 1975 started when a candle flame, used to test for air leakage, ignited highly flammable insulation in cable penetrations between the cable spreading room and the Unit 1 reactor building. It was the last in a series of fires at the same plant. The fire spread to the cable spreading room and to the Unit 1 reactor building, where within an area roughly 40 feet by 20 feet, about 1,600 electrical cables were damaged. As a result, control power was lost for equipment such as valves, pumps and blowers. All of the emergency core cooling systems for Unit 1 were rendered inoperable as were parts of the Unit 2 system.

The functioning of the non-energency cooling systems was disrupted significantly during the accident. After the fire began, the main steam isolation valves closed precluding the normal delivery of high pressure cooling water and thereby causing the operators to lower pressure and rely on low pressure pumps. This method of cooling was successful for a few hours until the fire caused four relief valves to close, forcing repressurization. The operator then relied on a control rod drive system pump to provide reactor coolant. Also, during the fire, short circuits in burned cables caused power to be fed backwards causing the failure of control panel indication lights designed to monitor electric power systems during emergencies. Many redundant instruments no longer worked, including all neutron monitoring. Ad hoc repairs in the control room were made in dense

[1]For a list of recent fires, see Attachment A to this opinion. Since NRC does not require the reporting of fires, the list may not be complete.

smoke by craftsmen wearing breather apparatus. The fire burned for seven hours, and it took 13 hours from the beginning of the fire to return to a normal mode of cooling.[2]

After the Browns Ferry fire, the NRC attempted to work with the licensees informally. In late 1976, the NRC staff set October 31, 1980 as the deadline for licensees to complete their fire protection modifications. While this informal approach succeeded with many licensees, some resisted the staff's efforts. Accordingly, in its April 13, 1978 decision on fire protection, the Commission instructed the staff to use its best efforts to maintain current schedules [i.e., October 31, 1980] for implementation of the reactor plant backfits required for fire protection.[3] The staff then sent letters to all licensees in August 1979 reminding them of the Commission's April order and the October 1980 deadline. These exhortations had little effect on some licensees and by May 1980, the recalcitrance of these licensees became a major NRC concern. In its May 23, 1980 order the Commission stated:

> "The staff has completed Safety Analysis Reports concerning fire protection for all operating reactors. The modifications recommended by the staff are not being implemented smoothly. Of utmost concern is the fact that some licensees, four and one-half years after the Browns Ferry fire, are resisting the modifications found necessary by the staff." *In the Matter of Petition for Emergency and Remedial Relief,* 11 NRC 707, 718.

The Commission then noted it had approved Appendix R with a deadline of November 1, 1980 for all items except alternate and dedicated shutdown systems unless the Commission approved extensions. However, the Commission noted that "since the issues involved are well known and have been under discussion for several years, the Commission anticipates approving few, if any, extensions." 11 NRC at 719.

In response to licensee comments on the proposed rule, the Commission subsequently relaxed the November 1980 deadline considerably to allow more time for implementation of fire protection modifications. Even this relaxation has not satisifed that minority of the licensees who have failed to comprehend the 1975 fire and the recent decisions of the Commission starting in 1978 and culminating in the promulgation of Appendix R.

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[2]For a more complete description of the Browns Ferry fire, see Browns Ferry Nuclear Plant Fire, Hearings Before the Joint Committee on Atomic Energy, September 16, 1975; Recommendations Related to Browns Ferry Fire, NUREG 0050, February 1976; Browns Ferry: The Regulatory Failure, Daniel F. Ford, et. al., Union of Concerned Scientists, June 10, 1976.

[3]In the Matter of Petition for Emergency and Remedial Relief, 7 NRC 400, 425.

The fact is that many plants still do not have the fire barriers, automatic suppression systems and alternate shutdown systems which are necessary to protect their shutdown capability. As a result, an unmitigated fire could disable those redundant systems necessary for safe shutdown of the plant.

For example, at Florida Power and Light's Turkey Point Units 3 and 4 the NRC's consultant, Brookhaven National Laboratory, stated, "It is not clear that this plant can sustain a fire and safely shutdown." Among its conclusions were (1) that the licensee had not demonstrated the ability to achieve cold shutdown within 72 hours and (2) that it had failed to demonstrate that fire damage to non-safety circuits would not prevent the operation of shutdown equipment.[4]

At Commonwealth Edison's Dresden Units 2 and 3, Brookhaven found that all fire areas had not been addressed to ensure that a safe shutdown capability would exist after any fire.[5] At Haddam Neck, owned by Northeast Utilities, Brookhaven found that the licensee had not undertaken to demonstrate that at least one method of achieving shutdown would survive any fire in any area. Among its specific conclusions were that the licensee had not shown the post-fire shutdown capability for achieving and maintaining reactor coolant level and reactor coolant pressure. Indeed, Haddam Neck had so many problems, Brookhaven recommended that the unit install a dedicated shutdown system which would be completely independent from the existing cable runs.[6] This has not been done.

The Commission has also discovered that there are problems for plants previously approved according to post-Browns Ferry guidelines. At Salem Units 1 and 2, owned by Public Service Electric and Gas Company, et. al., a special NRC review team found that a single fire could fail all instrument channels, including the independent instrumentation provided for alternative shutdown. The review team concluded that this condition presented an immediate safety concern and required immediate corrective actions.[7]

Finally, the Commission has recently discovered that there are fire protection problems at Browns Ferry itself. The staff found significant violations, including failure to provide automatic sprinkler protection for crucial pumps, failure to adequately protect emergency battery rooms and failure to provide adequate fire watches. The NRC staff found that these violations could have prevented a crucial safety system from performing its intended function under certain conditions. NRC required an immediate 24-hour fire watch at the intake structure and in the emergency battery

[4]Interim Report, Post Fire Shutdown Capability, Turkey Point Units 3 and 4, March 31, 1981.
[5]Interim Report on Post Fire Shutdown Capability, Dresden Station, Units 2 and 3, March 11, 1981.
[6]Interim Report, Post Fire Shutdown Capability, Haddam Neck Plants, April 1, 1981.
[7]See Safety Evaluation Report, Supplement No. 6, Salem Unit 2, May 1981.

room pending TVA modifications. Because these requirements were in the license, the NRC has imposed a $45,000 civil penalty.

The foregoing history is, of course, not the technical justification for Appendix R. As the Commission decision states, the record in the proceeding is sufficient for that purpose. The Browns Ferry experience and the conditions actually existing in the plants demonstrate that a stay of Appendix R would further delay urgently needed fire protection measures in those plants that have thus far declined to adopt some or all of them. Without an enforceable rule, the enhanced protection shown necessary by the 1975 Browns Ferry fire will not be put in place in the remaining plants. This enhanced protection was originally put on a generous and informal 5-year schedule. Appendix R in its current form grants a very substantial extension to that schedule. A further stay would penalize the management of those licensees who have complied, would reward delay, would undercut six years of NRC staff effort, and would expose the public to undue risk. The Commission does well to reject it.

Attachment A

CONCURRING OPINION
OF COMMISSIONER BRADFORD,

Fires at Operating Plants Reported in PN's (From 6/78 - Present)

Date of Event	PN Number	Facility	Description
11/13/78	78-192	Cook - 2	Hydrogen fires in bushing area of electrical generator
11/28/78	78-196	Clinton - 1	Propane gas explosion and fire in drywell - one fatality
12/12/78	78-208	Dresden-3	Fire occurred in the main transformer
1/5/79	79-01	J.C. Summer Unit 1	Minor vandalistic fires at various locations - usually toilets
2/23/79	79-31	Dresden - 3	Fire in main transformer - power transferred immediately to aux. trans.
7/12/79	79-238	TMI - 2	Small fire in radiation monitoring readout panel for aux. bldg. waste gas system

Date of Event	PN Number	Facility	Description
9/4/79	79-388	Indian Point Unit 2	Insulation fire - reactor coolant pump tube oil piping saturated insulation with oil and ignited
9/30/79	I-79-07	H.B. Robinson Unit 2	Lagging fire on cold leg piping caused by lubricating oil leak
10/16/79		Maine Yankee	Fire in diesel generator turbocharger exhaust piping
12/27/79	PNO-II-79-48	Surry - 2	Small portable electrical heater caught fire in the containment of Unit 2
1/21/80	PNO-II-80-13	Browns Ferry Unit 1	Small smoldering fire in a cable tray beneath the turbine building operating floor - was detected and extinguished by plant personnel
2/2/80	PNO-II-80-19	Brunswick Units 1 & 2	A temporary frame building, used for office space, was destroyed by fire
3/7/80	PNO-I-80-36	Calvert Cliffs Unit 2	A small fire occurred in diesel generator room #21 - extinguished within minutes

Date of Event	PN Number	Facility	Description
3/21/80	PNO-III-80-58	Midland	Fire occurred in two trailers used by personnel designing small bore piping
4/21/80	PNO-TMI-80-26	TMI - 1 & 2	Fire broke out at a trailer next to TMI observation center
4/22/80	PNO-I-80-61	Nine Mile Point Unit 1	A fire, which resulted from lube oil which had leaked from a main turbine shaft driven feed water pump, was detected
6/3/80	PNO-II-80-98	Surry - 1	A fire in an instrument bus voltage transformer disabled one of four instrument buses causing unit to trip from 100% power
10/5/80	PNO-III-80-180	Quad-Cities - 2	Small oil fires caused by oil on the hot main stream lines
11/6/80	PNO-TMI-	TMI - 2	Fire in site trailer
12/9/80	PNO-TMI-80-54	TMI - 1	Fire in reactor building (containment) sump; extinguished within minutes

Date of Event	PN Number	Facility	Description
12/15/80	PNO-III-80-230	D. C. Cook - 2	Reactor tripped due to fire in the generator pilot excitor unit
3/7/81	PNO-II-81-22	Watts Bar - Units 1 & 2	Fire in a digital computer system in communications room of the facilities control room

Significant Fires at Plants Occurring Prior to 6/78

Date	Facility	Description
4/13/67	Peach Bottom, Unit 1	A fire occurred in the insulation of pipes from the helium cooling system located in the basement of the containment building. Fire occurred while the plant was shutdown for modifications.
3/12/68	San Onofre	A fire occurred in and was confined to three overhead cable trays stacked one above the other in the 480V switchgear room. The fire was caused by overloaded and inadequately ventilated electrical cables supplying the pressurizer heaters. The reactor was at power when the fire occurred.
11/4/71	Indian Point, Unit 2	A fire occurred in the Primary Auxiliary Building and was concentrated in a small construction shack inside the building. The construction shack and its contents were destroyed. Also, three motor control centers, which controlled ESF equipment and were located above the shack, were badly damaged. The fire was caused by arson.

Cite as 13 NRC 838 (1981) **CLI-81-12**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **Docket No. 50-389 CP**

FLORIDA POWER & LIGHT COMPANY
(St. Lucie Nuclear Power Plant, Unit No. 2) **June 15, 1981**

Upon review of certain generic issues raised by ALAB-603, 12 NRC 30 (1980), regarding the loss of all AC power at the St. Lucie facility, the Commission concludes that ALAB-603 neither (1) establishes generic guidelines for determining the design basis events to be used for plant design and operation nor (2) designates station blackout as a design basis event as that term is used by the regulatory staff.

NRC: AUTHORITY (RECONSIDERATION)

Reconsideration is a well-recognized power inherent in the Commission's authority to decide in the first instance. *Albertson* v. *FCC*, 182 F.2d 397, 399 (D.C. Cir. 1950), *Trujillo* v. *General Electric Co.*, 621 F.2d 1084, 1086 (10th Cir. 1980).

NRC: AUTHORITY (RECONSIDERATION)

As long as the Commission retains jurisdiction it can reconsider an earlier decision not to review an Appeal Board decision. The Commission retains jurisdiction over a final decision only for the sixty days in which a party may seek judicial review under the Hobbs Act. *American Farm Lines* v. *Blackball Freight*, 397 U.S. 523, 540 (1970); *Pan American Petroleum Corp.* v. *FPC*, 322 F.2d 999, 1004 (D.C. Cir. 1963).

LICENSING BOARDS: CONSIDERATION OF GENERIC ISSUES (SAFETY)

In individual licensing proceedings involving unresolved generic safety issues, the regulatory staff must provide the Licensing Board with evidence explaining why resolution of those issues can be deferred. *Gulf States Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977).

ADJUDICATORY BOARDS: DELEGATED AUTHORITY

The pendency of a Commission plan for the development and articulation of safety objectives for nuclear power, 45 Fed. Reg. 71023 (October 27, 1980), should not inhibit adjudicatory boards from examining closely any accident sequence which in their judgment poses an unacceptable risk to the public health and safety. Probabilistic or numerical calculations may be used in such examinations and the boards have a responsibility to mandate whatever mitigative actions they deem necessary to protect adequately the public health and safety when such actions are supported by the record.

MEMORANDUM AND ORDER

Introduction

This decision completes the Commission's review of certain issues raised by the decision of the Atomic Safety and Licensing Appeal Board, ALAB-603, 12 NRC 30 (1980), regarding designation of the loss of all AC power as a design basis event. Because of the generic nature of these issues, the Commission invited briefs from persons other than the parties. Briefs on the issues were received from the NRC staff and the Atomic Industrial Forum Committee on Reactor Licensing and Safety.[1] In addition, a Memorandum

[1]Applicant Florida Power and Light Company contends that the Commission has no jurisdiction to review ALAB-603 because the review time provided by the rules expired before the Commission exercised its review authority. 10 CFR 2.786(a). Applicant's jurisdictional argument is incorrect because the Commission's rules do not explicitly address reconsideration. Reconsideration is a well-recognized power inherent in the Commission's authority to decide in the first instance. *Albertson* v. *FCC*, 182 F.2d 397, 399 (D.C. Cir. 1950), *Trujillo* v. *General Electric Company*, 621 F.2d 1084, 1086 (10th Cir. 1980). Thus as long as the Commission retains jurisdiction it can reconsider an earlier decision not to review an Appeal Board decision. Moreover, reconsideration does not disturb finality significantly because the Commission retains jurisdiction over a final decision only for the sixty days in which to seek judicial review under the Hobbs Act. *American Farm Lines* v. *Blackball Freight*, 397 U.S. 523, 540 (1979), *Pan American Petroleum Corp.* v. *Federal Power Comm.*, 322 F.2d 99, 1004 (D.C. Cir. 1963).

was issued on December 27, 1980 by two of the Administrative Judges who were members of the Appeal Board which decided ALAB-603. The Commission has determined that these filings fully present the issues and that oral argument would not aid our deliberations. For the reasons discussed below, the Commission has determined that ALAB-603 does not establish generic guidelines for determining the design basis events to be used for plant design and operation and does not establish station blackout as a design basis event as that term is used by the staff.

Events Leading To Review

A. Decision Below

In ALAB-603, the Appeal Board concluded its consideration of the adequacy of electric power systems for Unit 2 of the St. Lucie nuclear power plant. The Appeal Board finding relevant to this review was that the probability of total loss of on-site and off-site AC power — station blackout was sufficiently high that protecting the plant against such an occurrence was warranted. Specifically, the Board found that the probability of occurrence of station blackout was in the range of one chance in ten thousand to one chance in one hundred thousand (10E-4 to 10E-5) per year which the Board noted is significantly higher than the threshold values in Section 2.2.3 of the Standard Review Plan (SRP) at which the staff requires analysis of the implications for plant integrity of certain off-site man-made hazards.[2] Consequently, the Board designated station blackout a design basis event for St. Lucie Unit 2 and directed that the applicant's Final Safety Analysis Report include: (1) an analysis demonstrating the plant's ability to operate through such an event; and (2) a detailed training program for station operation during a blackout transient and for the restoration of AC power.[3]

B. Commission Action

No party petitioned the Commission for review of ALAB-603; and on October 14, 1980 the time expired for Commission *sua sponte* review of that decision. The staff has been reviewing the generic issue of station blackout since 1977 under Task Action Plan A-44. On November 10, 1980 the Director, Nuclear Reactor Regulation (NRR), responded to the Chairman's

[2]The SRP threshold values for the probability of occurrence of initiating events leading to exposures in excess of 10 CFR Part 100 guidelines are one chance in one million (10E-6) per year for a conservatively calculated probability of occurrence and one chance in ten million (10E-7) per year for a realistically calculated probability of occurrence.

[3]These conditions were included in the construction permit for St. Lucie Unit 2 by an amendment issued on September 18, 1980.

request for further information on the status of Task Action Plan A-44 - Station Blackout (TAP A-44). That response included a memorandum from the Director, Division of Systems and Reliability Research, to the Director, NRR, which alerted the Commission to certain staff positions which had not been presented in the staff's filings before the Appeal Board and the Commission. These staff positions raised important generic issues regarding the impact of the Appeal Board's decision on the regulatory process. As a result, on December 22, 1980 the Commission decided to reconsider its previous determination not to review ALAB-603.[4] Upon reconsideration, the Commission affirmed the license amendments which the Appeal Board ordered for the St. Lucie Unit 2 construction permit but took review on the following generic issues:

(1) What are the generic implications of using the threshold probabilities in Section 2.2.3 of the Standard Review Plan as guidelines in determining the design basis events to be used for plant design and operation?
(2) Granting the need for protective measures against loss of all AC power for some reasonable period of time, is designation of station blackout as a design basis event the appropriate regulatory framework in which to consider such measures pending completion of the staff generic study TAP A-44?

Positions On The Issues

A. Use of Threshold Probabilities in the Standard Review Plan as Guidelines for Determining Design Basis Events

The Administrative Judges state that ALAB-603 does not present this issue because the decision to consider station blackout as a design basis event for St. Lucie Unit 2 was based on an independent judgment of the probability of occurrence of that event. They explain that threshold probability values in the Standard Review Plan were looked to only for perspective and guidance.

Staff believes that the Administrative Judge's clarification of ALAB-603 clearly shows that it should not be interpreted to mandate use of the threshold probabilities in Section 2.2.3 of the Standard Review Plan as guidelines for determining design basis events. Moreover, staff contends that if those probabilities were to be used for this purpose, such use would have a severe impact on the regulatory process because there are a large number of accident sequences with an estimated probability of occurrence exceeding one in ten million per reactor year and which could produce or

[4]CLI-80-41 (1980).

result in core melt or severe core damage. Allocation of staff resources to evaluate these sequences would require substantial additional staff personnel. If personnel were provided for this purpose by diverting staff resources from other activities, staff believes that the result could be an increase in risk to public health and safety.

The Atomic Industrial Forum (AIF) believes that the Appeal Board's use of the Standard Review Plan values as decision criteria misinterpreted staff's intent regarding the use of those values. In AIF's view, staff intended those values to be used as screening criteria for excluding consideration of accidents involving the presence or use of hazardous materials in the vicinity of a plant. Those values were not intended to be used to determine the need to design against accident sequences like station blackout. Moreover, the use of those values as decision criteria would result in the incorporation of measures to prevent or mitigate the effects of many accident sequences which AIF believes are insignificant contributors to reactor risk.

AIF also notes that the Commission has initiated a proceeding to establish quantitative safety goals.[5] AIF believes that this proceeding provides the appropriate vehicle for establishing probabilities to be used in decisions regarding the need for additional protective measures in plant design and operation.

B. Designation of Station Blackout as a Design Basis Event

Staff contends that the Appeal Board used the term design basis event only to denote those events whose consequences require mitigation to protect public health and safety. On the basis of this interpretation, staff believes that the Appeal Board's imposition of mitigative conditions was appropriate because it was consistent with the logic in *River Bend*[6] as applied to the unresolved generic safety issue in TAP-A-44, Station Blackout. Accordingly, staff believes that it was appropriate for the Appeal Board to designate station blackout a design basis event for St. Lucie Unit 2

The AIF believes that any decision to designate station blackout as a design basis event should be based on comprehensive probabilistic risk assessments that include a comparison of the risk from this event with the risk from other events. Moreover, AIF appears to suggest that additional measures to reduce the risk associated with station blackout should be

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[5]45 Fed. Reg. 71023 (October 27, 1980).

[6]*Gulf State Utilities Co.* (River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977). In *River Bend*, the Appeal Board held that in individual licensing proceedings involving unresolved generic safety issues, staff must provide the Licensing Board with evidence explaining why resolution of those issues can be deferred.

considered only if the overall risk from all accident sequences exceeds a predetermined quantitative safety goal.

Decision

A. Use of Threshold Probabilities in the Standard Review Plan as Guidelines for Determining Design Basis Events

Section 2.2.3 of the Standard Review Plan establishes numerical thresholds for the probability of occurrence of certain events which the staff considers in evaluating the design of a plant. Those events are limited to potential accidents resulting from the presence of hazardous materials or activities in the vicinity of the plant. Staff considers such an event if a realistic calculation of the expected rate of occurrence of potential exposure in excess of Part 100 Guidelines results in a value exceeding one part in ten million per year (or a conservative calculation results in a value exceeding one part in a million per year).

The Appeal Board, in ALAB-603, explicitly recognized the narrow applicability of the threshold values contained in Section 2.2.3 of the Standard Review Plan. The Board looked to these values as guidelines, not as established requirements for identifying potential accidents requiring additional consideration. Moreover, as two members[7] of the Board stated in their subsequent memorandum of December 22, 1980, the Board's treatment of station blackout was based on its independent assessment of the probability of the event for St. Lucie Unit 2 as established by the evidentiary record. Thus, in our view, ALAB-603 does not establish any single numerical threshold for the mandatory consideration of accident sequences. The Appeal Board found, as a matter of judgment, that the probability of station blackout at St. Lucie was high enough to warrant additional measures to protect the public health and safety. That judgment was based on the entire record of the St. Lucie proceeding. Under these circumstances, the probability values calculated for that particular event should not be interpreted as establishing a generic numerical threshold to be used for future consideration of accident sequences.

The Commission has adopted a plan for the development and articulation of safety objectives for nuclear power. "Plan for Developing a Safety Goal," 45 Fed. Reg. 71023 (October 27, 1980). This effort should provide the context for resolving the generic issue of a numerical threshold for the analysis of accident sequences. However, the pendency of the safety goal matter should not inhibit the boards from examining closely any accident

[7]The third Board member is no longer with the Commission.

sequence which in their judgment poses an unacceptable risk to the public health and safety. Probabilistic or numerical calculations may be used in such an examination and boards have a responsibility to mandate whatever mitigative actions they deem necessary to protect adequately the public health and safety when such actions are supported by the record.

B. Designation of Station Blackout as a Design Basis Event

The term "design basis event" is not defined in the regulations. However, staff's licensing review of a nuclear power plant includes an analysis of the plant's responses to certain postulated accidents referred to as design basis events. These accident scenarios are chosen on the basis of staff's engineering judgment and are not necessarily identified as design basis accidents from a calculation of their probability of occurrence. In ALAB-603, the Appeal Board did not use the term design basis event as it has been used by the staff. Rather, the Appeal Board used that term in a more general sense to denote an event which posed an unacceptably high risk to the public health and safety unless preventive or mitigative measures were taken. There is no indication in ALAB-603 that the Appeal Board intended to go further and subject station blackout to the regulatory regime established by the staff for considering design basis events. Thus, the Appeal Board's use of this phrase was, as they have indicated, as a label for the purpose of expressing its judgment that additional measures were required at St. Lucie to deal with the possibility of station blackout either by lowering the probability or by mitigating the consequences.

Conclusion

For the reasons discussed above, the Commission finds that ALAB-603 does not establish any generic guidelines for determining the design basis events to be used for plant design and operation and does not establish station blackout as a design basis event as that term is used by the staff.

The separate views of Commissioner Gilinsky and additional views of Commissioner Ahearne are attached.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
this 15th day of June 1981.

Additional Views of Commissioner Ahearne

I concur in the Commission's opinion. I note that its practical effect is to instruct the boards that the Appeal Board decision does not establish that a particular event or sequence of events automatically requires further consideration whenever the probability of occurrence exceeds some numerical threshold.

Commissioner Gilinsky's Separate Opinion

The Appeal Board acted correctly in independently evaluating the risks posed by station blackout at St. Lucie and in requiring that steps be taken to prevent or mitigate the consequences of such events.

Had the Appeal Board included in its decision the reasoning presented in its memorandum of December 22, Commission review of this case would have been unnecessary. It is now clear that the Appeal Board did not intend to accord the rough probability guidelines used by the staff in certain safety reviews a more formal status. The one-chance-in-a-million-per-year threshold used by the staff in its reviews[1] is not a precise tool for determining which events outside the plant are so probable that preventive or mitigative measures must be taken. It has not been approved by the Commission.

Unfortunately, the Commission opinion goes beyond these findings and introduces unnecessary elements of uncertainty concerning which possible accidents need to be dealt with in the licensing process. A rational safety review process assumes a uniform threshold of safety significance for possible events which need to be protected against. (The commonly used measure of safety significance is probability times consequences.) By rejecting as a threshold for such review and action not only the one-chance-in-million-per-year used by the staff, but also the higher estimate used by the Appeal Board for the probability of station blackout, and putting nothing in their place but the observation that the Board's judgment "was based on the entire record of the St. Lucie proceeding," the opinion makes NRC's choice of accidents which must be analyzed and protected against seem almost capricious. The Commission should acknowledge what would seem to be implicit in its decision, that events of safety significance (though not necessarily of probability) comparable to, or greater than, station blackout at St. Lucie should be analyzed to determine whether preventive or mitigative actions are required.

As a final matter, the term "design basis accident" is not defined in the Commission's regulations or in any other Commission document. It is not enough to speculate on what the staff or the boards mean when they say that something is or is not a "design basis accident." If the Commission is to use the term it ought to define it.

[1]Section 2.2.3 of the Standard Review Plan used by the staff provides that when certain events occurring off-site have a conservatively calculated probability of occurrence of one-chance-in-a-million-per-year, or a realistically calculated probability of occurrence of one-chance-in-ten-million-per-year, the implications for plant integrity of these events must be analyzed.

Cite as 13 NRC 847 (1981) **CLI-81-13**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of

PETITION OF SUNFLOWER COALITION **June 24, 1981**

The Commission denies a petition seeking to terminate or suspend the Agreement State program with the State of Colorado for alleged violation of the Uranium Mill Tailings Radiation Control Act and failure to protect the public health and safety.

ATOMIC ENERGY ACT: COOPERATION WITH STATES

Section 274 of the Atomic Energy Act permits the Commission to enter into Agreements authorizing the States to regulate source materials, byproduct materials, and small quantities of special nuclear materials if it finds that the state program is in accordance with subsection o, compatible with the Commission's program, and adequate to protect the public health and safety. 42 USC § 2021(o).

ATOMIC ENERGY ACT: COOPERATION WITH STATES (TERMINATION)

The Commission retains the authority to terminate or suspend a State Agreement and reassert its licensing and regulatory authority should the Commission find that (1) such termination or suspension is required to protect the public health and safety, or (2) the State has not complied with one or more requirements of Section 274. 42 USC § 2021(j).

ATOMIC ENERGY ACT: COOPERATION WITH STATES (URANIUM MILLS AND MILL TAILINGS)

The Uranium Mill Tailings Radiation Control Act (UMTRCA) gives the Commission direct regulatory authority over uranium mill tailings by adding them to the definition of byproduct material in Section 11e of the Atomic Energy Act, 42 USC 2014(e). Agreement States, however, may continue to regulate mill tailings if they comply with certain requirements of UMTRCA. An Agreement State's licensing and regulatory standards for uranium mills and mill tailings must, to the extent practicable, be at least as stringent as the Commission's standards. Pub. L. No. 95-604, Section 204(e)(1); 92 Stat. 3037; 42 USC § 2021(o)(3).

ATOMIC ENERGY ACT: COOPERATION WITH STATES (URANIUM MILLS AND MILL TAILINGS)

Agreement States must be in compliance with the requirements of UMTRCA by November 8, 1981. In the interim, States are to exercise their licensing authority over byproduct material in a manner which, to the extent practicable, is consistent with the requirements of Section 274(o) of the Atomic Energy Act; and the Commission is authorized to ensure that States implement Section 274(o) to the extent practicable. Pub. L. No. 96-106, Section 22; 93 Stat. 799.

ATOMIC ENERGY ACT: COOPERATION WITH STATES (URANIUM MILLS AND MILL TAILINGS)

Section 274(o) of the Atomic Energy Act requires States to have procedures for licensing and regulation of uranium mills and mill tailings which include public hearings and written environmental analyses. Pub. L. No. 95-604, Section 204(e)(1); 92 Stat. 3037; 42 USC § 2021(o)(3). Such public hearings need not be full, formal adjudicatory proceedings; and the environmental analyses need not mirror Federal NEPA procedures.

ATOMIC ENERGY ACT: COOPERATION WITH STATES (TERMINATION)

A State Agreement is not to be permanently terminated or revoked for technical failures to comply with Section 274 or for single incidents of State inaction, but only in exceptional circumstances. Rather, the NRC is to cooperate with Agreement States and obtain their compliance through its review procedures. The power to terminate the Agreement is to be one of last resort where all others fail.

ATOMIC ENERGY ACT: COOPERATION WITH STATES (TERMINATION)

Before the Commission may terminate a State Agreement, it must first provide the State with notice and opportunity for a hearing; and, before it can justify instituting proceedings to terminate pursuant to Section 274(j), it must have some reason to believe that such termination may be necessary to protect the public health and safety or to secure the State's compliance with Section 274.

MEMORANDUM AND ORDER

On May 26, 1981, petitioner Sunflower Coalition (Sunflower) filed with the Commission a petition claiming that the Agreement State program with Colorado should be terminated or suspended for Colorado's failure to protect the public health and safety, and that the Commission and Colorado have violated the Uranium Mill Tailings Radiation Control Act (UMTRCA), P.L. 95-604, by not enforcing its requirements (principally public hearing and environmental analyses) to the extent practicable prior to November 8, 1981. Sunflower also claims that the Commission violated UMTRCA in not specifically making a finding as to the practicability of Colorado's meeting those requirements. Sunflower filed a lawsuit on January 19, 1981, in the Federal District Court, District of Colorado against the NRC, the State of Colorado and various state officials raising these same claims. Since the petition specifically refers to Sunflower's complaint in the District Court, for purposes of considering Sunflower's petition on the merits, the Commission will treat the petition as if the allegations of the complaint are a part of the petition.[1]

STATUTORY FRAMEWORK

A discussion of the statutory framework of this case is necessary to understand the allegations of Sunflower's petition. Section 274 of the Atomic Energy Act permits the Commission to cede to the States

[1] Because of the short court-imposed time deadline for Commission action and because Sunflower's petition has been addressed directly to the Commission, we will, in these circumstances, consider the merits of the petition ourselves, treating it as a 10 CFR 2.206 petition, rather than referring it to a staff office for action under 10 CFR 2.206 and 10 CFR .40.

jurisdiction to regulate source materials, byproduct materials and smal quantities of special nuclear materials. The procedures and criteria fo: doing so are carefully spelled out in the statute. The Commission i: authorized to cede its jurisdiction over nuclear materials if it finds that th(state program is "in accordance with subsection o", "compatible with th(Commission's program" and "adequate to protect the public health an(safety." Under a State Agreement, the Commission relinquishes a portioı of what had earlier been the exclusive pre-emptive Federal jurisdiction ove nuclear materials, and the State assumes jurisdiction.

The Atomic Energy Commission and the State of Colorado entered int(such an agreement on January 16, 1968, effective February 1, 1968. *See* 3: *Fed. Reg.* 2400 (January 31, 1968). It was signed after the Governor certifie(the State's desire to assume regulatory responsibility and after th(Commission determined, as then required by the Atomic Energy Act, tha Colorado's program for the control of radiation hazards was compatibl(with the Commission's program for the regulation of such materials an(was adequate to protect the public health and safety. 42 U.S.C. 2021(d]

The NRC retains the authority to terminate or suspend the Agreemen with Colorado and reassert licensing and regulatory authority should th(Commission find that "(1) such termination or suspension is required t(protect the public health and safety, or (2) the State has not complied witl one or more requirements of this section." 42 U.S.C. 2021j; Section 274j o the AEA. The NRC must give the State reasonable notice and aı opportunity for a hearing before terminating or suspending all or part of th Agreement.[2] *Ibid.* The NRC is also required periodically to review th Agreement and the actions taken by the State under the Agreement t(ensure that the State complies with the provisions of Section 274. *Ibic*

Prior to the passage of the Uranium Mill Tailings Radiation Control Ac of 1978 (UMTRCA), the regulation of uranium mill tailings was a Stat responsibility pursuant to its inherent police power, whether or not th State had entered into an Agreement with the Commission. An Agreemen with the Commission was of no relevance for this purpose because th Commission had no regulatory authority over mill tailings to relinquish to State.

The passage of UMTRCA changed this legal structure. The Act add uranium mill tailings to the definition of byproduct material in 42 U.S.C 2014(c) and by so doing gives the Commission direct regulatory authorit over those mill wastes. UMTRCA also amends section 274 of the AEA t(

[2]The NRC may temporarily suspend the Agreement without notice and hearing in a emergency situation. 42 U.S.C. 2021j(2), Pub. L. 96-295 Section 25, 94 Stat. 787. This provisio was added to the Act in June 1980.

provide that the Agreement States may continue to regulate mill tailings if they comply with certain requirements of UMTRCA. The State's licensing and regulatory standards for uranium mills and mill tailings must, to the extent practicable, be at least as stringent as the Federal standards. Pub. L. 95-604, Section 204(e)(1); 92 Stat. 3037; 42 U.S.C. 2021(o)(2), Section 274o of the AEA. In addition, the State must require procedures which include public hearings and written environmental analyses on licensing actions. Pub. L. 95-604, Section 204(e)(1); 92 Stat. 3037; 42 U.S.C. 2021(o)(3); Section 274o(3) of the AEA.

As originally enacted on November 8, 1978, UMTRCA provided that the States were not obligated to be in compliance with its requirements until November 8, 1981. However, the Act was unclear whether the Commission and the States had concurrent licensing jurisdiction in this three-year interim period. Consequently, in November, 1979 Congress amended UMTRCA to clarify that it intended Agreement States to license in the interim but that, during that transition period, the States were to exercise their authority over byproduct material "in a manner which, to the extent practicable, is consistent with the requirements of section 274o of the Atomic Energy Act ...." Pub. L. 96-106, Section 22; 93 Stat. 799. The 1979 amendment also gave the NRC authority to ensure that section 274o of the AEA is implemented by a State "to the extent practicable during the three year period" between November 8, 1978 and November 8, 1981. *Ibid.*

The Sunflower Petition

Sunflower's first claim is that the NRC has disregarded the requirement of UMTRCA that prior to November 8, 1981 Colorado exercise, and the NRC ensure that Colorado exercise, its authority in a manner which is, to the extent practicable, consistent with section 274(o) of the AEA. Sunflower claims that Colorado has proceeded with the regulation and licensing of several uranium mills without requiring a written analysis of environmental impact and without holding public hearings involving the environmental analyses as required by 274(o)(3)(A) and (C). Further, Sunflower states that the NRC should have made a determination of the impracticability of these requirements at the time of the NRC's review of the State program. If no such determination was made, NRC should "declare that any licensing or other action taken by Colorado without compliance with the subsection (o) requirements not then determined to be impracticable is invalid and should assert NRC jurisdiction over those matters notwithstanding Colorado's agreement." Petition of Sunflower Coalition, p. 3.

In making this argument, Sunflower clearly misinterprets the 1979 clarifying amendment to UMTRCA which governs effectiveness of

UMTRCA during the interim period until November 8, 1981. That amendment does not require the NRC to make a specific, formal determination that implementation of any of the requirements of section 274(o) is impracticable. The amendment states only that the NRC shall have authority to ensure implementation of the requirements; it does not specify a means of doing so.

Since the passage of UMTRCA the NRC has been working closely with all Agreement States, including Colorado, to bring the states' regulatory programs into compliance with UMTRCA as quickly as possible. The Office of State Programs (OSP) and the Nuclear Material Safety and Safeguards (NMSS) division have an ongoing exchange of information and technical assistance program with the Agreement States. These offices are in almost constant communication with the states providing them with information, reviewing their programs, reviewing individual licenses or environmental analyses, participating in public hearings, and providing states with technical assistance. The NRC's approach to implementing UMTRCA has been to encourage and aid the states, in a relatively informal manner, to comply. This approach has been found to be more flexible and more effective than a formal approach such as that suggested by Sunflower would be.

This informal approach is supported by the legislative history of the 1979 amendment. The amendment as orginally proposed by Senator Domenici in the Senate only preserved the authority of the states to regulate mill tailings in the three-year interim period; it had no "to the extent practicable" requirement:

> On or before the date three years after the date of enactment of this Act, notwithstanding any amendment made by this title, any State may exercise any authority under State law respecting byproduct material, as defined in section 11e.(2) of the Atomic Energy Act of 1954, in the same manner, and to the same extent, as permitted before the enactment of this Act.

In the House, Representative Udall added the "extent practicable" language to the amendment:

> My amendment changes the Senate Amendment to provide that specified standards and procedures in the act would be effective immediately to the maximum extent practicable. Further the Nuclear Regulatory Commission is declared to retain authority to assure that agreement States in exercising their licensing activities implement the requirements to the maximum extent practicable. Cong. Rec. H 9772 (October 26, 1979).

The intent of Representative Udall based on the language of the amendments was to make the Agreement States apply the requirements of UMTRCA to the extent practicable in the interim period. The role of the NRC is not as clear from the language, however. Senator Domenici indicated in the discussions on the floor of the Senate that the NRC was to be a mentor, rather than a policeman:

> With regard to NRC's role during the interim period, this amendment would not give the Commission any new authority over individual State licensing determinations. Thus, the Commission would not have the authority to issue, deny or revoke licenses in the Agreement States. Rather, NRC's role during the interim period will be to assist the Agreement States in upgrading their regulatory programs to meet the new requirements. One approach which has already been used effectively in this process is the offer of NRC technical assistance to the State on a consultant basis. Another useful tool is the grant program established by the Mill Tailings Act. The amendment provides an added directive to the Commission to work in cooperation with the Agreement States to improve the effectiveness of their regulatory efforts as soon as possible. Cong. Rec. S 14356 (October 29, 1979)

This is precisely what the NRC has done. Through its information exchange and technical assistance program with Colorado, the OSP and NMSS have kept abreast of the implementation of UMTRCA in Colorado. In addition, the State has submitted to the OSP voluminous material which indicates that Colorado has indeed complied with those requirements of section 274o which Sunflower alleges have been neglected — i.e., 274o(3)(A), (C), and (D).

After review of the state's program, the NRC sent to Colorado two "report cards" on July 25, 1980 and on February 20, 1981, informing the State of what steps Colorado must take for the state to be in compliance with UMTRCA so that the NRC and Colorado may enter into an amended agreement prior to November 8, 1981. In addition, the OSP sent to all agreement states on July 11, 1980, a letter stating that it was not only necessary, but feasible, for states to prepare written analyses of the environmental impact of licensed activities during the interim period. There have also been numerous written and oral contacts with Colorado concerning implementation of requirements of UMTRCA.

Since UMTRCA was amended in 1979, only two Colorado licensing proceedings have progressed to the public hearing stage. In both cases, Colorado has complied with the requirements of Section 274(o)(3)(A) and

(C). These two licensing actions were the Cotter Corporation mill at Canon City, Colorado and the Homestake Pitch Project in Saguache County, Colorado.

The Cotter Corporation Canon City site has been an active milling site since it was licensed by the AEC in 1957. The most recent licensing action concerning that site concerns the construction of a new impoundment system to hold both past and future tailings and the construction of an expanded milling operation. The Colorado Department of Health (CDH) required the Cotter Corporation to submit an environmental report and supplements and extensive engineering studies. The company began constructing the mill in 1977 (before the passage of UMTRCA) and had completed 50% of the construction in April 1978 when Colorado adopted regulations requiring authorization for prelicensing construction. These new regulations provided for exemptions, one of which was granted to Cotter, and Cotter resumed construction.

Public hearings were held in May 1979 at Canon City, Colorado. Public notice of these hearings was provided in advance as well as notice of availability of the draft executive summary. The draft executive summary contains a summary of the licensing review and the written environmental analysis of the CDH staff among other things. In addition, the NRC prepared a detailed, written environmental assessment at the request of the state. At the hearings, the state permitted submission of both written and oral comments and permitted members of the audience to ask questions of panel members. Several members of the NRC staff participated in the hearing. After reviewing the comments and testimony at the public hearings, and all other comments submitted, the CDH issued a Final Executive Licensing Review Summary (FELRS) which summarized CDH's environmental and safety review process and assessment. In the review process and in the Executive Licensing Summaries, Colorado considered the factors listed in Section 274(o)(c)(i)-(iv). Even though this licensing action and construction of the new impoundments and mill had begun prior to UMTRCA and the 1979 amendment to UMTRCA, Colorado applied the requirements of Section 274o to the proceeding as much as was practicable under the circumstances.

The Homestake Mining Company's Pitch Project is the only other licensing action to reach the hearing stage. Homestake applied in 1976 to build a conventional uranium mill next to an existing mining operation. CDH required Homestake to submit an environmental report and supplement. In 1977, the U. S. Forest Service, on whose land the site is located, determined that an Environmental Impact Statement (EIS) must be prepared. The draft EIS was prepared by the Forest Service, the NRC and the State of Colorado and was issued in July 1978. The final EIS (FEIS) was issued in April 1979. CDH also required additional geotechnical, hydrology and engineering studies on the tailings disposal proposals. CDH's Preliminary Executive Licensing Review Summary (PELRS) was then published on November 10, 1980. Notice was also given that individuals or groups could apply for status as a party and make a presentation at the hearing.

The hearing was held on December 10 - 12, 1980 with a hearing panel and three parties — the Colorado Open Space Council, the Gunnison Valley Alliance and an individual party. All participants were sworn in, and the parties could cross-examine them. Members of the NRC staff were present at the hearing. Allowance was also made for general public participation in the hearing; and members of the general public were permitted to make statements to the hearing panel, but were not permitted to cross-examine participants.

The FELRS was published and a license issued for the Homestake Pitch Project in March 1981. The state considered in its review and in the PELRS, the EIS and the FELRS all of the factors listed in section 274o(C)(i)-(iv).

These, then, are the only two licensing actions in Colorado to reach the hearing stage. In its other pending actions, the state has been preparing to comply with the hearing and environmental analysis portions of UMTRCA when the time comes. In addition, the NRC has been working closely with the state to provide environmental assessments and/or other technical assistance in many cases.

Colorado has to the best of NRC's knowledge prohibited major construction activity prior to complying with the provisions of 274o(C) as required by 274o(D). The only exception of which NRC is aware is the Cotter case in which construction of the mill began before UMTRCA and construction of the impoundment began before the 1979 clarifying amendments were passed by Congress. Even in that instance Colorado complied with 274o(C) before the site was licensed for operation.

Colorado's procedures are, therefore, sufficient to comply with the requirements of section 274o of the AEA.[3] Sunflower cites no specific incident of failure by Colorado to prepare an environmental analysis, to hold a public hearing or to otherwise comply with sections 274(o)(3)(A), (C), or (D). In light of OSP's review of the information provided by the State of Colorado and the information obtained in that office's exchange of information program with the State, the Commission sees no basis for an

[3]The legislative history is clear that the public hearing need not be a full, formal adjudicatory proceeding; the environmental analyses need not mirror Federal NEPA procedures. *See* Vol. 124, No. 168 - Part II Cong. Rec. H 12968 (Oct. 14, 1978) (daily ed.) (remarks of Congressman Dingell); Vol. 124, No. 167 - Part II Cong. Rec. S 18750 (Oct. 13, 1978) (daily ed.) (remarks of Senator Wallop).

order to initiate proceedings pursuant to section 274j of the AEA to terminate or suspend all or part of the agreement with Colorado, or to otherwise divest *Colorado of its authority over* uranium mill tailings.[4]

Alleged Deficiencies in Colorado Program

Sunflower's second claim is that in its periodic reviews of Colorado's radiation control program the NRC has repeatedly identified several shortcomings in the Colorado program; yet, NRC has not acted to suspend or revoke Colorado's Agreement. The deficiencies enumerated in Sunflower's Complaint in the district court are:

a) severely inadequate personnel staffing, particularly with regard to uranium mill licensing and compliance, resulting in an increasing backlog of overdue inspection and enforcement work, and inadequate protection of the public health and safety;
b) inadequate laboratory facilities and data handling facilities, resulting in inefficiency and errors detrimental to the public health and safety; and
c) the lack of legal authority to impose an appropriate and necessary range of civil penalties or other sanctions for violations of radiation protection regulations in order to protect the public health and safety.

Sunflower also asserts that these alleged deficiencies in Colorado's program have resulted in serious incidents of failure by Colorado to protect the public health and safety — specifically, incidents at the site operated by Cotter Corporation near Canon City, Colorado, at the site operated by Union Carbide Corporation near Uravan, Colorado and at the site operated by Sweeney Mining and Milling Corporation near Boulder, Colorado.

1. Evaluation of State Programs

The AEC/NRC has performed thirteen periodic onsite reviews of the Colorado radiation control program since Colorado became an Agreement State on February 1, 1968. In each year since 1968 the Colorado agreement program has been found to be adequate to protect the public health and safety and compatible with the AEC/NRC program. The conclusions concerning adequacy and compatibility are based on (1) the onsite reviews during which the NRC staff reviews various aspects of the State program

[4]Since Colorado like all agreement states is regulating mill tailings until November 8, 1981 pursuant to its police powers rather than through agreement with the Commission, Commission action to enforce the practicability provision of the 1979 amendment to UMTRCA might take a form other than a hearing on the agreement.

utilizing the NRC "Guide for Evaluation of Agreement State Radiation Control Programs," (2) The NRC-Agreement States Exchange-of-Information program, and (3) routine correspondence and day-to-day contacts regarding regulatory matters.

The NRC "Guide for Evaluation of Agreement State Radiation Control Programs" provides guidelines for specified program indicators which the staff considers important objectives in managing an effective radiation control program. The indicators and guidelines are not of equal importance in terms of the State's ability to protect the public health and safety. For this reason, the indicators are categorized. For example, Category I indicators address program functions which directly relate to the state's ability to protect the public health and safety. Category II indicators address program functions which *support* the primary elements of the program.

2. "Deficiencies"

The NRC has commented to Colorado on several occasions on the need to increase staffing levels. Staffing level is, however, a Category II indicator. The level of staffing supports the primary program functions, such as licensing and compliance activities. Failure of a State to meet NRC recommended staffing level guidelines will not, in and of itself, endanger a State's ability to protect the public health and safety. During the NRC's 1980 onsite review of the Colorado radiation control program, the NRC staff commented to the State that the current staffing level did not appear to meet current NRC recommended guidelines regarding staffing. The staffing level was not found, however, to result in any significant weakness in any Category I indicator. In other words, the State has been able to maintain a program capable of protecting the public health and safety.

The NRC has never found Colorado's laboratory facilities inadequate. The NRC has in past years recommended improvements in laboratory conditions, such as increasing space. During the 1980 review, the NRC recommended that the State shorten the turnaround time for laboratory analyses of environmental samples obtained during mill inspections. The NRC has never found the technical capabilities of the State laboratory to be inadequate.

The NRC routinely recommends that Agreement States use automatic data processing to more efficiently manage licensing and compliance data. In the NRC "Guide for Evaluation of Agreement State Radiation Control Programs," automatic data processing is included under a Category III indicator. Category III indicators address desirable support functions, not those which directly impact public health and safety.

With regard to the assessment of civil penalties, the NRC frequently recommends that States obtain the authority to assess such penalties as a supplemental enforcement option. NRC experience has shown such an option to be a desirable feature of a regulatory enforcement policy. It is, however, not essential for the protection of the public health and safety. The Agreement States have for years been able to manage effective enforcement programs without the civil penalty option. During the last two onsite review meetings, the NRC recommended that Colorado prepare written procedures for handling escalated enforcement actions.

In sum, the deficiencies enumerated by Sunflower have never been cited as a cause of a "serious incident of failure to protect the public health and safety" on the part of the Colorado radiation control program. Each deficiency is correctable, and NRC does not think that termination or suspension of the 1968 Agreement is necessary or required because of them.

3. "Incidents"

Sunflower lists in its complaints "serious incidents of failure by Colorado to protect the public health and safety" and then lists three uranium milling and processing sites - the Cotter Corporation site near Canon City, Colorado, the Union Carbide Corporation site near Uravan, Colorado, and the site operated by Sweeney Mining and Milling Corporation near Boulder, Colorado. Neither in its complaint nor in its petition to the NRC does Sunflower state what these "serious incidents of failure" might consist of; it merely lists the three sites in its complaint and does not mention them at all in its petition. In reviewing its files, the NRC staff noted that it has the following information on "incidents" at the enumerated sites.

A failure of a small tailings impoundment area at the Sweeney mill on May 4, 1980 resulted from flash flooding due to heavy rains and melting snow. The State issued an order prohibiting further operation and requiring the licensee to take remedial action. The state's enforcement action was appropriate, and this incident cannot be construed as evidence of the State's inability to protect the public health and safety.

The NRC is aware of no specific "incidents" at the Cotter Corporation Canon City site. There is a long-standing groundwater contamination problem at the site which the state and the NRC have been attempting to solve by requiring the Company to transfer existing tailing piles to new impoundments. In addition, the Colorado Bureau of Investigation (CBI) in 1979-1980 investigated allegations of improper practices and violations of CDH rules and regulations by the Cotter Corporation. The CBI report specifically stated that CBI found no evidence of misconduct or deliberate failure to perform a duty on the part of Colorado Department of Health personnel. The CBI report did, however, make certain suggestions for changes in CDH's inspection and enforcement procedures with which NRC agreed and which CDH has implemented. The NRC does not consider this a serious failure by the state to protect the public health and safety which could justify withdrawing Colorado's agreement state status.

The NRC is also aware of no specific "incident" at the Union-Carbide Uravan site. However, during 1979, the CDH and NRC geotechnical advisors did become concerned about potential instability of the existing tailings impoundments. The state issued emergency orders in December 1979 and January 1980 to cease discharge of tailings into the areas and briefly shut down operations. Union Carbide built a rock berm and buttress to protect the existing tailings piles. The state also ordered Union Carbide to prepare a waste management plan for current and long-term operations. The CDH (with NRC technical assistance) is also dealing with the following issues at the site: safety of tailings impoundments, reclamation and long term care sureties, upgrading mill health and safety provisions, radiological reviews, and Union Carbide's long-term waste management options. Again, the NRC knows of no serious incident of failure to protect the public health and safety at this site.

Section 274j of the Atomic Energy Act

The NRC retains the authority under section 274j of the AEA to terminate or suspend an agreement state and reassert its own licensing authority. However, Congress' clear intent was that agreement states were to regulate agreement materials and that once granted, their authority is not to be revoked lightly. The legislative history of this section states that this authority to terminate "represents a reserve power, to be exercised only under extraordinary circumstances." H.R. Rep. No. 1125, 86th Cong. Session 1 (1959). p. 12.

An agreement is not to be permanently terminated or revoked for technical failures to comply with Section 274 or for single incidents of state inaction,[5] but only in exceptional circumstances. Rather, the NRC is to cooperate with agreement states and through its review process obtain

[5]This is made even more clear by the 1980 amendment to Section 274j which provides for temporary suspension of all or part of an agreement. The emergency power to terminate is limited to only those cases where (1) an emergency situation exists which requires immediate action to protect the health and safety of the public, *and* (2) the State has failed to take steps necessary to contain or eliminate the dangers within a reasonable time. The temporary suspension is to remain in effect only for as long as the emergency exists. P.L. 96-295; 94 Stat. 787 (June 30, 1980). Congress stated that this authority would be only rarely needed by NRC and that it intended the emergency power to be used only as a last resort. S.Rep. No. 176, 96th Cong. Session 2 (1979).

compliance by states. The power to terminate the agreement is to be one of last resort where all others fail.

Before the Commission may terminate an agreement, it must first provide a state with notice and opportunity for a hearing; and, before it can justify instituting proceedings to terminate, NRC must have some reason to believe that such termination may be necessary to protect the public health and safety or to secure the State's compliance with provisions of section 274.

In this case, NRC is not aware of any basis which would justify permanently terminating or suspending Colorado's agreement. Colorado has to the extent practicable complied with the provisions of section 274(o)(3)(A)(C) and (D). The "deficiencies" in Colorado's program enumerated by Sunflower do not rise to the level where the State's ability to protect the public health and safety is in doubt. The Commission finds, therefore, that there is no reason to institute proceedings pursuant to Section 274j to terminate Colorado's agreement state status or to otherwise divest Colorado of its authority over uranium mill tailings. Sunflower's petition to do so must be denied.

Chairman Hendrie's Additional Views, Commissioner Ahearne's Additional Comments, and Commissioner Gilinsky's Separate Views are attached.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.
the 24th day of June, 1981

Chairman Hendrie's Additional Views

I do not believe that there is sufficient basis for Commissioner Gilinsky's concerns about the efforts of our Office of State Programs to ensure compliance with UMTRCA. Certainly, as the Commission's Memorandum and Order shows, there is nothing to support these concerns in the case of Colorado.

Commissioner Ahearne's Additional Comments

I note there is some confusion about the applicable procedures for dealing with this type of request. Although I have concurred in the Commission's approach for this case. I believe we should look further at some of the underlying issues. There are important differences between our relationship to licensees and our relationship to Agreement States and, therefore, there should be some differences in the procedures to be followed.

Commissioner Gilinsky's Separate Views

I have voted to deny this petition because the facts in this case do not appear to justify taking action to terminate or suspend this Agreement. However, I am concerned that the Office of State Programs is not taking sufficiently vigorous steps to insure that the Agreement States exercise their authority over byproduct material in a manner which is, to the extent practicable, consistent with the requirements imposed by NRC under UMTRCA.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Joseph M. Hendrie, Chairman
Victor Gilinsky
Peter A. Bradford
John F. Ahearne

In the Matter of **Docket No. 50-395A**

SOUTH CAROLINA ELECTRIC & GAS COMPANY and SOUTH CAROLINA PUBLIC SERVICE AUTHORITY
(Virgil C. Summer Nuclear Station, Unit No. 1) **June 26, 1981**

The Commission denies a petition that it make a "significant changes" determination under Section 105c(2) of the Atomic Energy Act, precluding statutory antitrust review of applicants in connection with their pending application for an operating license for the Virgil C. Summer facility.

NRC ANTITRUST REVIEW: SIGNIFICANT CHANGES DETERMINATION

To constitute "significant changes" under Section 105c(2) of the Atomic Energy Act, the changes must (1) have occurred since the previous statutory antitrust review; (2) be fairly attributable to the licensee in a causation sense; and (3) have antitrust implications that would be likely to warrant Commission remedy.

NRC ANTITRUST REVIEW: SIGNIFICANT CHANGES DETERMINATION

Although a change in circumstances may have been anticipated with approval at the time of a previous NRC antitrust review, the later actual occurrence of the change could constitute a significant change within the meaning of Section 105c(2) if the contours of the actual change were not anticipated and could not reasonably have been anticipated.

NRC ANTITRUST REVIEW: SIGNIFICANT CHANGES DETERMINATION

To qualify as a "significant change" under Section 105c(2) of the Atomic Energy Act, the change, *inter alia,* must be reasonably attributable to the licensee in the sense that the licensee has had sufficient causal relationship to the change such that the later review would not be unfair; it need not be established that the licensee can be held legally accountable for the change.

NRC ANTITRUST REVIEW: SIGNIFICANT CHANGES DETERMINATION

The NRC must make a finding that "significant changes" have occurred since the last antitrust review before it can formally request the Attorney General's advice pursuant to Section 105c(2) of the Act. To be "significant," the changes must be reasonably apparent from required submittals of the licensees, staff investigations, or papers filed by the petitioner; discovery and examination of witnesses are not permitted in connection with this determination.

NRC: ANTITRUST AUTHORITY

If any court of competent jurisdiction finds an antitrust violation by an NRC licensee with some nexus to its nuclear license, the Commission is empowered to take whatever course of action it deems appropriate, including conditioning or withdrawing the license.

NRC ANTITRUST REVIEW: STATE REGULATION

While decisions left open to parties under state regulatory systems, and thus dictated by business judgment and not regulatory coercion, may be subject to findings of antitrust violations. Activities conducted pursuant to state statutory regulatory requirements are neither violations of the antitrust laws nor the policies underlying those laws.

DECISION

The Commission denies the petition[1] of Central Electric Cooperative, Inc. (Central) for an affirmative "significant changes" determination under section 105c(2) of the Atomic Energy Act of 1954, as amended (the Act), 42 U.S.C. 2135(c)(2) (significant changes decision). The effect of today's decision is to preclude statutory antitrust review of South Carolina Electric & Gas Co. (SCEG) and South Carolina Public Service Authority ("Santee Cooper" or "Authority") (jointly, applicants) in connection with their pending application for a license to operate the Virgil C. Summer facility.[2] Our reasons follow.

I. BACKGROUND

A. The June 30, 1980 Order

The background of this proceeding through June 30, 1980 is set forth in our order of that date, CLI-80-28, 11 NRC 817, which discussed the standards for a significant changes decision, tentatively decided some of the issues comprehended in this matter, and requested the Department of Justice's ("Justice") threshold views on the ultimate likelihood that the Commission would need to place remedial antitrust conditions on the Summer license. Because we had for the first time proposed criteria for a significant changes decision in our analysis of the instant matter,[3] we invited the parties and Justice to provide comments on the criteria and our application of them to the *Summer* facts. In light of the "staleness" of the record, we further granted an opportunity for the parties to advise of any recent developments.

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[1]On December 6, 1978 Central first petitioned the Commission for a finding of significant changes. Pursuant to the Commission's order of January 26, 1979, Central amended that petition on January 31, 1979. "Petition" in this opinion refers to the amended petition unless otherwise stated.

[2]By this order we also deny the March 23, 1981 Petition to Intervene and Request for Hearing of Fairfield United Action insofar as it requested significant changes determination and antitrust hearing. Because the petition brings no information or allegation of significant changes other than that already considered and found insufficient in this decision, we need not reach issues of timeliness, sufficiency, standing and the like.

[3]By way of review, the first criterion required that the changes alleged shall have taken place since the previous statutory antitrust review, the second that they should be fairly attributable to the licensee in a causation sense, and the third established that changes would be considered "significant" only when the competitive structure, as changed, would likely warrant and be susceptible to a greater than *de minimis* license modification.

B. Responses to the Commission's June 30, 1980 Order

Central, SCEG, Santee Cooper, NRC staff and Justice all responded to the Commission's June 30, 1980 Order:[4]

Central agreed with the Commission's analysis in the main, disagreeing only with application of the third criterion insofar as it discussed the limits on the extent to which state regulation displaces the antitrust laws. Central argued that the Atomic Energy Act's expressed national policy in favor of competition overrides South Carolina's policy in favor of territorial limitations. Thus Central concludes that *Parker* v. *Brown*[5] does not immunize any of applicant's anticompetitive actions. Central also reported inconclusive settlement negotiations with SCEG and Santee Cooper individually. Central alleged that one provision of the agreement with Santee Cooper being negotiated (later reported as adopted) precluding Central from extending its transmission lines was unenforceable as violative of the antitrust laws.[6]

SCEG, was in fundamental disagreement with the Commission's criteria and analysis. It argued that the definition of significant change should be a substantial change in the competitive structure "allowing for circumstances fairly predictable in the natural course of events" and limited to changes which themselves have negative antitrust implications.[7] SCEG also argued that the Commission's distinction between assessing causation under the second criterion and determining a violation of the antitrust laws was not only wrong but "legally ridiculous" and that the Commission improperly

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[4]Page references are respectively to the following documents:

Comments of the Petitioner Central Electric Power Cooperative, Inc., August 25, 1980

Letter from Troy B. Conner, Jr. (counsel for SCEG) to Samuel J. Chilk, August 22, 1980

Letter from T. C. Nichols (Vice Pres., SCEG) to Samuel J. Chilk, August 25, 1980

Response of South Carolina Public Service Authority to the Nuclear Regulatory Commission's Request for Comment on its "Significant Changes" Criteria and for a Factual Update, August 25, 1980

NRC Staff Response to Commission Request for Comments, August 29, 1980

Response of the U.S. Department of Justice to the Nuclear Regulatory Commission's Request for Comment on its "Significant Changes" Criteria and the Application of those Criteria, October 10, 1980.

[5]*Parker* v. *Brown*, 317 U.S. 341 (1943), is the leading case for the proposition that actions taken pursuant to valid state regulation are immune from the prohibitions of the Federal antitrust laws.

[6]At 9-10.

[7]Letter from Conner at 3.

sought the advice of Justice. As a bottom line,[8] SCEG urged a finding of no significant changes with regard to itself, leaving any potential for operating license conditions only directed toward Santee Cooper.[9]

Finally, SCEG reported it was considering a proposal from Central regarding "wheeling",[10] which SCEG characterized as Central's only specific transmission service request,[11] and indicated that there had been no changes in SCEG's competitive relationship with Central and Santee Cooper since December 1978.[12]

Santee Cooper commented that "while the Commission's three legal criteria for a 'significant changes' determination may in the abstract provide a valid test, the application of this three-pronged test to the instant facts gives rise to a result that is squarely inconsistent with Congressional purpose."[13] Santee Cooper noted a number of recent developments including agreement with Central enabling it to obtain an ownership interest in future generation facilities constructed by Santee Cooper and to join with Santee Cooper in coordination and planning of future generating and transmission facilities.[14]

NRC Staff was in basic agreement with the Commission's criteria, but expressed a concern with respect to the Commission's application of *Parker* v. *Brown* and the Commission's understanding of the reach of (state action) immunity.[15] Staff's "preliminary conclusions are that Central is being availed increased power supply options ...; that these new power supply opportunities ... enhance its own economic well-being; and finally, that these new developments are pro-competitive in that many of Central's previous allegations of anticompetitive effects resulting from changed circumstances have been redressed."[16]

[8]*Id.* at 16 n. 36.

[9]SCEG refuted what it believed to be the Commission's position that Santee Cooper turned down Central's proposal for an ownership share in Summer. The Commission's statement was that Santee Cooper turned down a proposal for joint ownership of transmission facilities. *See* SCEG letter from Conner at 9. *See also* CLI-80-28, *Supra,* 11 NRC at 836.

[10]" 'Wheeling,' a term of art, refers to the 'transfer by direct transmission of displacement electric power from one utility to another over the facilities of an intermediate facility.' *Otter Tail Power Company* v. *United States,* 410 U.S. 366, 368 (1973)," *Toledo Edison Co.* (Davis-Besse Nuclear Power Station, Units 1, 2 and 3), ALAB-560, 10 NRC 265, 275, fn. 24, (1979).

[11]At 3.

[12]At 7.

[13]At 20.

[14]Santee Cooper advised of (1) a recent amendment to the South Carolina Constitution authorizing Santee Cooper to become a part owner with cooperatives in electric generation and transmission, and (2) empowering Santee Cooper to own jointly with Central generation and transmission facilities. At 8, 9 and 10.

[15]At 8.

[16]At 2 (citation omitted).

Justice was in accord with the Commission's first two criteria but urged the Commission to modify the third criterion. Justice objected to that criterion because in its view it entailed an antitrust review prior to the significant changes determination contrary to the statutory scheme in section 105c(2). Moreover, Justice averred that there was no mechanism for obtaining necessary information from licensees in that context.[17] Justice proposed that the third criterion should be: whether the changes are substantial within the competitive environment, i.e., changes in the structure of the market or in the conduct of the licensee with respect to the construction or operation of the licensed plant. Justice noted that it would generally be difficult to determine whether such changes are pro- or anti-competitive without an antitrust review and that it therefore did not include in the criterion a requirement that the change must be adverse.[18] Justice stated that events unforeseeable at the previous antitrust review, events that were a distinct possibility at the time of previous review but had become certain, and conduct that had previously not been ripe for review at the construction permit (CP) review stage could all constitute significant changes. Justice urged that neither a determination of whether changes were pro- or anti-competitive nor a determination whether remedies were available was appropriate at the significant changes determination stage.[19]

Regarding application of the criteria to the instant facts, Justice noted that its review of the pleadings revealed several strongly controverted allegations of changes that would be sufficient for a significant changes determination if the Commission's own staff found that they were meritorious. But Justice declined to make any preliminary or threshold review. It advised, however, that Central "has made one uncontroverted allegation that may contribute "significant changes" within the meaning of Section 105c(2). Justice discussed that change as follows:

> There is no dispute that in 1978 SCEG and Santee Cooper lobbied for, and the South Carolina legislature enacted, legislation restricting the area in which Santee Cooper can compete. It is likely that as a consequence of this legislation, Santee Cooper has altered its "activities or proposed activities" under the license in that it ceased competing for the business of municipalities and cooperatives other than Central, outside its three-county service area. This change in Santee Cooper's "activities or proposed activities" would have taken place since the prior antitrust review of March 31, 1972. Thus, if the Commission were to conclude that Santee Cooper has changed its conduct as a result of

[17]At 4-5.
[18]At 6 and n. 12
[19]At 6-7.

the South Carolina statute, the only remaining issue would be whether this change in conduct is reasonably attributable to the licensee(s).

The Department concurs with the Commission's suggestion that, as a matter of law, a licensee's lobbying activities can be sufficient to attribute conduct under the resulting statute to the licensee. However, the Department defers to the Commission to determine whether Santee Cooper's lobbying activities are sufficient in this case.[20]

In light of this view Justice did not reach the issue of *Noerr-Pennington* (lobbying for anticompetitive legislation is protected by the 1st Amendment), or *Parker* v. *Brown* (actions in conformance with valid state regulation are protected) immunities.[21] According to Justice "the only issue at this stage is whether licensees are responsible or answerable under the Atomic Energy Act, not whether they are liable under the Sherman Act. Justice emphasized:

> [A] "significant changes" determination imposes no liability; it merely triggers antitrust review. Even if an antitrust hearing eventually resulted and the Commission found it necessary to impose license conditions to remedy a situation inconsistent with the antitrust laws, those conditions need not constitute sanctions comparable to those that could flow from liability under the Sherman Act.[22]

C. The Agreement Between Central and Santee Cooper

On January 14, 1981, counsel for Santee Cooper filed a "Power System Coordination and Integration Agreement Between South Carolina Public Service Authority and Central Electric Power Cooperative, Inc." (Agreement) with that Commission. The following day we requested Central, applicants, Justice and the staff to comment on the Agreement's effect, if any, on our pending determination. The main points to emerge from those comments follow.[23]

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[20]At 9.

[21]*Id.*

[22]At 10-11.

[23]Page citations are respectively to the following documents:

Comment of Central Electric Power Cooperative, Inc., January 23, 1981

Comments of South Carolina Electric & Gas Company in response to Commission Order of January 15, 1981, January 23, 1981.

Response of South Carolina Public Service Authority to the Nuclear Regulatory Commission's January 15 Order Requesting Comment on the Agreement Between Central and the Authority, January 23, 1981

Central maintained that "the agreement does not eliminate the reasons for a finding of significant changes."[24] In support of that view it raised the spectre that the Internal Revenue Service might challenge some of the provisions thereby defeating the contemplated transactions relating to requirements and joint ownership of generating units. Assuming survival of the Agreement, which Central clearly sees as preferable, Central argued that it is "tied to the pricing or coordination terms of a single party [Santee Cooper]" unless it can obtain contracts for coordination with base load units or wheeling from SCEG and other utilities who are the very ones, according to Central, who have joined in illegal market division agreements.[25] As a final point Central indicated willingness for post-licensing antitrust review for the facility.[26]

SCEG asserted that the Agreement affords Central the access to power and mechanism for "power exchange services" which it has sought from either SCEG or Santee Cooper.[27] Quotations from the Agreement were marshalled to support the proposition that Central no longer has any needs from SCEG.[28] Moreover, SCEG said that the Commission had decided that applicant's conduct leading up to South Carolina's territorial decision was not a factor in the significant changes decision.[29] The elimination of this factor served as a further basis for SCEG's view that the significant changes determination should be negative.

Santee Cooper commented that the Agreement removes any arguable basis for imposing any license condition to ameliorate possible antitrust problems.[30] Central is guaranteed a reliable source of power at cost of service and "Central will under the Agreement approved by REA purchase its bulk power requirements with a few limited exceptions from the Authority." "Thus," Santee Cooper maintains, "there is no basis for further inquiry into Central's allegations regarding [SCEG's] unwillingness to wheel power for Central purchased from sources other than the Authority, particularly since Central has not alleged that [SCEG] has refused to carry out any specific wheeling transaction that was requested."[31] Santee Cooper

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Comments of the Department of Justice in Response to the Nuclear Regulatory Commission Order of January 15, 1981, February 6, 1981

NRC Staff Reponse to Commission's Order of January 15, 1981.

[24]At 1.

[25]At 5.

[26]At 14.

[27]At 2.

[28]At 9.

[29]At 3.

[30]At 6.

[31]At 7.

argued as a legal matter that the Commission does not have "carte blanche" to impose antitrust licensing conditions[32] and that the Atomic Energy Act's "legislative history makes abundantly clear, [that] Section 105c was not intended to be a no fault statute for restructuring electric power markets."[33]

Department of Justice declined to make the factual determination whether the Agreement is sufficient to, in effect, eliminate the anticompetitive effect of any change that may have occurred. Justice did note that certain recent comments, particularly those of Central, caused it to believe that the South Carolina territorial legislation may have had "less competitive significance to Central than may have appeared at first blush."[34] Moreover, Justice took the occasion to explain its previous advice on the third criterion. It stated that: "In making this determination the Commission should take into account whether an antitrust review would serve no useful purpose and, thus, would be inconsistent with the Congressional intent that antitrust reviews at the operating license stage not be lightly undertaken."[35]

NRC Staff's assessment was that the Agreement provides "ostensibly reasonable opportunities" for Central "to obtain its future generation and transmission needs with the Authority on a jointly-planned and jointly-coordinated basis, with accompanying guarantees for 'cost of service' rates."[36] Because the Agreement basically gives Central what it sought in its Petition and because traditional NRC antitrust remedies go no further than the Agreement, providing only a "general charter" for dealings between the utilities, staff concludes that the Agreement diminishes the possibility that the third criterion can be met.[37] Moreover, staff notes that SCEG has given Central its assurances that it will wheel power[38] and that staff has seen no "factual material that would lead to the conclusion that SCEG is explicitly or constructively refusing, in an anticompetitive manner, to provide Central with power services."[39]

II. ANALYSIS

With the foregoing as background, we now turn to the merits of the Petition. We shall first consider the criteria to be used for the significant

[32]At 8.
[33]At 10.
[34]At 4 n. 4.
[35]At 3 n. 3.
[36]At 11.
[37]Staff finds unpersuasive Central's concerns resulting from IRC uncertainties.
[38]At 9.
[39]At 12.

changes determination, then discuss the requirement of a factual basis for the significant changes, and finally apply the criteria to the instant petition.

A. Criteria for the Significant Changes Determination

In our June 30, 1980 Order we clarified that, in considering whether there had been changes since the last antitrust review, the operative date must be the date of the last actual review. We adhere to that view. However the more precise issue arises whether a change anticipated by the review at the construction permit stage but in fact occurring since that review meets the requirement of the statute represented by this first criterion.

Taking into consideration the careful balance struck by the Congress in deciding whether to have any antitrust review at the operating license stage, we find that Justice's suggestion that a significant change could occur when "what had been possible was now certain"[40] goes too far in one direction. However we also find that Applicants' suggestion that if an event was foreshadowed at the earlier review stage, it may never be a significant change goes far in the other direction. We believe that where some change was anticipated with approval at the previous stage the later actual occurrence of the change could constitute a significant change within the meaning of section 105c(2) if the contours of the actual change were not anticipated and could not reasonably have been anticipated. Thus, for example, if an applicant at the construction permit stage was known to be negotiating with another utility to permit access and those negotiations were later successfully concluded, the occurrence of that event in itself would not be the subject of a significant change. But if that event were, hypothetically, to be linked to the other utility's agreement to an anticompetitive policy regarding transmission that had not been considered at the antitrust review and was not reasonably to be expected, then the access transaction would be a significant change, assuming it also met the other two criteria.

2. Causation

Our June 30, 1980 Order established as the second criterion that the change or changes must be reasonably attributable to the licensee in the sense that the licensee has had sufficient causal relationship to the change that it would not be unfair to permit it to trigger a second antitrust review. We adopt the criterion.

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[40] At 6.

The applicants have criticized the criterion on the ground that Congress meant to include as significant changes only those for which licensees could be held legally accountable.[41] We reiterate our view that SCEG's formulation ignores the elements of causation and of fairness and reasonableness that the Joint Committee on Atomic Energy took care to include in their report.[42] A finding of significant changes, and *a fortiori* a finding that one criterion for such a finding has been met, is limited to the purpose of determining whether an antitrust review should be held and does not determine the outcome of any such review. Applicants, by arguing that legal accountability must be established at the threshold, would require that the review be essentially completed before it can even be commenced. Any threshold forecast of outcome pursuant to the third criterion is only to avoid a review where no purpose is to be served by holding one, such as where changes have occurred but are not anticompetitive or where anticompetitive effects of changes are beyond the Commission's power to remedy.

3. Significance

We decided in our June 30, 1980 Order that "significance" must here be read to mean that the changes have *antitrust implications* that would be *likely* to warrant *Commission remedy.* We affirm that decision.

This criterion has provoked the widest divergence in views, with the poles represented by Justice on the one hand and SCEG and Santee Cooper on the other.

As the staff recognized, "this third criterion appropriately focuses, in several ways, on what may be 'significant' about any changes since the last ... review. Application of this third criterion should result in termination of NRC antitrust reviews where the changes are pro-competitive *or have de minimis anticompetitive effects.*" (Emphasis provided) The staff correctly discerned that the third criterion has a further analytical aspect regarding remedy: "Not only does [it] require an assessment of whether the *changes* would be likely to warrant Commission remedy, but one must also consider the type of remedy which such changes by their nature would require."[43] The third criterion does not evaluate the change in isolation deciding only whether it is pro or anticompetitive. It also requires evaluation of

[41]SCEG's objection is bound up in its insistence that *Noerr-Pennington* shields it from our causation finding. We will return to this subject when we analyze our decision on the instant facts.

[42]*See* H.R. Rep. No. 91-1470, *supra.*

[43]Staff's February 10, 1981 Response at 7, noting that early identification of possible NRC remedies is not novel with regard to invocation of NRC antitrust proceedings.

unchanged aspects of the competitive structure in relation to the change to determine significance.

B. Requirement of a Factual Basis for the Changes Alleged

In our June 30, 1981 Order we explained the role of the significant changes determination, observing that a finding that significant changes have occurred must precede a *formal request for the Attorney General's advice* in any statutory antitrust review. Congress has made it abundantly clear that absent such a finding there is to be no antitrust review proceeding at the operating license stage. That Congressional directive may not be circumvented by expanding a petition for significant changes into a proceeding with all the attributes of a full-fledged hearing — discovery, examination and cross-examination of witnesses and the like. In sum, we do not believe Congress intended that we conduct a proceeding to ascertain whether to have a proceeding. Inherent in that result is a recognition that the parties, other than the Commission do not have discovery or the other means for determining facts commonly associated with formal adjudication.

Thus, we understand Congress's meaning to be that changes in order to be significant must also be reasonably apparent. They must be alterations in the competitive structure or the activities of the licensees discernible from applicants' required submittals, from staff's investigations, or from papers that are filed. In particular when petitioners request a significant changes determination we expect that the changes which have taken place will be known to them so that they can inform us of them with the factual basis underlying their allegations.[44] If that, together with staff investigation, does not enable us to determine that significant changes have occurred, then the petition must be denied.

This result is consistent with Congress's expressed intent not casually to burden applicants with a second antitrust review after an extensive antitrust review at the construction license stage.[45] We can not embark on a second

[44]Accordingly, Central's motion for permission to conduct discovery, which we have held in abeyance, is denied.

[45]Parties may be reminded that other forums exist in which to try allegations of antitrust violations. Furthermore, we are bound to transmit to Justice such allegations as are made to us. *See* Section 105a of the Act. We consider in this matter that Justice is on notice of Central's allegations. Moreover, in the event that a court of competent jurisdiction finds an antitrust violation by a licensee with some nexus to its nuclear license, we are empowered to take whatever course of action we deem appropriate including among other things conditioning the license or withdrawing it. This is, of course, true before and after an operating license has been granted. Additionally, as a separate matter, if the facts and information supplied by applicants and relied on by us in granting a license are found to be false, the license may be in jeopardy and other measures are available. *See* Section 186a of the Act and *Houston Lighting & Power*

antitrust review without specific facts which show that all of the criteria for the significant changes determination are met.

We do not intend the leeway we have allowed Central for repeated filings to set a precedent;[46] rather, we wished to be liberal in entertaining whatever information Central put before us by way of compensation for the special circumstance to which we adverted in our Opinion, namely that no statutory review of Santee Cooper was conducted by the Attorney General at the time it became associated with SCEG in the construction of the Summer facility.[47]

C. Application of the Criteria to this Matter

1. First and Second Criteria

We adhere entirely to our earlier views regarding application of the first two criteria, and add only a few words about each.

Concerning the first criterion, it has been suggested that because South Carolina had already embarked on territorial legislation the new legislation here at issue should be considered to have been anticipated. We think this is wrong. Neither the fact of legislation nor its provisions were sufficiently foreseen that any account of them was taken at the previous antitrust review.

With regard to the second criterion, the Department of Justice has persuasively refuted the argument that the *Noerr-Pennington*[48] doctrine prevents the Commission's causation finding:

> The only issue at this stage is whether licensees are responsible or answerable under the Atomic Energy Act, not whether they are liable

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Co. (South Texas Units 1 and 2), 5 NRC 1303, 1311 (1977). Accordingly, facts relied on to make a significant changes determination have this status.

[46] In our tentative June 30, 1980 Order we addressed the issue of timeliness and concluded that under the circumstances and in the interest of fairness we should regard Central's petition as timely. We relied, in part, on the fact that Central had not had unambiguous notice of opportunity for antitrust comment. 11 NRC 829-830. We had accepted at face value Central's statement that as soon as it had learned it might have rights it could assert in this proceeding it retained an attorney for the purpose of studying whether it could obtain an antitrust review, and immediately following that study filed the instant petition. Petition at 1. Central has since our June 30, 1980 Opinion cast these facts in a different light, indicating that as early as August, 1977 Central chose to exercise whatever rights it had in this forum. *See* Comments of Petitioner Central Electric Power Cooperative, Inc., August 25, 1980 at 14-15. However, in light of the decision we have reached on the significant changes determination we need not today decide what effect, if any, our new understanding should have on the timeliness decision.

[47] As we explained in our June 30, 1980 Order, since *Fermi* (Detroit Edison, *et al* (Enrico Fermi Atomic Power Plant, Unit No. 2), 7 NRC 583, 587-9, *aff'd* ALAB-475, 7 NRC 752, 755-56 n. 7 (1978)) review of a new co-owner is required. 11 NRC at 830-831.

[48] *Eastern Railroad Presidents Conference* v. *Noerr Motor Freight, Inc.*, 365 U.S. 127 (1960); *United Mine Workers* v. *Pennington*, 381 U.S. 657 (1965).

under the Sherman Act. A "significant changes" determination imposes no liability; it merely triggers an antitrust review.[49]

The role of the Noerr-Pennington doctrine is to assure that there will be no liability or penalty for the exercise of First Amendment rights. Such a penalty could not result from the conduct, as opposed to the outcome, of formal antitrust review. Of course the outcome of formal review would necessarily recognize activities protected by Noerr-Pennington are not antitrust violations. Consequently the *Noerr-Pennington* doctrine could be considered in connection with the third criterion. However, we need not reach that issue in this case.

2. Third Criterion

As indicated, our inquiry here is essentially whether there is sufficient likelihood that the Commission's remedial powers will be exercised so that some purpose would be served by entering on the process of antitrust review.

In its Petition Central summarized the significant changes it believes to have occurred as follows:[50]

> (a) SCEG exercised its monopoly power in the power exchange market by conditioning Santee-Cooper's participation in the Summer Unit in exchange for an elimination of competition [with SCEG by Santee Cooper] at retail and wholesale.
>
> (b) Santee-Cooper has changed its competitive role and marketing policy [vice-a-vis Central] and has thereby aligned itself with SCE&G.
>
> (c) SCE&G and Santee-Cooper do not compete for loads of 750 KW or greater outside of the three county area, although both of these utilities have agreed [by agreeing to seek the legislation] to compete with Central's members [member cooperatives] for these loads.
>
> (d) Santee-Cooper has implemented a dual rate policy for large loads [thereby charging higher rates to at least one new purchaser-member of Central].

[49]Justice's Response, October 10, 1980 at 10.

[50]Bracketed information within the quotation represents our understanding of Central's meaning based on the entire Petition and other documents submitted to us.

Although one change meeting the first and second criteria is sufficient to permit consideration of the entire competitive structure in evaluating the third criterion, we note that each of Central's seven alleged changes, if it in fact occurred, meets the requirements of the first and second criteria.

(e) Santee-Cooper has agreed to restrict it sales in the wholesale market.

(f) Both SCE&G and Santee-Cooper have refused to provide power exhange services and facilities to Central, thereby preventing Central from constructing and operating bulk power facilities.

(g) Santee-Cooper has offered [to Central] to acquire control over Central's bulk power supply function.[51]

The anticompetitive treatment of large loads and wholesale supply complained of in (c) and (e) above must be immediately recognized as not subject to our remedial powers. While Central has framed its assertions in terms of "agreement" between applicants and, consistent with Central's view that *Parker* v. *Brown* has no application here, has ignored the role of state legislation, we do not follow Central's lead. As we have explained before, the law seems clear to us that activities conducted pursuant to state statutory regulatory requirements are neither violations of the antitrust laws nor the policies underlying those laws.[52] In (c) and (e) the activities that are the subject of the alleged "agreement" are required by the state as a part of a state regulatory plan. With respect to them, applicants have no freedom of choice. Thus, they may not be the subject of our license modifications.

Items (b), (d), (g), and part of (f) concern the activities of Santee Cooper. Whatever may have been the status of these allegations before the Agreement between Central and Santee Cooper,[53] that Agreement has laid them to rest. The Agreement deals in a comprehensive fashion with the relationship between the parties and provides for the furnishing of power and power exchange services. As SCEG, Santee Cooper and staff have pointed out, it is clear that Santee Cooper and Central continue to be aligned and that benefits to Santee Cooper through access to the Summer facility will be available to Central. Apart from the role of the South

[51]Petition at 49. Central has at various times offered somewhat different formulations of these assertions, as have the applicants and staff. *See*, e.g., Staff's Response to Amendment Petition of Central, March 19, 1979 at 25. However, we will accept Central's formulation in its Petition as subject for our response.

[52]We reiterate our view that decisions left open to parties under the state regulatory system and thus dictated by business judgment, not regulatory coercion, may be subject to findings of antitrust violations. As we understand it, the reach of *Parker* v. *Brown* does not extend beyond what is required for the integrity of the state regulatory plan. In that regard we quoted the language of *Philadelphia* v. *U.S. Bank* defining the bounds of federal regulatory exemption vis-a-vis the antitrust laws and adopted as a part of our test that activities at the free choice of parties and not "repugnant" to the state plan would be subject to remedial action where required in the interest of the antitrust laws or the policies underlying those laws.

[53]Our Opinion disposed of (d).

Carolina legislation, other allegations of changed roles have been resolved by the Agreement.[54]

We had earlier indicated that the "dual rate" allegations of item (d) seemed to us to be insubstantial because Santee Cooper's charter obliges it to supply power at "cost of service". As a result of the Agreement, we are convinced that nothing survives of this allegation. Regarding item (g), it is unclear whether an "offer" can be construed to be an anticompetitive change; however we need not decide that issue because Central has agreed with Santee Cooper on transmission, as well as generation, planning and operations, retaining distinct ownership of facilities or portions of them.

Viewing allegations (c), (d), (g), and (f), as it pertains to Santee Cooper, in their totality, we have not been persuaded that, reading the South Carolina legislation in tandem with the Agreement, Central's competitive position has deteriorated since the last review of the Attorney General.

There remain the allegations of (a) and the part of (f) dealing with SCEG.

In our Order we adverted to the claim in (a) that SCEG wielded monopoly power to coerce Santee Cooper to seek territorial legislation as the price for access to the Summer facility. We explained that even if we found that SCEG has committed a Sherman Act violation here, as alleged, that in itself would not repeal South Carolina's laws and would not remove the *Parker* v. *Brown* immunity from actions commanded by state law. This is not to say that were we to find that SCEG has used access to a nuclear facility as a club to coerce behavior we would be powerless to take remedial action. Nonetheless, having reviewed all of Central's statements, we do not find sufficient substance in the papers filed by Central to support this claim. By affidavit we have been informed that "it was common knowledge in early 1973 that [SCEG] was conditioning participation by Santee Cooper in the Summer Unit upon enactment of the territorial law" and that "Electric and Gas representatives at the State House have been telling that [SCEG] is not going to sell the Authority power out of the nuclear plant and then have it compete with Electric and Gas." We do not think that such generalized hearsay would be sufficiently detailed and reliable evidence on which to base our decision. Moreover, the contemporaneous analysis of the events leading to the passage of the legislation which Central presented as support for its view is not internally consistent. That same document states that SCEG could not have coerced Santee Cooper's action because it was

[54]More detailed discussion of (f) will be presented *infra* with comments on (f) as it pertains to SCEG.

common knowledge that the Justice Department would have assured that Santee Cooper got a share of the Summer facility.[55,56]

With regard to (f), allegations regarding refusal to provide power exchange services have been of concern to us. Central's Petition stated that it required power exchange services from "either Santee-Cooper or SCE&G"[57] and had been unable to conclude such arrangements. As we discussed above, the Agreement has altered the situation between Central and Santee Cooper so that our concerns have been alleviated. It is notable that a Constitutional amendment and legislation were apparently required to empower Santee Cooper to own and develop transmission in conjunction with Central. These things have now been accomplished, and in any event this matter has been resolved between Central and Santee Cooper. The resolution substantially reduces and arguably eliminates the importance of SCEG's failure to conclude arrangements to wheel power for Central. Nonetheless, we note that SCEG advised that it has not refused to wheel power, that it will provide *ad hoc* transmission services, that it continues its negotiations with Central which we must assume are conducted in good faith. Central has offered no facts which dispute this, besides the bare allegation of refusal to deal in the petition.[58]

Moreover, staff states that it knows of no SCEG refusal to provide power services to Central.[59] Furthermore, we consider SCEG's assertions as having been provided us for the purpose of securing a license with all that that entails.[60] As such we need not reject them without a more specific and

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[55]*See* Central's Reply Brief, March 19, 1975, Kelly affidavit, ¶ 4, 6 and Attachment "Proposed Senate Bill 389, ¶ 9.

[56]In the event that negotiations between SCEG and Santee Cooper had not been successfully concluded as had been anticipated, we might have looked favorably on a suggestion that there had been "significant changes" because the anticipated event had not occurred.

[57]Petition at 5.

[58]Central has advised that SCEG rejected a proposed general agreement in the nature of license conditions, and has provided that proposal for the record. Reply Brief, Exhibit B. We cannot agree that SCEG's single failure to accept this particular proposal constituted a refusal either to wheel or to negotiate.

[59]Staff stated: "The only refusal which has come to Staff's attention is the refusal of Santee Cooper to build jointly owned transmission lines with Central ...." NRC Staff Response to Amended Petition, March 19, 1979 at 51.

[60]*See South Texas, supra,* 5 NRC at 1311.

detailed basis than Central has presented.[61] Thus we accept SCEG's statements as true.

Before concluding our analysis we comment on one additional allegation that recurred in Central's presentation.[62] Central has argued from time to time that Santee Cooper would only permit its participation in the Summer facility on terms with heavy financial penalties.[63]

From all that we can determine the Agreement including the provision for Central's option to purchase a share of Summer was made by the parties in good faith. We understand that the terms stated would, if exercised, require Central to pay its prorated share of actual costs. On its face, such an offer does not seem consistent with "heavy financial penalties" as alleged by Central.

III. CONCLUSION

For all the foregoing reasons we decline to find that significant changes have occurred in the activities or proposed activities of applicants within the meaning of section 105c(2). We therefore do not request the formal advice of the Attorney General.[64]

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Wasington, D.C.
the 26th day of June, 1981.

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[1]Central's Petition asserts only as follows:

> Central's efforts to secure power exchange services from SCE&G have also been unsuccessful. In the Saluda hydroelectric relicensing proceeding before the Federal Power Commission, Central sought license conditions which would require SCE&G to engage in power exchange transactions. These license conditions were patterned after those used in NRC proceedings. In March 1977, SCE&G refused to engage in the power exchange services requested by Central, except that it did agree to wheel discrete amounts of power between discrete points on a case by case basis. Such a wheeling policy is hardly sufficient and would obviously frustrate Central's attempt to enter the bulk power business.
>
> Central did direct further inquiries to SCE&G on wheeling, but SCE&G has yet to make a response. Petition at 46.

[2]Central has made no argument of other unchanged circumstances whose competitive aspect has altered as a result of changes reasonably attributable to applicants.

[3]E.g., Reply Brief at 20.

[4]All pending motions consistent with this result and those overtaken by time are rendered moot, those inconsistent with this result are denied.

Cite as 13 NRC 881 (1981) ALAB-642

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-395 OL**

SOUTH CAROLINA ELECTRIC AND GAS COMPANY, *et al.*
(Virgil C. Summer Nuclear Station, Unit 1) **June 1, 1981**

The Appeal Board reverses the Licensing Board's grant of an untimely intervention petition (LBP-81-11).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

Untimely petitions for intervention must be judged by a balancing of the five factors set forth in 10 CFR 2.714(a):

(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.

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

The appellate review of licensing board application of the five factors used in ruling upon untimely intervention petitions is governed by the "abuse of discretion" standard. *Nuclear Fuel Services, Inc.* (West Valley Reprocessing Plant), CLI-75-4, 1 NRC 273, 275 (1975); *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-420, 6 NRC 8, 13 (1977); *Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-354, 4 NRC 383, 389, 390 (1976).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS

In reviewing licensing board decisions on untimely intervention petitions, appeal boards may closely scrutinize the factual and legal ingredients of the analysis underlying the licensing board's ultimate conclusion. *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-420, 6 NRC 8 (1977); *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 2), ALAB-384, 5 NRC 612 (1977); *Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-354, 4 NRC 383 (1976).

RULES OF PRACTICE: DISCOVERY

Parties to a proceeding are entitled to obtain in advance of hearing much more than simply a summary statement of the bases for their adversaries' claims and some identification of potential witnesses; "[i]n modern administrative and legal practice, pretrial discovery is liberally granted to enable the parties to ascertain the facts in complex litigation, refine the issues, and prepare adequately for a more expeditious hearing or trial." *Pennsylvania Power and Light Co.* (Susquehanna Steam Electric Station, Units 1 and 2), ALAB-613, 12 NRC 317, 322 (1980), quoting from *Pacific Gas and Electric Co.* (Stanislaus Nuclear Project, Unit 1), LBP-78-20, 7 NRC 1038, 1040 (1978).

RULES OF PRACTICE: UNTIMELY INTERVENTION PETITIONS (BROADENING ISSUES FACTOR)

A belated intervention petition need not introduce an entirely new subject matter in order to "broaden the issues" for the purposes of 10 CFR 2.714(a); an expansion of issues already admitted to the proceeding also qualifies.

ADJUDICATORY PROCEEDINGS: ROLE OF HEARING BOARDS

Where necessary in the interest of insuring that a proper record is compiled on all matters in controversy (or raised by it *sua sponte*), hearing boards have the right and the responsibility to take an active role in the examination of witnesses.

OPERATING LICENSING PROCEDURES: RESPONSIBILITY OF NRC STAFF

An operating license may not issue unless and until the NRC staff makes the requisite findings specified in 10 CFR 50.57 — including the ultimate finding that such issuance "will not be inimical to *** the health and safety of the public." As to those aspects of reactor operation not considered in an adjudicatory proceeding (where one is conducted), it is the staff's duty to insure the existence of an adequate basis for each of the requisite Section 50.57 determinations.

APPEARANCES

Mr. Joseph B. Knotts, Jr., Washington, D.C., for the appellants South Carolina Electric and Gas Company *et al.*

Mr. Steven C. Goldberg for the appellant Nuclear Regulatory Commission staff.

Dr. John C. Ruoff, Jenkinsville, South Carolina, and **Mr. Robert Guild**, Columbia, South Carolina, for the appellee Fairfield United Action.

DECISION

This operating license proceeding involves Unit 1 of the Summer nuclear facility, located in Fairfield County, South Carolina. It was instituted more than four years ago by the publication of a notice of opportunity for hearing. 42 *Fed. Reg.* 20203 (April 18, 1977). In response to that notice, one intervention petition and request for a hearing (that of Brett Allen Bursey) was filed and, in 1978, granted. LBP-78-6, 7 NRC 209.[1] In addition, the State of South Carolina was given leave to participate in the proceeding under the "interested State" provisions of 10 CFR 2.715(c).

The prehearing stage has extended over a protracted period of time. The proceeding is, however, now ready for trial. On March 9, 1981, the Licensing Board issued a memorandum in which, acting upon the agreement of the parties, it tentatively set the commencement of the

[1] The notice required petitions to intervene to be filed within 30 days (*i.e.*, by May 18, 1977). 42 *Fed. Reg.* at 20204.

evidentiary hearing for June 22, 1981. Subsequently, that date was confirmed.

As of March 9, the necessary contemplation was that the hearing would embrace those contentions of Mr. Bursey which had been admitted to the proceeding, together with certain questions which the Board itself had raised *sua sponte*. See 10 CFR 2.760a. The further expectation was that the participants would be four in number: the applicants; Mr. Bursey; South Carolina; and the NRC staff. But precisely two weeks later, on March 23, a new face appeared on the scene. Armed with a plethora of proposed contentions of its own, an organization comprised of Fairfield County residents — entitled Fairfield United Action (hereafter FUA) — filed a petition for leave to intervene.

It is the action taken by the Board below on that petition which has now brought the proceeding before us. Over the objection of both the applicants and the staff,[2] on April 30 the Board granted the FUA petition and accepted 10 of its 27 contentions for litigation. LBP-81-11, 13 NRC 420. Dissatisfied with that result, those parties have appealed under 10 CFR 2.714a. FUA urges affirmance.[3]

I.

No one disputes that, as the Licensing Board determined, FUA has satisfactorily demonstrated the requisite standing to intervene. On that score, its petition is supported by the affidavits of several of its members containing averments that they (1) reside, work and engage in outdoor recreational activities in the vicinity of the Summer site; and (2) have authorized FUA to represent their interests through participation in this proceeding. That is plainly sufficient to satisfy the interest requirements of 10 CFR 2.714(a). See *Houston Lighting and Power Co.* (Allens Creek

[2]Neither Mr. Bursey nor South Carolina took a position on the controversy.

[3]No appeal has been, or could be, prosecuted by FUA from the rejection of the remaining 17 contentions. This is because the Commission's Rules of Practice "do not permit a person to take an interlocutory appeal from an order entered on his intervention petition unless that order has the effect of denying the petition in its entirety". *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-585, 11 NRC 469, 470 (1980), and authorities there cited.

At the conclusion of its brief in support of the grant of intervention, FUA requested oral argument. Such requests are addressed to the discretion of this Board and will be granted only if at least one member votes in favor of it. 10 CFR 2.763; Appendix A to 10 CFR Part 2, Section IX(e). In this instance, the Board unanimously concluded that the parties' positions on the issues presented by the appeals have been adequately developed in the briefs and that oral argument would not be helpful.

Nuclear Generating Station, Unit 1), ALAB-535, 9 NRC 377, 389-400 (1979).

The controversy focuses instead upon the Licensing Board's treatment of the question whether FUA nevertheless should be denied intervention because of the extreme belatedness of its petition and the imminence of the evidentiary hearing. As the Board correctly recognized, in resolving that question it was required to look to the five factors which 10 CFR 2.714(a) mandates be balanced when a belated petition is at hand:

(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.

In its decision, the Board discussed each of these factors in turn. LBP-81-11, *supra,* 13 NRC at 423-428. Its ultimate conclusion was that, collectively, the factors justified allowing the eleventh hour introduction of some, but not all, of the FUA contentions and, thus, supported the grant of intervenor status to the organization. *Id.* at 428. FUA was cautioned, however, that it must "take the proceeding as it currently stands * * *". *Id.* at 423.

It is well-settled that the appellate review of licensing board application of the five factors is governed by the "abuse of discretion" standard. See, *e.g., Nuclear Fuel Services, Inc.* (West Valley Reprocessing Plant), CLI-75-4, 1 NRC 273, 275 (1975); *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-420, 6 NRC 8, 13 (1977); *Project Management Corp.* (Clinch River Breeder Reactor Plant), ALAB-354, 4 NRC 383, 389, 390 (1976), and cases there cited. But it is equally clear that this standard does not foreclose our close scrutiny of the factual and legal ingredients of the analysis underlying the board's ultimate conclusions. ALAB-420, *supra;* ALAB-354, *supra; Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 2), ALAB-384, 5 NRC 612 (1977). And we think that the obligation to undertake such an examination is particularly apparent in the circumstances of this case.

As will be discussed in greater detail *infra,* the Licensing Board did not find that FUA was warranted in waiting until March 1981 before seeking to intervene. As also will be seen, our own appraisal of the record confirms that FUA's tardiness was manifestly unjustified. This being so, the validity of the grant of the petition so close to the start of the hearing perforce hinges upon whether a compelling showing has been made by FUA on the other four factors. Once again, by March 9 when the hearing date was set (if not long before), the applicants and the staff had every right to assume that both the issues to be litigated and the participants had been established with finality. Simple fairness to them — to say nothing of the public interest requirement that NRC licensing proceedings be conducted in an orderly fashion — demanded that the Board be very chary in allowing one who had slept on its rights to inject itself and new claims into the case as last-minute trial preparations were underway.

For the reasons which follow, we are persuaded that FUA's showing on the controlling factors fell fatally short of what might have provided a sufficient foundation for a discretionary allowance of tardy intervention. Accordingly, the April 30 order cannot stand.

II.

For the purposes of its analysis, the Licensing Board divided FUA's contentions into two groups. The first consisted of the ten contentions which were ultimately admitted to the proceeding; they broadly dealt with corporate management (Nos. 1, 2, 27) and emergency planning (Nos. 7-13). The second group embraced the 17 rejected contentions — covering such widely diverse subject matter as financial qualifications (Nos. 3 and 4); seismicity (Nos. 5 and 6); steam generator tube integrity (No. 14); quality control (No. 15); diesel generator reliability (No. 16); class 9 accidents (No. 17); anticipated transients without scram (No. 18); license condition implementation (No. 19); storage and transportation of spent fuel (Nos. 20-22); health effects of radiation releases during normal plant operation and as a result of the uranium fuel cycle (No. 23); systems interactions (No. 24); control room design (No. 25); and hydrogen control (No. 26).

A. In its decision, the Licensing Board summarized the variety of reasons assigned by FUA for the failure to have sought intervention on any issue at a much earlier date. LBP-81-11, *supra,* 13 NRC at 422. In large measure, those reasons were found insubstantial. *Id.* at 423. Nevertheless, the Board concluded that, in light of the revisions made in the Commission's criteria for emergency planning following the Three Mile Island accident, FUA had good cause to wait *until the middle or latter part of 1980* before filing its contentions on that subject. "[B]ecause of the Commission's focus on management capability in the post-TMI era", the Board reached a similar conclusion with regard to "the delay in filing the management capability contentions". *Id.* at 423-424.

We need not determine here whether the Board was right in that view. Be that as it may, the post-TMI events cannot possibly serve to justify FUA's election to wait until the end of March 1981 to file its petition. In this connection, as the Board itself emphasized, the final rule establishing new and specific standards for on-site and off-site radiological emergency plans was published on August 19, 1980. 45 *Fed. Reg.* 55402. And we have been pointed to no more recent developments in the corporate management area which might be taken as having first triggered FUA's obligation to put forward its concerns on that subject.[4]

B. The Board below nevertheless found the "good cause" factor "to be of almost no weight (or of slight weight against petitioner) in deciding upon the intervention with regard to the corporate management and emergency planning issues". 13 NRC at 424. Central to this finding was the Board's articulated belief that no other party to the proceeding had been disadvantaged by the filing in March (rather than considerably earlier) and that the progress of the proceeding would not be delayed. *Id.* at 424.[5]

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[4]It appears from the petition to intervene (at p. 4), that FUA had assumed prior to mid-February 1981 that "its interests were being represented, to some extent, by" Mr. Bursey. Only then, when it was given reason to doubt the continuing validity of that assumption, did FUA undertake "an immediate and thorough inquiry into the status of this proceeding and its rights and remedies". As the Board below correctly observed, that excuse is not acceptable. See *Duke Power Co.* (Cherokee Nuclear Station, Units 1, 2 and 3), ALAB-440, 6 NRC 642, 644-45 (1977).

Apart from stressing its misplaced reliance upon the Bursey intervention, in its appellate brief (at p. 3) FUA reiterated its complaint below respecting the asserted lack "for several years" of a "properly managed" local public document room. Whether or not this assertion has factual substance, it too provides an inadequate explanation for the March filing of the inervention petition. As FUA acknowledges (Br. p. 2), its representatives attended a November 25, 1980 prehearing conference in this proceeding. At that time, if not before, it had a full opportunity to acquire whatever information may have been necessary to undergird its petition. Yet it waited another four months — as it admits (Br. pp. 2-3), because of the Bursey intervention.

[5]It is not entirely clear from an earlier statement in the Board's discussion on this point whether the Board might have thought that these considerations bear upon the *existence* of good cause for the tardy filing in March, as opposed to the possible significance of the absence of such cause. We have specifically in mind the observation that "[h]ad that added delay in filing disadvantaged any parties other than petitioner itself (by circumscribing its prehearing activities), or delayed the proceedings, we might find a lack of good cause". 13 NRC at 424.

Obviously, whether there is "good cause" for a late filing depends *wholly* upon the substantiality of the reasons assigned for not having filed at an earlier date. For their part, the consequences of the tardiness are to be looked at in connection with the other factors (most particularly the fifth one, dealing with delay and the broadening of the issues). We shall assume that the Licensing Board recognized this consideration and that its finding quoted in the text was intended to mean only that the "good cause" factor did not weigh heavily against FUA in the overall assessment of the delinquent petition.

We disagree with the Board on both scores. It seems manifest to us that the introduction of FUA and its accepted contentions into the proceeding less than two months before the scheduled trial date has prejudiced other parties. Further, a delay in the progress of the proceeding is not merely a theoretical possibility but rather a very likely proximate result of the belated intervention.[6]

1. Had FUA sought and obtained intervention in a more timely fashion, the applicant and the staff could have instituted discovery against it without jeopardizing the present commencement date for the evidentiary hearing. The Licensing Board acknowledged that fact but went on to express the opinion that "discovery would not have benefitted them on the issues we are admitting". This is said to be so because FUA "has made full disclosure in its supplemental petition of the bases for its contentions, including the names or offices of its potential witnesses to the extent we are admitting its contentions, for the Board will not allow additional witnesses". 13 NRC at 425.

The principal difficulty with that line of reasoning is that it ascribes too limited a role to the discovery process. Parties to a proceeding are entitled to obtain in advance of hearing much more than simply a summary statement of the bases for their adversaries' claims and some identification of potential witnesses whose testimony might support those claims. Rather, as we had recent occasion to stress, "[i]n modern administrative and legal practice, pretrial discovery is liberally granted to enable the parties to ascertain the facts in complex litigation, refine the issues, and prepare adequately for a more expeditious hearing or trial". *Pennsylvania Power and Light Co.* (Susquehanna Steam Electric Station, Units 1 and 2), ALAB-613,

[6]At the April 7-8 prehearing conference, the Licensing Board announced that, if not completed during the June 22-July 3 period, the evidentiary hearing would resume on July 13 and continue through July 24 (Tr. 666). This was later confirmed in a May 14 "notice of scheduling of evidentiary hearing".

On May 12, FUA filed a "motion for continuance" in which it called attention to the fact that FUA and its representatives are also parties to a rate proceeding pending before the South Carolina Public Service Commission. That proceeding (involving one of the present applicants) is scheduled to commence on July 13. Asserting that it lacked the resources to appear simultaneously in both proceedings, FUA asked that, unless the state proceeding were rescheduled, the July 13 hearing session in the NRC proceeding be postponed.

On the date of the filing of FUA's brief with us (May 20), the motion was pending before the Licensing Board (and it still is). Yet, FUA did not refer to it in that brief. Particularly because one of the signatories was a member of the Bar (see fn. 12, *infra*), we find the omission disturbing. Clearly, were the motion to be granted, there might well be a delay in the completion of the evidentiary hearing as a direct consequence of FUA's intervention. This being so, FUA should have acknowleged the existence of the pending motion in the course of its argument (Br. pp. 11-12) that the late intervention would cause no "relevant" or "unproductive delay".

12 NRC 317, 322 (1980), quoting from *Pacific Gas and Electric Co.* (Stanislaus Nuclear Project, Unit 1), LBP-78-20, 7 NRC 1038, 1040 (1978). In the same vein, the Supreme Court has noted that, as a result of the availability of discovery, "[t]he way is now clear, consistent with recognized privileges, for the parties to obtain the fullest possible knowledge of the issues and facts before trial". *Hickman* v. *Taylor*, 329 U.S. 495, 501 (1947).

The short of the matter is that, because of FUA's inexcusable tardiness, the other parties to the proceeding have been effectively deprived of the opportunity to obtain "the fullest possible knowledge" of what FUA proposes to adduce in support of its contentions. To be sure, the Board directed that "the parties cooperate in informal discovery" with respect to the "applicant's and [s]taff's evolving positions on emergency planning". 13 NRC at 425. But, irrespective of precisely what the Board may have had in mind in that regard, it seems reasonably apparent that the contemplation was *not* that either the applicants or the staff would undertake to determine the metes and bounds of FUA's case by means of interrogatories, depositions, document discovery and requests for admissions. In any event, time would have not permitted such an exploration — at least so long as the June 22 hearing date remained inviolate.[7]

2. Equally unpersuasive is the Licensing Board's treatment of the impact of the tardy intervention upon the ability of the applicants and the staff to seek summary disposition of one or more of FUA's admitted contentions. The Board opined that neither the corporate management nor the emergency planning issues are now susceptible of summary disposition. 13 NRC at 426. By that, the Board presumably meant that a trial could not be entirely avoided on those issues. But it scarcely follows that *none* of the *specific* claims set forth in FUA's numerous contentions would be disposable summarily — in part if not in whole.[8] Thus, by countenancing FUA's intervention at such a late date that pretrial resort both to discovery and to summary disposition procedures became practical impossibilities, the Board has created the substantial danger that hearing time will be unnecessarily expended and, thus, wasted.

3. The Licensing Board reasoned that, because "the corporate management and emergency planning issues had already been admitted to the proceeding (by Board question or intervenor [*i.e.*, Bursey] contention)", the issues would not be broadened by FUA's admission to the proceeding on those subjects. 13 NRC at 425. We cannot agree.

Only one of Mr. Bursey's contentions even remotely brings into question the applicant's managerial capabilities: in contention A2, that intervenor

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[7]In this connection, it is our understanding that the prefiled testimony was due on May 28.
[8]Some of those specific claims are summarized *infra*, pp. 891-892.

asserted that the applicants lack the financial qualifications to operate anc decommission the facility both safely and in compliance with NRC regulations. For its part, the Licensing Board manifested at a November 25 1980 prehearing conference its "concern" that the proposed addition of the South Carolina Public Service Authority as a co-owner of the facility migh "compromise management responsibility for the public health and safety" See December 30, 1980 memorandum and order (unpublished), at pp. 6-7

The FUA contentions go well beyond those matters, into applicants' competence to operate a nuclear facility. Contention 1, for example, asserts broadly that the "overall corporate managment of the Applicant is sufficiently inexperienced in the operations of a nuclear power facility and is generally deficient in management abilities essential to the safe operation of a nuclear power plant or properly to respond under accident conditions". Contention 2 challenges the adequacy of the "hands on" experience of the applicants' "reactor operator staff".[9] And contention 27 disputes the adequacy of the applicants' technical and management resources to fulfill new regulatory requirements imposed as a consequence of the Three Mile Island accident.

Insofar as emergency planning is concerned, Mr. Bursey's single contention in that area (A8) focused upon the applicants' asserted lack of adequate preparations for "the implementation of [its] emergency plan in those areas where the assistance and cooperation of state and local agencies are required". Our examination of the record does not disclose that the Board has undertaken on its own to raise additional emergency planning issues. Yet the FUA contentions manifestly have done precisely that. Thus, it is claimed in various subparts of contention 7 that, among other things, the applicants' plan does not meet minimum staffing requirements; that realistic estimates of evacuation times have not been developed; that adequate means have not been provided for the protection of those without access to motor vehicles; that no provisions have been made for the distribution and use of "radioprotective" drugs; that on-site emergency first aid capability is inadequate; and that the applicants' meteorological monitoring equipment does not satisfy NRC requirements. The other FUA emergency planning contentions (8 through 13) likewise contain assertions which broaden significantly what Bursey contention A8 called upon the

[9]At the April 7-8 prehearing conference, the Board below alluded to a "question" raised by the Advisory Committee on Reactor Safeguards in the corporate management "area" (Tr. 478-79). The question was not there identified more precisely. From the April 30 order, 13 NRC at 427, it appears that the question dealt in part with the applicants' "hands-on operating experience". What the Board left unclear was whether it was then raising that question itself. If not, the ACRS concern necessarily will have to receive staff attention before an operating license is issued. See pp. 895-896, *infra*.

applicants and the staff to confront in their prefiled testimony and at the hearing.

The Licensing Board undoubtedly was aware of the expansive reach of the FUA contentions. It is a fair inference, therefore, that the Board thought that, for the purposes of Section 2.714(a), a belated petition can be held to "broaden the issues" only if it introduces an entirely new subject matter. But such an interpretation is at odds with the commonly understood meaning of "broaden", *i.e.*, "to extend the limits of".[10] And there is no reason to assume that the Commission had any other meaning in mind. To the contrary, in assessing this factor in *West Valley*, CLI-75-4, *supra*, 1 NRC at 276, the Commission emphasized the fact that "substantially identical" issues to those presented in the late petition had been raised by other parties. As has been seen, FUA's contentions are far from "substantially identical" to either those of Mr. Bursey or the Board's management responsibility question.

C. We now turn to the factor which the Licensing Board thought weighs "most heavily" in FUA's favor with respect to its corporate management and emergency planning contentions. According to the Board, FUA can be expected to make a substantial contribution to the development of a sound record on those subjects. Its explanation for this conclusion was contained in one sentence: "As is apparent from FUA's pleadings and from the general discussion at the prehearing conference, petitioner's members have become well versed [on corporate mangement and emergency planning matters], independently of any intention of intervening in this proceeding, through their participation in rate-making proceedings and in the ongoing emergency planning". 13 NRC at 426.

In addition, while acknowledging that it "perhaps" did not constitute grounds for allowing FUA intervention, the Board recorded its conviction that Mr. Bursey was incapable of making a significant contribution to the development of the record. The Board pointed to that intervenor's manifested "inability to effectively manage his case" and suggested that it could not count on assistance from him in the resolution of the corporate management question that it had raised (although "valuable assistance" on that question was to be expected of the staff). 13 NRC at 426-427.

As we see it, the Board's perception of Mr. Bursey's abilities and his likely contribution to the proceeding could not possibly serve as justification for allowing FUA to come into the proceeding at the last moment. It is often the case that one or another of the parties to a proceeding will give the presiding board legitimate cause to question its ability to make an effective

[10] *Webster's Third New International Dictionary* (1971), at p. 280.

presentation on the issues in controversy. When confronted with such a situation, the board may well have to take a more active role in the proceeding itself. For example, it may find it necessary to undertake its own interrogation of the witnesses.[11] This, it seems to us, is the appropriate course to follow — rather than opening the door, as the hearing date approaches, to another would-be party which seeks not merely to participate in the record development on the then-existing matters in controversy, but also to expand the issues to be heard.

In appraising the ruling below on the factor at hand, we accordingly eschew any comparison of FUA's seeming capabilities with those of Mr. Bursey. Instead, our inquiry is restricted to whether the record supports the Licensing Board's conclusion that FUA's likely contribution is of sufficient magnitude to favor strongly allowing its intervention at this time.

1. FUA is represented in this proceeding primarily by Dr. John C. Ruoff.[12] According to his affidavit appended to the intervention petition, Dr. Ruoff possesses a PhD in history and is a self-employed "research consultant to a variety of nonprofit and community-based organizations". In recent years (1979-80), he participated as an intervenor on his own behalf in a rate proceeding conducted before the South Carolina Public Service Commission, which involved the lead applicant (South Carolina Electric and Gas Company). "[T]hrough that proceeding", it is averred, he "became educated and informed about the organization, management and operation of the Applicant and the design, construction, and plans for the operation" of the Summer facility. Further, his participation in the programs of FUA over the past year has enabled him to "become educated on the subject of the design and operation of nuclear power plants and the probable effects of [Summer] operation".

2. At the April 7-8, 1981 prehearing conference which, *inter alia*, addressed the FUA petition, Dr. Ruoff told the Licensing Board that he did not have an available witness to support the management capability contentions in that petition (Tr. 467). Instead, it is his apparent intention to restrict himself to the cross-examination of applicant (and possibly staff)

[11]See 10 CFR 2.718(g). See also, *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-283, 2 NRC 11, 20 (1975), where "the Board made a determined effort to insure that the issues were thoroughly explored".

[12]On the second day of the April 7-8 prehearing conference, Robert Guild, Esquire, of the Bar of South Carolina entered a special appearance for the purpose of addressing on FUA's behalf the legal issues raised by the untimeliness of the intervention petition (Tr. 494). Along with Dr. Ruoff, Mr. Guild also signed the brief which has been submitted to us on the instant appeals. It appears from FUA's May 12 motion for a continuance (see fn. 6, *supra*) that Mr. Guild's participation at the evidentiary hearing would be restricted to providing FUA with assistance on any legal issues which may arise. We therefore assume that Dr. Ruoff would be solely responsible for the examination of witnesses and anything else required to develop FUA's position on the substantive issues.

witnesses (Tr. 477, 479, 482, 657-58). And, as previously noted (p. 888, *supra*), in its April 30 order the Board made it plain that FUA will not be permitted to add witnesses at this point.

Without far more particularization of his experience and knowledge than is set forth in his affidavit or was provided at the April 7-8 conference, we are unable to discern any basis for concluding that Dr. Ruoff's participation as a cross-examiner is imperative to the development of a comprehensive record on the applicants' management capability. While his involvement in the state rate proceeding may well have acquainted him with details of the financial structure of the lead applicant, it is not immediately obvious why it would have provided unusual insight into that company's competence to operate a large nuclear facility (as raised by FUA's contentions 1, 2 and 27). Nor was the Board below given reason for confidence that such insight might have been supplied by Dr. Ruoff's unspecified role in unspecified FUA programs.

We do not intimate, of course, that Dr. Ruoff would be incapable of making *any* contribution through cross-examination of applicant or staff witnesses. All that we determine, or need decide, is that FUA's showing on the "record development" factor was not strong enough to warrant, standing alone, the grant of its inexcusably and materially late petition. In this connection, as noted above it is both the right and the responsibility of the Licensing Board to examine witnesses itself, if necessary in the interest of insuring that a proper record is compiled on all matters in controversy (or raised by it *sua sponte*). We take official notice that the two technical members of the Board below have served on the Licensing Board Panel for nine and eight years respectively, during which period each has sat on numerous licensing proceedings. That being so, it surely does not demean Dr. Ruoff's credentials to suggest that the Board is at least as well-equipped to pursue any relevant lines of inquiry as might be Dr. Ruoff on the basis of his participation in a single rate proceeding and less than one year's association with a community-based organization.

3. FUA does propose to present one or more witnesses in support of its emergency planning contentions. At the April 7-8 prehearing conference, Dr. Ruoff made specific reference to Dr. Janet Greenhut and Marlene Bowers Andrews (Tr. 592-96). Dr. Greenhut is a physician and FUA member. Dr. Ruoff informed the Board that, because he had not been able to obtain "as yet" an expert on radiological health, he might call upon her to testify. He noted that "Dr. Greenhut has done some research into that area with some medical literature" (Tr. 596). Ms. Andrews was described by Dr. Ruoff as "an expert in psychology who has been doing work on nuclear emergencies, radiological emergencies" (Tr. 595). She was said to have agreed to appear as a FUA witness (*ibid.*).

Apart from those named individuals, Dr. Ruoff expressed an interest in calling "the emergency preparedness people from the four county area, the four counties within the plume exposure pathway, emergency planning zone" (Tr. 593). He conceded, however, that he had not obtained a commitment from any such persons to testify on FUA's behalf (*ibid.*). He also reaffirmed the assertion in the FUA petition (as part of the basis for contention 7) that FUA has members (including himself) who possess "unique" knowledge of the demography, roads, traffic patterns and topography of the area surrounding the Summer site (Tr. 596). It is unclear, however, whether he proposed to produce the testimony of some of those members and it is even more doubtful that the Board below would now permit him to add them to the witness list.[13]

What appears from these disclosures is no more than that FUA may be in a position to assist the development of the record on a few — but well short of all — of the numerous assertions made in its emergency planning contentions. Just how significant that assistance might be is problematic. It depends, of course, on the state of the knowledge of FUA's proposed witnesses on the subjects they would address. Dr. Greenhut and Ms. Andrews are the only potential witnesses who have been specifically identified. What the Board was told about their qualifications and possible testimony was plainly too sparse to permit an informed judgment regarding their likely contribution.[14]

D. We have no quarrel with the Licensing Board's conclusions respecting the remaining two factors.[15] 13 NRC at 427-428. Given the Board's appraisal of the manner in which Mr. Bursey is carrying forward his own intervention, there is little reason to suppose that he would

[13]FUA contention No. 13 is concerned with off-site radiation monitoring. In a colloquy with the Board, Dr. Ruoff noted that the derivation of that contention was discussions FUA had had with the Union of Concerned Scientists. He conceded that he had not obtained a witness to support the contention. He also acknowledged that the contention did not parallel any of Mr. Bursey's contentions. Tr. 621.

[14]At several points both in its petition and during the prehearing conference, FUA made mention of various employees of the lead applicant who assertedly would shed some light on the corporate management and emergency planning questions raised by the petition. In a May 13, 1981 order (at p. 9), the Licensing Board directed that those employees be made available at the hearing for FUA examination. We do not deem them to be FUA witnesses and, further, find no basis for conjecture on how fruitful FUA's examination of them might prove to be.

In the same order (at pp. 9-11), the Board ruled that FUA also would be permitted to cross-examine on the issues raised by Mr. Bursey's contentions — which encompass several subjects (*e.g.*, seismicity) apart from corporate management and emergency planning. There is an equal lack of basis for an informed prediction respecting the utility of FUA's exercise of that privilege.

[15]*I.e.*, the availability of other means whereby the petitioner can protect its interest and the extent to which other parties will represent that interest.

adequately represent FUA's interest. Moreover, once again, the FUA and Bursey claims differ in significant measure. And while the applicants and the staff point out that FUA members might choose to make limited appearance statements, we are not persuaded that, in the circumstances of this case, their interest would be fully protected by such restricted participation in the proceeding. Nor do we perceive other means which might serve that purpose.

But, as the Licensing Board itself correctly observed, those factors "are given relatively lesser weight than the other factors". 13 NRC at 427. Indeed, it is most difficult to envisage a situation in which they might serve to justify granting intervention, after the hearing date was set, to one who (1) is inexcusably late; (2) seeks to expand materially the scope of the proceeding; and (3) offers, at best, a marginal showing with respect to its ability to make a truly significant, substantive contribution. In the present context, for the very reason that, as FUA puts it (Br. p. 9), "[t]his proceeding represents the best forum for the protection of [its] interest in health and safety matters regarding the Summer Nuclear Station", the organization should have filed its intervention petition at a much earlier date. By instead remaining on the sidelines while the proceeding moved closer and closer to trial, it voluntarily assumed the precise risk which has now materialized: that its participation in the proceeding could no longer be sanctioned without destructive damage to both the rights of other parties and the integrity of the adjudicatory process itself.

E. For the foregoing reasons, the denial of the FUA petition was mandated. Although understandably hesitant to deprive FUA of the opportunity to ventilate its seemingly genuine concerns at the hearing which is about to commence, in the totality of circumstances the Licensing Board simply had insufficient justification under the Commission's Rules of Practice for allowing this crucially tardy intervention.

It does not follow from FUA's exclusion from the proceeding that its concerns perforce will be ignored in the licensing of this reactor. Insofar as they overlap either matters placed in controversy by Mr. Bursey or issues raised by the Board *sua sponte* (see 10 CFR 2.760a), it will be the Board's responsibility to require their adequate evidentiary exploration. To the extent that they go beyond the bounds of the hearing as fixed prior to the belated FUA intervention attempt, under the long-prevailing regulatory scheme these concerns fall within the province of the staff. In all events, an operating license may not issue unless and until this agency makes the findings specified in 10 CFR 50.57 — including the ultimate finding that such issuance "will not be inimical to * * * the health and safety of the public". As to those aspects of reactor operation not considered in an adjudicatory proceeding (if one is conducted),[16] it is the staff's duty to insure the existence of an adequate basis for each of the requisite Section 50.57 determinations.

Insofar as it granted the intervention petition of Fairfield United Action, the April 30, 1981 order of the Licensing Board, LBP-81-11, 13 NRC 420, is *reversed* and the cause is *remanded* with instructions to deny that petition as untimely.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

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[16]On the operating license level, a hearing is required only in response to a successful petition for leave to intervene and request therefor. Section 189a. of the Atomic Energy Act of 1954, as amended, 42 U.S.C. 2239(a).

Ms. Kohl, concurring:

I join fully in the Board's opinion. I take this opportunity only to make two brief additional points.

1. FUA's papers, particularly those prepared by Dr. Ruoff and filed before the Licensing Board, represent an impressive — albeit unsuccessful — effort to participate in and contribute to this important proceeding. Given the quality of these pleadings and the asserted interest of its members in the Summer facility, it is especially difficult to understand why FUA, which was incorporated in early September 1980, waited over six months before taking any formal action in furtherance of that interest.[1] None of the reasons FUA offered for the delay — set forth by the Licensing Board, 13 NRC at 422 — proves persuasive. Indeed, its inaction is inconsistent with its professed concern about this plant and this proceeding.

2. One means does exist, however, by which FUA can contribute to this proceeding without being afforded party status. The organization can furnish financial, technical, legal, or other assistance to the sole existing intervenor, Mr. Bursey. *Virginia Electric and Power Co.* (North Anna Station, Units 1 and 2), ALAB-289, 2 NRC 395, 399 (1975). This, of course, provides no fully satisfactory substitute for direct participation (see p. 894-895 *supra*). But if FUA is sincere in its interest — and there is no reason to doubt that it is — it will grasp this opportunity enthusiastically.[2]

[1]Even after FUA's representatives attended a November 25, 1980, prehearing conference, the organization took no immediate action to formalize its involvement. See fn. 4, *supra*.

[2]I note in this connection that FUA's counsel, Mr. Guild (see fn. 12, *supra*), at one time was to have appeared in this proceeding as a witness for Mr. Bursey on this Contention A2 (May 13, 1981, Order at pp. 3, 11-12). Thus, there is an ostensible connection between FUA and the intervenor that would facilitate an offer (and acceptance) of assistance from the former.

Cite as 13 NRC 898 (1981) **ALAB-643**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Christine N. Kohl

In the Matter of **Docket No. 50-395 OL**

SOUTH CAROLINA ELECTRIC AND GAS COMPANY, *et al.*
(Virgil C. Summer Nuclear Station, Unit 1) **June 15, 1981**

The Appeal Board denies an application filed pursuant to 10 CFR 2.788 for a stay of the effectiveness of the Board's earlier decision (ALAB-642), which reversed a Licensing Board's order granting an untimely intervention petition, pending the filing and disposition of a petition for Commission review of the Appeal Board's decision.

RULES OF
PRACTICE: STAY PENDING APPEAL

In determining whether to grant an application for a stay, boards must consider the following criteria:

(1) Whether the moving party has made a strong showing that it is likely to prevail on the merits;
(2) Whether the party will be irreparably injured unless a stay is granted;
(3) Whether the granting of a stay would harm other parties; and
(4) Where the public interest lies.

10 CFR 2.788(e).

APPEARANCES

Dr. John C. Ruoff, Jenkinsville, South Carolina, for the movant Fairfield United Action.

Messrs. Joseph B. Knotts, Jr., and Dale E. Hollar, Washington, D.C., for the South Carolina Electric and Gas Company *et al.*

Mr. Steven C. Goldberg for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

In ALAB-642, 13 NRC 881 (June 1, 1981), we reversed the April 30, 1981 Licensing Board order[1] which granted the untimely petition of Fairfield United Action (FUA) for leave to intervene in this operating license proceeding. FUA now applies under 10 CFR 2.788 for a stay of the effectiveness of ALAB-642 pending the filing and disposition of a petition for Commission review of that decision. Applying the four criteria set forth in Section 2.788 (e),[2] we deny the application. Because, however, the evidentiary hearing is due to commence one week from today (June 22), as an accommodation to FUA we are transmitting its papers, together with the responses filed by other parties,[3] to the Commission for such action, if any, as it may deem appropriate.[4]

[1]LBP-81-11, 13 NRC 420.

[2]That subsection reads:

> In determining whether to grant or deny an application for a stay, the Commission, Atomic Safety and Licensing Appeal Board, or presiding officer will consider:
>
> (1) Whether the moving party has made a strong showing that it is likely to prevail on the merits;
>
> (2) Whether the party will be irreparably injured unless a stay is granted;
>
> (3) Whether the granting of a stay would harm other parties; and
>
> (4) Where the public interest lies.

These are the same factors which have long governed the grant or denial of stays in the federal courts. See *Virginia Petroleum Jobbers Ass'n* v. *FPC*, 259 F.2d 921, 925 (D.C. Cir. 1958).

[3]Both the operating license applicants and the NRC staff oppose the grant of a stay.

[4]We wish to make clear that our ruling on the stay application is *not* being referred to the Commission; *i.e.*, we are not affirmatively calling upon the Commission to review that ruling. Rather, we are simply giving recognition to the imminence of the evidentiary hearing. It would be difficult, if not impossible, for FUA formally to renew its stay application before the Commission in time to enable full consideration and disposition by that body.

1. In ALAB-642, we noted that the appellate review of licensing board action on belated intervention petitions is governed by the "abuse of discretion" standard. 13 NRC at 886. Asserting that it is likely to succeed on the merits of its petition for Commission review, FUA urges that we nevertheless failed to apply that standard. This is said to follow from the absence of a finding by us that the Licensing Board abused its discretion in granting the FUA petition. Stay application, pp. 2, 5.

True enough, ALAB-642 does not contain an explicit finding to that effect. But it leaves no room for reasonable doubt regarding our conclusion that, on the record before it, the Licensing Board could not allow FUA's eleventh hour intervention as a discretionary matter. See, *e.g.*, 13 NRC at 887, 895. Put another way, a licensing board simply has no latitude to admit a new party to a proceeding as the hearing date approaches in circumstances where (1) the extreme tardiness in seeking intervention is unjustified; (2) the certain or likely consequence would be prejudice to other parties as well as delay in the progress of the proceeding, particularly attributable to the broadening of issues; and (3) the substantiality of the contribution to the development of the record which might be made by that party is problematic.

We need not rehearse the bases assigned in ALAB-642 for our determination that each of those circumstances is here present. Suffice it to note that nothing now offered by FUA prompts our reassessment of the matter.

The stay application is silent with regard to our finding of a lack of good cause for the tardiness of the intervention petition. Nor does it address the considerations which underlay our conclusions regarding prejudice to other parties and delay.[5] Rather, FUA focuses almost exclusively (stay

In this connection, 10 CFR 2.788(f) provides that "[a]n application to the Commission for a stay of a decision or action by an *** Appeal Board will be denied if a stay was not, but could have been, sought before the Appeal Board". It was doubtless this provision which prompted the filing of the stay application with us in the first instance. Unfortunately, subsection (f) sheds no illumination on what might constitute circumstances in which a stay need not be sought initially from this Board. Presumably, however, the urgency of the perceived need for a stay was not thought by the Commission to be such a circumstance; had it been, the subsection likely would have so indicated.

It appears from the stay application (at pp. 3-4) that FUA took our statement (13 NRC at 888) that "delay in the progress of the proceeding [was] a very likely proximate result of the belated intervention" as resting solely on the then pending FUA "motion for continuance" discussed in accompanying footnote 6. In this connection, we are told that the South Carolina Public Service Commission has now changed the date of its proceeding, with the consequence that the continuance motion has become moot.

In actuality, however, the conclusion respecting delay was founded principally upon other considerations: that the FUA intervention would broaden the issues significantly and, because application, pp. 3, 4) upon its claimed ability to contribute to the development of the record.

On that score, FUA relies in large measure on events subsequent to its admission to the proceeding on April 30 — most particularly, the submission of the prepared testimony of its two proposed witnesses (Dr. Greenhut and Ms. Andrews)[6] and the filing of responses in opposition to motions for summary disposition directed to contentions of intervenor Brett Allen Bursey.[7] But even if it were proper for us to take account of this new material for present purposes,[8] the result reached in ALAB-642 necessarily would remain unchanged. Although FUA's recent filings may well bear out the observation in the concurring opinion regarding the quality of its earlier pleadings,[9] we are left unpersuaded that its contribution at the evidentiary hearing would be of such magnitude as to tip the overall balance in favor of permitting it to enter the proceeding at this juncture.

2. Turning to the second stay criterion, it is plain that FUA will not be *irreparably* injured if a stay is denied. Were the Commission ultimately to reverse ALAB-642 and order FUA reinstated as a party, the necessary consequence would be that the evidentiary record (if closed by then) would have to be reopened to enable FUA's participation. FUA opines (stay application, p. 5) that the Commission likely would be loath to grant that relief. We decline to indulge in any such conjecture. No matter what might be the posture of the proceeding at the time, we must and do assume that the petition for review will receive a fair appraisal and that, should the Commission disagree with our decision, it will have no hesitancy to provide FUA with the full remedy to which it would thereupon become entitled.

3. FUA maintains (stay application, pp. 6-7) that the grant of a stay would occasion "little" harm to other parties. We think otherwise. Among other things, it would require those parties to devote time and resources at the hearing to the new and numerous issues which FUA seeks to inject into the proceeding. Should the Commission not disturb ALAB-642, this expenditure would be irretrievable.

4. FUA's argument on the final criterion (public interest) is founded on the premise that its full participation in the proceeding is required in order to assure a fully-developed record "on important issues of health and safety". Stay application, p. 8. We have previously noted our nonaccep-

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of its lateness, would foreclose resort to summary disposition procedures on those issues. See 13 NRC at 889-891. The stay application does not dispute that this is so.

[6]See ALAB-642, 13 NRC at 893.

[7]Mr. Bursey was admitted to the proceeding in 1978. See LBP-78-6, 7 NRC 209.

[8]Our appraisal of the correctness of a licensing board's determination on the various factors to be considered in passing upon a late petition (see ALAB-642, 13 NRC at 885) perforce must be founded upon what was before that board when the determination was made.

[9]13 NRC at 897.

tance of that premise. 13 NRC at 892-894, 895. Further, as the concurring opinion points out (*id.* at 897). "FUA can contribute to this proceeding without being afforded party status" by furnishing "financial, technical, legal, or other assistance" to intervenor Bursey. Although the stay application does not allude to that fact, FUA may yet elect to "grasp this opportunity enthusiastically" (*ibid.*).

Application for a stay of ALAB-642 *denied.*
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Richard S. Salzman, Chairman
Dr. John H. Buck
Dr. W. Reed Johnson

In the Matter of **Docket Nos. 50-275 OL**
50-323 OL
(Seismic Proceeding)

PACIFIC GAS AND ELECTRIC COMPANY
(Diablo Canyon Nuclear Power Plant, Units 1 & 2) **June 16, 1981**

Following an evidentiary hearing on seismic issues, the Appeal Board affirms (except for security plan issues which are before another Board), two partial initial decisions rendered by the Licensing Board on the Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Plant (LBP-79-26, 10 NRC 453 (1979); and LBP-78-19, 7 NRC 989 (1978).

RULES OF PRACTICE: STANDING TO APPEAL

In Commission practice as in judicial proceedings, only a party aggrieved may appeal.

RULES OF PRACTICE: BRIEFS

Issues not briefed may be deemed waived.

RULES OF PRACTICE: ADMINISTRATIVE FAIRNESS

The resolution of issues of fact in favor of one side suggests neither bias nor error on the tribunal's part; without more, the appropriate inference is that the evidence of the prevailing party was the more persuasive.

RULES OF PRACTICE: ADMINISTRATIVE FAIRNESS

In administrative hearings as in court cases, rulings and findings made in the course of a proceeding are not in themselves sufficient reasons to believe that the tribunal is biased for or against a party.

REGULATORY GUIDES: STATUS

Regulatory guides are advisory rather than obligatory; they do not lay down mandatory directives but delineate problem-solving techniques the staff deems acceptable from past experience.

REGULATIONS: INTERPRETATION

Regulations, like statutes, may neither be read in isolation nor interpreted piecemeal.

TECHNICAL ISSUES DISCUSSED:

Seismic design criteria;
Safe shutdown earthquake;
Response spectrum;
Operating basis earthquake;
Plate tectonics;
Length of the Hosgri fault;
Near-field motion characteristics;
Magnitude saturation;
Distance saturation;
Peak ground acceleration;
Effective acceleration;
Focusing;
High stress drop;
Motion on rock versus soil;
Vertical versus horizontal accelerations;
Tau effect;
Torsional excitation;

Soil-structure interaction;
Damping;
Material strength;
Structural ductility;
Combination of loads.
Response of El Centro Power Plant to Imperial Valley Earthquake.

APPEARANCES

Messrs. **Bruce Norton** and **Arthur C. Gehr**, Phoenix, Arizona, and **Malcolm H. Furbush** and **Philip A. Crane, Jr.**, San Francisco, California, for the Pacific Gas and Electric Company, *applicant.*

Messrs. **David S. Fleischaker**, Washington, D.C., and **Joel Reynolds** and **John R. Phillips**, Los Angeles, California, for the San Luis Obispo Mothers for Peace *et al., joint intervenors.*

Messrs. **Herbert H. Brown** and **Lawrence Coe Lanpher**, Washington, D.C., and **J. Anthony Kline** and **Byron S. Georgiou**, Sacramento, California, for the Governor of California, *amicus curiae.*

Messrs. **William J. Olmstead, James R. Tourtellotte, L. Dow Davis, IV,** and **Edward G. Ketchen** for the Nuclear Regulatory Commission staff.

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DECISION

Following contested hearings on the Pacific Gas and Electric Company's application for licenses to operate its Diablo Canyon Nuclear Power Plant, the Licensing Board found the plant adequately designed to withstand any earthquake that can reasonably be expected. LBP-79-26, 10 NRC 453 (Partial Initial Decision of September 27, 1979). The joint intervenors[1] have appealed that "seismic decision." Before we could decide their appeal, however, they moved to reopen the record for new evidence derived from a subsequent major California earthquake. We granted the motion and took that evidence ourselves. ALAB-598, 11 NRC 876 (June 24, 1980). The following decision covers all the seismic issues, whether raised in the initial appeal or at the reopened hearing.[2]

[1]Joint Intervenors are the San Luis Obispo Mothers for Peace (SLOMFP); the Scenic Shoreline Preservation Conference, Inc.; the Ecology Action Club; Sandra A. Silver; Gordon Silver; John J. Forster; and Elizabeth Apfelberg.

[2]The Licensing Board's September 27, 1979 decision also resolved contentions about the risk of aircraft and missiles striking the plant and the adequacy of the plant's "physical security plan." See LBP-79-26, 10 NRC 453, 459-463, 507. No exceptions were taken from the Board's determinations on the former issue; SLOMFP's separate security appeal is before another Board and not covered in this decision. See ALAB-580, 11 NRC 227 (1980).

An earlier partial initial decision by the Board below covered environmental matters. LBP-78-19, 7 NRC 989 (June 12, 1978). No exceptions were taken from that decision. In accordance with customary Appeal Board practice, however, we have reviewed it on our own motion. See pp. 995-996, *infra*.

As a result of actions directed by the Commission following the accident at Three Mile Island in March 1979, certain matters in this case are still before the Licensing Board, including issues involving emergency planning and other "lessons learned" as a result of the TMI incident.

I

NATURE OF THE CASE

A. The Decision Below

For purposes of the issues now before us, we need sketch only the salient points of this proceeding's anfractuous history. The 750-acre Diablo Canyon site is located on the California coast halfway between San Francisco and Los Angeles, some twelve miles southwest of the City of San Luis Obispo. Like other locations in that State, the area is subject to earthquakes. This was known when the Pacific Gas and Electric Company (PG&E) was initially authorized to construct two nuclear-powered electric generating stations at Diablo Canyon.[3] Since that time, however, new information bearing on the seismic potential of that site has come to the surface. These new developments lie at the heart of this appeal.

We explained on an earlier occasion in this case that

> [a]ll nuclear power plants must be designed and built to protect the public from the hazards of radioactive releases should the plant be subjected to movements in the earth's crust. And such considerations were taken into account when the Diablo Canyon facility was initially proposed for its Pacific coast site. At that time the Nacimiento Fault was taken to be the nearest major active fault, some 18 to 20 miles northeast of the plant. The facility was designed, engineered, and constructed to withstand earthquake damage on this basis. But, years after construction was approved and well underway, that assumption was discovered to be ill-founded.
>
> Subsequent offshore explorations for petroleum have revealed that, at its closest point, the "Hosgri fault" lies only a few miles off the site of the Diablo Canyon facility. That proximity raised the likelihood that an earthquake in the vicinity of San Luis Obispo might be "considerably more severe" than initially anticipated. In light of this intervening development, the plant's design was extensively reanalyzed by the applicant, the staff, and the ACRS. Their consensus was [that] the Diablo Canyon facility as constructed, with some design modifications,

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[3]See Docket No. 50-275, 4 AEC 89, 92-93 (1968) (Unit 1); and Docket No. 50-323, 4 AEC 447, *passim* (1970), *affirmed*, ALAB-27, 4 AEC 652, 653-55 (1971) (Unit 2).

would withstand safely the more severe earthquake shocks now reasonably anticipatable.[4]

The joint intervenors do not share that consensus. They opposed PG&E's applications for licenses to operate the Diablo facility and appeared as adverse parties in the Licensing Board hearing convened to consider them.[5] According to that Board, intervenors' seismic contentions focused on four main areas (LBP-79-26, 10 NRC at 463): (1) the largest earthquake reasonably anticipatable on the Hosgri Fault; (2) the vibratory ground motion such a seismic event would induce at the plant site; (3) the proper criteria for evaluating the plant's ability to survive that event; and (4) the plant structures' responses to its tremors.[6]

Following a long trial, the Board rendered its seismic decision on September 27, 1979. LBP-79-26, 10 NRC 453. In it, the Board below concluded that the "Hosgri Fault"[7] had been fully and properly analyzed and found it capable of producing an earthquake of 7.5 magnitude (7.5M)[8] (*id.* at pp. 468-78). The Board deemed that value "very conservative and an appropriate basis for the Diablo "Safe Shutdown Earthquake" or "SSE" (*id.* at pp. 478-85). An SSE is the seismic event "which produces the

[4]ALAB-519, 9 NRC 42, 45 (1979) (footnotes omitted) (on petition to subpoena two ACRS consultants as witnesses). We note that the statement in the first paragraph of the above to the effect that the Diablo Canyon facility was designed, engineered, and constructed to withstand an earthquake on the Nacimiento Fault some 18 or 20 miles from the plant was taken from a 1969 letter from the USGS to the staff. Testimony now on the record shows that in actuality applicant also designed the plant to withstand an earthquake of magnitude 6.75 with a focus 12 miles from the plant. See testimony of John A. Blume, fol. Tr. 6099, at pp. 9-11 (see *NOTE* below).

[5]Under the Atomic Energy Act of 1954, 42 U.S.C. §§ 2011 *et seq.*, a utility seeking to build and operate a nuclear power plant must obtain separate permits or licenses at both the construction and the operation stages of the project. *Vermont Yankee Nuclear Power Corp.* v. *NRDC*, 435 U.S. 519, 525-27 (1978); *Power Reactor Co.* v. *Electricians*, 367 U.S. 396, 404-05 (1961).

NOTE: The names of all witnesses with their education and present positions are listed alphabetically in Addendum I. "Tr." references are to the transcript of the Licensing Board hearing; "R.Tr." references are to the transcript of the reopened seismic hearing before this Board in October 1980.

[6]Joint Intervenors' contentions appear in full at LBP-79-26, 10 NRC at 468, 478, 486, 490 and 492.

[7]This offshore fault is named for the geologists Hoskins and Griffiths, who found it in the course of private underwater explorations for petroleum. Its existence was not made public until 1971, however, after construction of the plant was authorized and underway. *Id.* at p. 470.

[8]Although the Board did not specify the exact magnitude scale, we believe the record is clear that the "magnitude" referred to is the "surface wave magnitude" Ms. See pp. 930-931 and fn. 198, *infra.*

maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional."[9] A nuclear power plant must be able to withstand the forces of an SSE without releasing dangerous quantities of radioactivity.[10]

The Licensing Board also predicted the maximum vibratory ground motion (in terms of acceleration, measured in units of gravity, "g") that an SSE might induce at the plant site. Intervenors' evidence was that this would be 1.15g. The Board, however, credited the staff and applicants' witnesses who testified that a 7.5M event on the Hosgri Fault would produce an "effective" acceleration no greater than 0.75g. The Board approved that figure as the anchor point for determining the basic *response spectrum*[11] used to evaluate the Diablo Canyon plant's ability to withstand an SSE. 10 NRC at pp. 486-89.

As mentioned, the SSE is the most powerful earthquake ever expected to occur at the plant site. A second seismic event also considered in designing nuclear plants is the "Operating Basis Earthquake" or "OBE." This is the strongest earthquake considered *likely* to occur during a plant's operating lifetime. Nuclear facilities must be designed and built to function through the OBE without creating undue risk to the public health and safety.[12]

On this point there was disagreement about the meaning of the governing regulations. Joint intervenors read them to direct that the maximum vibratory ground motion at the plant site during an OBE be assumed in every case to be half that induced by an SSE, or 0.375g at Diablo Canyon. The applicant and staff, however, construe the regulations to call for a resort to half the SSE value only where no justification has been made for determining OBE ground motion on the basis of specific site conditions. The Licensing Board agreed with the latter construction and found 0.2g to be an appropriate value for the Diablo Canyon OBE. LBP-79-26, 10 NRC at pp. 490-91.

Finally, the Board sanctioned the response spectra developed for evaluating the various plant systems' capabilities of safely withstanding earthquake stresses and approved the methods employed to design and test the Diablo facility against those standards. LBP-79-26, 10 NRC at pp. 492-507.

The Board's decision contained a series of findings to the effect that (1) the safety-related structures, systems and components of the Diablo

[9]"Seismic and Geological Siting Criteria for Nuclear Power Plants," 10 CFR Part 100, App. A, § III(c).
[10]*Id.* at § I.
[11]The concept and purpose of "design response spectra" are discussed in the opinion below at pp. 493-506. See also pp. 923-924, *infra.*
[12]10 CFR Part 100, App. A, § III(d).

Canyon plant will perform satisfactorily through a safe shutdown earthquake; (2) the plant's "Category I" safety systems "will be adequate to insure (a) the integrity of the reactor coolant pressure boundary, and (b) the capability to shut down the reactor and maintain it in a safe condition;" and (3) the facility meets the requirements to function safely through an operating basis earthquake. *Id.* at p. 507.

B. The Reopened Proceeding

The joint intervenors timely filed and briefed a number of exceptions to the seismic decision. Thereafter, the Governor of California — who had not participated in the hearings below — sought to become a party on appeal and support one of intervenors' exceptions. Over the applicant's objections we allowed the Governor to participate as *amicus curiae.* ALAB-583, 11 NRC 447 (March 12, 1980).

On October 15, 1979, about three weeks after the Licensing Board had rendered its seismic decision, a large earthquake[13] struck California's Imperial Valley. This area, some 250 miles southeast of the Diablo Canyon site, is known for seismic activity. For that reason, an array of seismographs — instruments that record the level of seismically-induced ground motions — were in place there. When data derived from that 1979 Imperial Valley earthquake ("IV-79") became available in early 1980, after this appeal had been briefed but before it was decided, joint intervenors moved us to reopen the record. They argued that this new information cast doubt on the validity of key seismic findings made by the Licensing Board and therefore must be taken into consideration.

We granted that motion and chose to receive the new evidence ourselves, framing the issues as questions focusing on our own concerns as well as the intervenors'. ALAB-598, 11 NRC 876 (June 24, 1980).[14] The reopened hearing was held in San Luis Obispo, California, beginning October 20, 1980. It consumed six full trial days; the seventeen witnesses who appeared and testified there included two ACRS consultants called by us.[15] Thereafter the parties and the *amicus* filed proposed findings of fact and conclusions of law on the reopened issues. These are considered in the opinion which follows, for the most part in the course of addressing the issues raised by the initial appeal. Those not covered are either rejected

[13]The magnitude of this earthquake has been variously estimated from 6.5 to 6.9. See fn. 83, *infra.*

[14]Our questions are appended to the decision to reopen. ALAB-598, 11 NRC at 888-92. We authorized the submission of testimony other than in direct answer to our questions provided that it was relevant to the reopened issues. *Id.* at p. 883, fn. 21.

[15]These are Drs. Mihailo Trifunac and Enrique Luco. The reasons for their appearances as Board witnesses are explained in ALAB-604, 12 NRC 149 (August 7, 1980).

because they are unsupported by the record or disregarded as immaterial or irrelevant to this decision.

II

THE HOSGRI FAULT

A. The Reason For Review

Seismology is an evolving science. Reflecting this, the Commission's regulations calling for its application to the siting and design of nuclear plants are complex and perhaps even abstruse. But their purpose is clear: to estimate the magnitude of the strongest earthquake that might affect the site of a nuclear power plant during its operating lifetime; to determine the most intense ground motion that a seismic event could cause there; and to ensure that the nuclear facility is designed and built to survive such an event without undue risk to the public.

The Hosgri Fault, in the Pacific floor some five kilometers offshore from the site location, is accepted by all parties as the geologic feature capable of triggering the largest seismic event at Diablo Canyon. That fault is of relatively recent discovery. Joint intervenors initially questioned whether it had been investigated sufficiently to justify a finding that it was capable of causing no more than a 7.5M earthquake. But after the issue had been tried, intervenors themselves (as well as the other parties) proposed that the Licensing Board make that finding. The Board did so; its opinion marshals the evidence it believed called for that conclusion (10 NRC at 470-85) and explains why, in the Board's judgment, "a 7.5 magnitude earthquake is a very conservative value for the safe shutdown earthquake" (*id.* at p. 485).

On appeal, joint intervenors "agree that the assignment of a 7.5 magnitude earthquake to the Hosgri Fault is acceptably conservative" (Br. at p. 13) but nevertheless filed 39 exceptions attacking the Board's path to that result. Intervenors did not expand upon those exceptions in their brief. They argue only that the Board below took a "one-sided" approach in resolving contested issues of fact against them in "almost every case" (Br. at p. 14). The applicant defends the rulings in question as supported by the "overwhelming weight of evidence" (Br. at pp. 16-20). The staff, however, questions whether the joint intervenors may even challenge a conclusion that they do not dispute simply because they would prefer that it rested on another ground. Stressing the parties' accord on the 7.5M figure, the staff says we may disregard the controversy as moot (Br. at p. 25).

In Commission practice as in judicial proceedings, only a party aggrieved may appeal[16] and issues not briefed may be deemed waived.[17] We normally would invoke those precepts and forego addressing an academic dispute over the best approach to an accepted result. We must eschew that course here. The conservatism (or lack of it) involved in the determination of the Hosgri Fault's potential for causing severe earthquakes is central to this case. This is so because the strength of seismic ground motion (acceleration) which the Diablo facility must be designed to withstand depends directly on the largest earthquake that can reasonably be forecast to occur on that fault. The aspersions cast on the Licensing Board's objectivity in weighing the evidence on this key issue perforce impugn its fairness generally and thus may not be ignored. We therefore reviewed the Licensing Board's Hosgri Fault determinations sufficiently to test its objectivity.

The Licensing Board indeed resolved the key factual disputes in this area against the intervenors' contentions. Our own review of the record satisfies us that the Board did so with justification. Because the intervenors chose not to brief their bias claims we think it sufficient to illustrate our reasons for rejecting them with one concrete but representative example. We do so by reviewing the evidence on the length of the Hosgri Fault.

B. Seismic Background

The evidence recounted in the opinion below illustrates that the present state-of-the-art in geology and seismology does not consider any single element to control the magnitude of an earthquake that a given fault is capable of triggering.[18] Rather, as Commission regulations provide, a number of geologic and seismic factors must be investigated in making that determination,[19] one of which is the overall length of the fault.[20]

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[16]*Rochester Gas and Electric Corp.* (Sterling Project, Unit 1), ALAB-502, 8 NRC 383, 393 fn. 21 (1978), *affirmed*, CLI-80-23, 11 NRC 731 (1980); *Consumers Power Co.* (Midland Plant, Units 1 & 2), ALAB-282, 2 NRC 9 (1975); *Toledo Edison Co.* (Davis-Besse Station), ALAB-157, 6 AEC 858 (1973).

[17]*Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), ALAB-573, 10 NRC 775, 786-87 (1979); *Public Service Co. of Indiana, Inc.* (Marble Hill Station, Units 1 and 2), ALAB-461, 7 NRC 313, 315 (1978), and cases there cited.

[18]It is not disputed that the Hosgri is such a "capable fault" within the meaning of the regulations. See 10 CFR Part 100, App. A, § III(g).

[19]10 CFR Part 100, App. A, § IV.

[20]Other factors to be considered include (a) the geology and seismology of Southern California; (b) the Hosgri Fault's relationship to major fault systems; (c) the correlation between maximum earthquake size and fault length; (d) type of faulting and past displacement; (e) seismic history, including the location of recent large earthquakes; and (f) the relationship between earthquake recurrence rate and intensity.

Basic Southern California geology and seismology is not in controversy.[21] As an appreciation of it is helpful in focusing the parties' disagreements, we summarize here the overview presented in the prefiled joint testimony of applicant's witnesses Douglas H. Hamilton and Richard H. Jahns.[22]

The central feature of California geology is the San Andreas Fault. This is displayed on the map contained in Figure 1 on the following page.[23] The San Andreas is now accepted as a major boundary between two of the large, some 80km thick, slabs or "plates" that comprise the earth's crust and "float" on its molten interior core. Modern seismologists attribute most major earthquakes to "plate tectonics," *i.e.,* movement of these plates relative to each other.[24]

The San Andreas Fault was formed some 100 million years ago. It runs north-northwest from the Gulf of California in an essentially straight line (except for an S-type bend northwest of Los Angeles) to San Francisco. From there it curves in and out along the California coastline for another 150 miles before turning northwesterly into the Pacific Ocean. In its passage through California, the San Andreas Fault splits both the Coastal Mountain Range and the Transverse Mountain Range. The latter runs east-west and reaches the coast in the Santa Barbara region, south of Diablo Canyon. Probably originating by under-thrusting of the American Plate by the Pacific Plate, the San Andreas changed about 20 million years ago to a right "strike-slip" motion between the plates. (In a "strike-slip" fault, the ground on one side of the fault moves horizontally and parallel to that on the other side. A "right strike-slip" means that an observer looking across the fault perceives the ground opposite as moving to his right; *vice versa* in a left strike-slip. See Figure 2 on page 916.) The San Andreas Fault is the only geologic feature that can be clearly traced without interruption from northern to southern California and the only regional fault on which both ends show divergent plate boundary features.[25]

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[21]Except for the location of an earthquake that occurred in 1927, which we touch upon later. See fn. 38, *infra.*

[22]Their testimony, bound into the transcript of the Licensing Board hearing, is cited hereinafter as "Hamilton-Jahns, fol. Tr. 4457."

[23]Figure 1 is an outline map of South Central California showing structural provinces and faults, taken from *Earthquakes: A Primar* by Bruce A. Bolt, W.H. Freeman and Company, Copyright © 1978. It also appears as Figure 8 in the Hamilton-Jahns testimony, fol. Tr. 4457.

[24]See Board Exhibit 3 for identification, Bruce A. Bolt, *Earthquakes: A Primer* (1978) at 12-17 (hereinafter cited as "Bolt, *Earthquakes*") Dr. Bolt appeared as a witness for the applicant in the Licensing Board hearing. Tr. 5446.

[25]These features are described as "spreading ridges centers in the Gulf of California on the south; the Mendocino triple junction on the north." Hamilton-Jahns, fol. Tr. 4457 at p. 8.

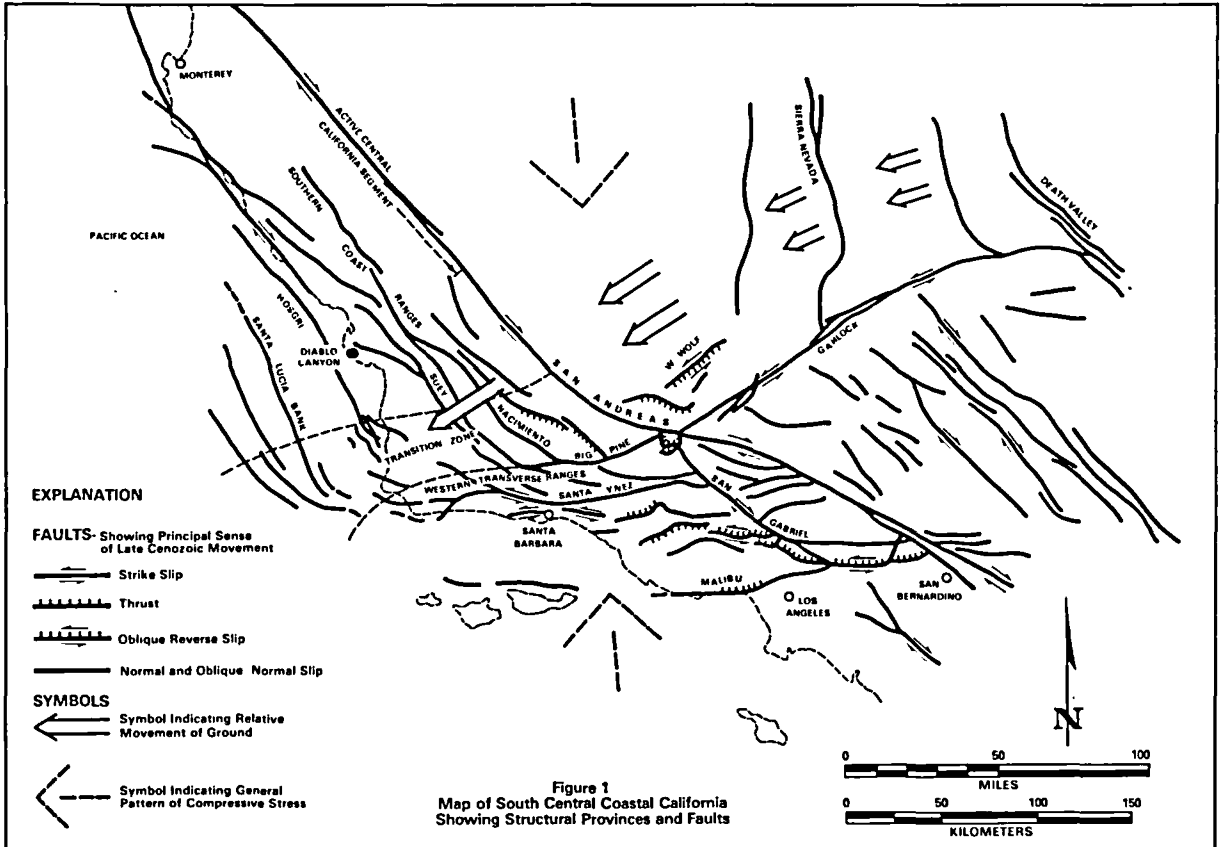

Figure 1
Map of South Central Coastal California
Showing Structural Provinces and Faults

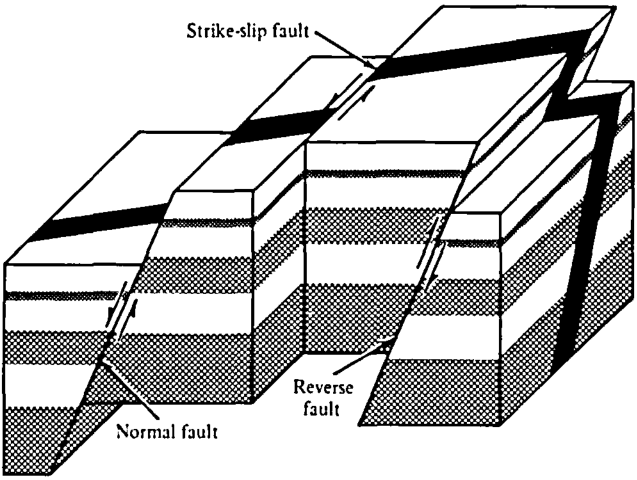

FIGURE 2 Diagram showing the three main types of fault motion. (From *Earthquakes: A Primer* by Bruce A. Bolt. W. H. Freeman & Company, Copyright © 1978.) Bolt defines a thrust fault as a reverse fault with a small dip.

Complications in the geology show up at the central area of the "S-bend" curve mentioned above. In this region (between Bakersfield on the northwest and San Bernadino on the southeast) the San Andreas Fault is intersected by the Garlock Fault. This fault runs approximately east-northeast to the Death Valley area and has a left strike-slip motion.[26] This indicates that the mass of the earth's crust north of the Garlock and east of the San Andreas is moving westward towards the Pacific. This crustal extension has influenced the deformation (*i.e.*, folding and faulting) caused by the north-south "compression"[27] of the Coastal Ranges to produce a boundary region or "transition zone" between the Coast Ranges and Transverse Range Provinces.[28] This region is marked by the "big bend" in the San Andreas.[29] The Hosgri and other faults which roughly parallel the Coastal Range Mountains to the west of the San Andreas show similar sharp bends to the east just north of the Transverse Mountain Range.[30]

[26]See Figure 2, *supra*, p. 917.
[27]Compressional stress produces folding of the crust (mountain forming) and "thrust faulting." In a thrust fault, the crust on one side of the fault moves vertically in relation to the crust on the opposite side, rather than horizontally along the fault as in a strike-slip fault. The difference between thrust faults and strike-slip faults is illustrated in Figure 2, *supra*, p. 917.
[28]Hamilton-Jahns, fol. Tr. 4457 at pp. 11, 15-17, 20-26.
[29]See Figure 1, *supra*, p. 916.
[30]*Ibid.*

C. The Length of the Hosgri

As we mentioned (p. 914, *supra*), fault length is one key factor considered in determining its maximum earthquake potential. The Licensing Board rejected intervenors' evidence that the Hosgri was 250 miles in length, finding instead, as the other parties contended, that it was only 90 miles long. LBP-79-26, 10 NRC at 472-75. The Board's evaluation of the record on this disputed point provides a fair test of its objectivity.

The applicant's evidence described a series of individual faults mapped to the west of and parallel to the San Andreas, the Hosgri among them (see Figure 3, p. 919, *infra*).[31] It is joint intervenors' position, propounded by their witnesses Drs. Eli Silver and Stephan Graham, that several of these faults are interconnected. The salient points of their testimony in support of this thesis are set forth in the margin below.[32] In essence, Dr. Silver traced

[31]Figure 3 appears in the record as Figure 16 appended to the Hamilton-Jahns testimony, fol. Tr. 4457.

[32]Dr. Silver's prefiled direct testimony appears as Joint Intervenors' Exhibit Number 49 (hereinafter "Silver, J.I. Exh. 49")

(a) As evidence supporting a connection between the Sur and San Gregorio Faults, Dr. Silver referenced a 1973 thesis prepared for the U.S. Navy Post-Graduate School at Monterey, California, by W.B. Woodson III, entitled "A Bottom Gravity Survey of the Continental Shelf Between Point Lobos and Point Sur, California," and a report by Dr. Graham and W.R. Dickinson, "Apparent Offsets of On-Land Geologic Features Across the San Gregorio-Hosgri Fault Trend in San Gregorio-Hosgri Fault Zone, California," that appeared in the California Division of Mines Geology Special Report (1973) pp. 13-23, edited by Dr. Silver and W.R. Normark. Dr. Silver testified that these reports "indicate that the Palo Colorado may be a minor splay fault off the main San Gregorio fault zone, and that the main trace comes ashore in the area of Hurricane Pt., to connect with the Sur fault zone." (Silver, J.I. Exh. 49 at 1, 2-4). According to Dr. Silver, the Woodson work included gravity mapping of the Point Sur area indicating that the main trace of the San Gregorio Fault tends into a gravity gradient southwest of Point Lobos and its "most likely course" is along this gradient to Hurricane Point, where it continues as the Sur Fault (*id.* at 1, 2-4 and 2-5).

(b) Dr. Silver acknowledged that data are lacking to connect the south end of the Sur Fault directly to the north end of San Simeon. But he added that aeromagnetic surveys reduce the gap between them to only 5km and these show the San Simeon trending offshore to the northwest with the two fault zones striking "directly toward each other." (*Id.* at 1, 2-5.)

(c) According to Dr. Silver, the situation between the San Simeon and Hosgri fault zones is similar to the Sur-San Simeon connection, *i.e.*, seismic proof (*e.g.*, observed geologic data) of the connection is lacking. Dr. Silver asserted, however, that available aeromagnetic data show that the San Simeon fault zone "projects southward along a magnetic high of the same strike," while "[t]he Hosgri, where mapped to the south, projects northward along the upper west flank of this magnetic high." Dr. Silver claims that if the Hosgri followed the magnetic high to the coast, "it would intersect a segment of the San Simeon fault zone, as seen on the aeromagnetic map, and would be within 1-2 km of the main trace of the San Simeon." Thus he explained that "[d]efinitive seismic reflection data are lacking in this zone, but the aeromagnetic data provide a guide to the possible location of the fault zone in this area." (*Id.* at p. 1, 2-5).

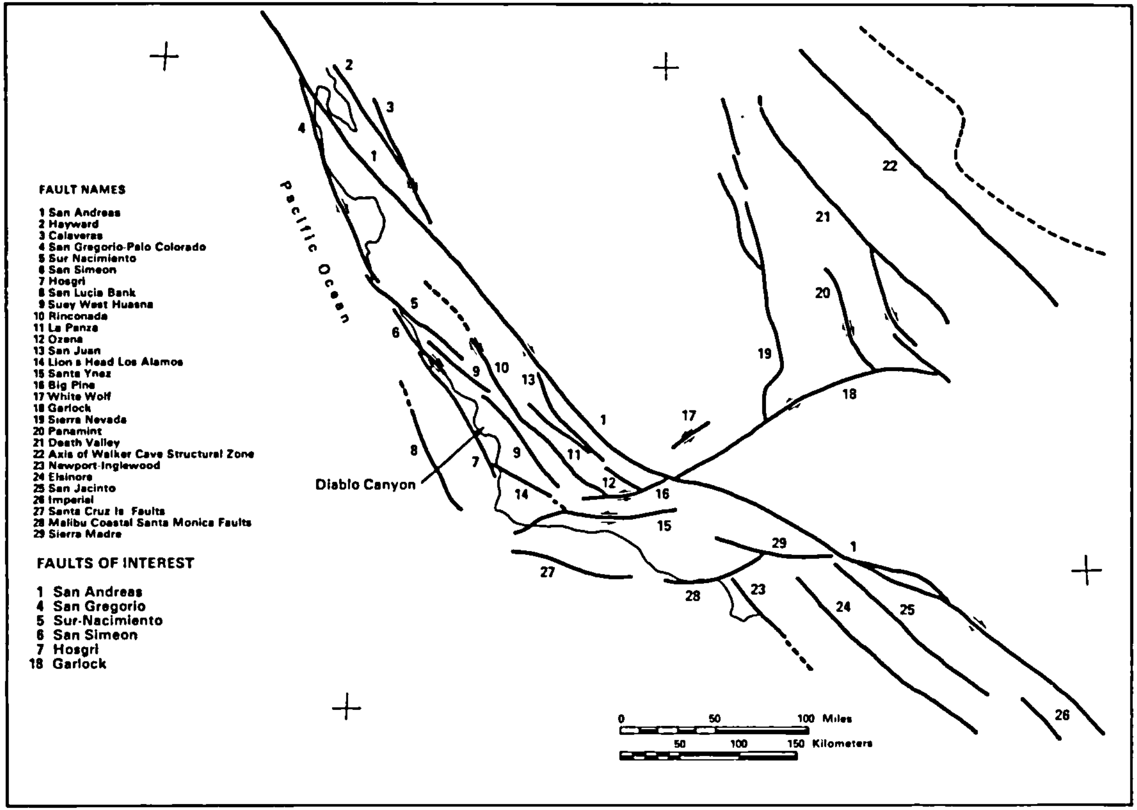

Figure 3 Major Faults. South Central and Southern California

the evidence he believed led to the conclusion that the Hosgri Fault connects directly to the San Simeon Fault north of it, the San Simeon in turn to the Sur-Nacimiento Fault further up the coast, and this to the San Gregorio which joins the San Andreas north of San Francisco, to form in all a 250-mile long fault.

Generally agreeing with Dr. Silver, Dr. Graham emphasized what he characterized as offsets in older "basement" rock formation (*i.e.*, structures underlying the rocks formed by sedimentation). These basement offsets, he testified, evidenced a right strike-slip displacement in excess of 100km that occurred *between 5 and 15 million years earlier*, and the existence of a lengthy, continuous fault system during that time is necessary to account for this movement. Dr. Graham postulated that these faults then represented the stress boundary between two major tectonic plates, but that the focus of that stress has now shifted eastward to the San Andreas fault zone.[33]

The thesis put forward by Dr. Silver and supported in some part by Dr. Graham was, however, undercut by testimony from another of intervenors' witnesses, Dr. Clarence Hall. He departed markedly from their positions

[33]Dr. Graham's testimony (hereinafter "Graham, J.I. Exh. 48"), supported Dr. Silver's view that the San Gregorio, Sur-Nacimiento, San Simeon, and Hosgri Faults form one continuous fault zone with an examination of basement rock formations on both sides of these faults. In attachment B to his prefiled direct testimony (and see also Tr. 6175) he outlines seven pairs of rock formation of ages between 15 and 60 million years that he concludes show a similar right-strike movement of about 115km in each of these faults. (Attachment B appears to be a pre-publication copy of a California Division of Mines and Geology Special Report dated 1978, prepared in conjunction with W.R. Dickinson. Mr. Dickinson did not testify.)

Dr. Graham further noted that the northern end of the San Gregorio merges directly into the San Andreas Fault near the entrance to San Francisco Bay. (See Figure 3, *supra*, p. 919.) To the north of this junction, Dr. Graham asserts that the basement rocks along the San Andreas show about 115km more fault slip than do the basement rocks on the San Andreas south of the junction. From this Dr. Graham postulates that the movement of the combined fault north of the junction must be the sum of the right-slip movements of the San Andreas and San Gregorio-Hosgri south of their point of merger. (Graham, J.I. Exh. 48, Attachment B.)

In Dr. Graham's judgment, the "[g]eologic evidence suggests that the San Gregorio-Hosgri was a continuous fault in the past that was the focus of shear resulting from the stresses generated by the movement of the North American and the Pacific Plates," but that "[t]he focus of shear appears to have moved east to the San Andreas fault zone." (Graham, J.I. Exh. 48 at 1-1 and 1-2). He concluded with the observations that "[a]lthough this evidence does not require thorough-going continuity in the present, it is suggestive" of this fact, and that he was unaware of geologic data that would preclude it (*ibid.*). He added that "[b]ased on our conclusions which we stated in the paper, particularly with respect to the rate of movement curves, it looks to us as though the predominant right slip, by our interpretation of the fault system, occurred between 15 and 5 million years [ago]" (Tr. 6364).

both on the Hosgri's origins and on its movements.[34] Among other things, Dr. Hall expressed the judgment that the Hosgri Fault was formed within only the last 5 million years. He further testified that the 80km of movement

[34]Dr. Hall was a rebuttal witness for intervenors and did not submit prefiled testimony on their behalf. However, the substance of many points in his testimony also appears in two of his published articles, in evidence as J.I. Exhs. 36, "San Simeon-Hosgri Fault System, Coastal California: Economic and Environmental Implications," *Science*, 190, 1291-94 (December 26, 1975); and 37, "Origin and Development of the Lompoc-Santa Maria Pull-Apart Basin and Its Relation to the San Simeon-Hosgri Strike-Slip Fault, Western California," in *California Division of Mines, Special Report Number 137*, hereinafter cited as "J.I. Exhs. 36 and 37."

Dr. Hall focused on the San Simeon and Hosgri Faults, with major emphasis on the southern end of the Hosgri. He asserted that "[c]omparison of stratigraphic sections exposed on opposite sides of the late Quaternary [*i.e.*, last 5 million years] San Simeon-Hosgri fault system at Point Sal and near San Simeon * * * strongly suggests large-scale lateral displacement." J.I. Exh. 36 at p. H-1. Dr. Hall further stated there that recent geologic mapping near San Simeon and the area between Santa Maria and San Simeon has revealed remarkable similarities between rocks west of the San Simeon Fault zone near San Simeon and east of the Hosgri Fault near Point Sal. *Id.* at p. H-2. See also Tr. 9474-75 and J.I. Exhs. 73 through 108.

In the remainder of his oral testimony and in J.I. Exh. 37, p. 25, Dr. Hall proposed "[a] speculative model * * * to account for the distribution of tertiary igneous, sedimentary and volcaniclastic rocks that lie within the Santa Maria-Lompoc region, Santa Barbara County, California." He used this model to suggest a possible southern extension of the Hosgri Fault, "because clearly if there is to be 80 kilometers of movement somewhere along the San Simeon-Hosgri Fault that movement must also include areas to the north and south. So what would happen? Where would that movement occur in the south? And we have then a possible landfall of the Hosgri Fault in the area between Purisima Point and Point Arguello." Tr. 9537-38.

Dr. Hall's model employs a "pull apart theory." This assumes that, prior to 15 or 16 million years ago (Tr. 9575), there existed what Dr. Hall called the Santa Maria River-Foxen Canyon-Little Pine-Lompoc-Solvang Fault (Tr. 9578). He postulates that this opened up and eventually formed the Santa Maria Basin, now bounded by the Santa Maria River and Foxen Canyon Faults on the north and the Lompoc-Solvang Fault on the south. His conclusion rests on certain rock sequences along the north side of the Santa Maria-Foxen Canyon Fault, which Dr. Hall asserts match similar sequences on the south side of the Lompoc-Solvang Fault, but are absent in the basin between those faults (see, *e.g.*, Tr. 9574-76). Dr. Hall testified that the Hosgri Fault formed after the development of that pull-apart basin, most likely 5-8 million years ago (Tr. 9579), when it merged with the Lompoc-Solvang Fault on the California coast (Tr. 9569-70).

Dr. Hall finds confirmation for his theory in the sequences and absences of rock formations in the log of the "Oceano Well." (This exploratory oil company well was drilled on the west side of the Hosgri offshore of Point Sal; in the process, a complete core of the lower part of the well was obtained so that the rock types could be recorded against the depth at which they were located.) In Dr. Hall's judgment, sequences from the west side of the Hosgri opposite Point Sal match rock sequences found in wells drilled in the Santa Maria Basin between the Lompoc Fault and Purisima Point. From this he deduces that the rock found around the Oceano Well came from the onshore California coast south of the Lompoc-Solvang Fault (Tr. 9554-55). Dr. Hall concluded that the Hosgri Fault was formed after the completion of the pull-apart basin some 5 million years ago, and that an 80 kilometer movement occurred after its formation, *i.e.*, *within* the last 5 million years.

which he attributes to it took place during that period. His testimony thus contradicts Dr. Graham's. As we just mentioned, Dr. Graham testified that the formation of and movements along the Hosgri occurred millions of years earlier. See p. 920, *supra*. Dr. Hall's own theories on the formulation and evaluation of the Hosgri Fault are themselves highly problematical — for one thing, they assume the movement of large masses of rock across a fault, testified to be a geologic and physical impossibility.[35] This does not, however, change the fact that on key points, the testimony of intervenors' witnesses was markedly divergent.

Beyond the disharmony in intervenors' own case, the opposing parties introduced persuasive evidence that the actual geology of the area is incompatible with intervenors' combined fault theory. Perhaps most significant are aeromagnetic studies[36] and direct testimony that the Hosgri and San Gregorio Faults are separated by a large and undisturbed mass of Franciscan bedrock.[37] The Board below relied on this, among other things, in rejecting the combined fault theory that was central to intervenors' case on fault length. LBP-79-26, 10 NRC at 475.

Further evidentiary support for the Board's finding that the Hosgri is not part of a 250-mile long system but an individual fault some 90 miles long is recited in the opinion below. (See, *e.g.*, LBP-79-26, 10 NRC at 485.) There is no cause to rehearse it here. For the reasons we gave at the outset of this

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[35]Compare Hall, Tr. 9614 *et seq.*, with Jahns, Tr. 10,035-49. Fairness requires acknowledgment that Dr. Hall candidly characterized his own theory as a "speculative model." J.I. Exh. 37 at p. 25.

[36]In this technique, aircraft equipped with continuous-recording magnetometers fly above the area being studied. Those devices measure variations in the earth's magnetic field which information can be used to determine the properties and types of rocks overflown.

[37]For example, applicant's witness Mr. Douglas Hamilton explained in his testimony that (Tr. 10,019-20):

> "Now Dr. Silver goes on to say that if projected northward the Hosgri fault would run on land near the San Simeon fault at a point 1 or 2 kilometers to the west.
>
> "The point I want to make is, if we make the projection we find that the San Simeon — the Hosgri fault would project to a place along the shoreline that does indeed lie a few kilometers west of the San Simeon fault. This is an area that has been very carefully mapped both by myself and my associates and also by Dr. Hall, and it's an area where there are very good exposures right along the seacliff with more scattered exposures inland. There is no major fault appearing in this area. There is no disruption of the geologic units that extend from right next to the San Simeon fault going out across this area. There is also no pattern in the branching faults that would permit some kind of a branch connection from the San Simeon fault to transfer to the Hosgri fault.
>
> "So the point simply is that if one takes the contention that Dr. Silver describes in his written testimony and you compare it with the data, you do not get a connection between the two faults. You, instead, find them separated, as we have claimed, by an intact mass of Franciscan bedrock that has several kilometers width and lies between the two."

discussion (p. 914, *supra*), we reviewed the record basically to see if there was a basis for intervenors' unparticularized claims of bias on the Board's part against their witnesses. Our discussion of the record on the Hosgri Fault's length is simply a representative illustration of why we found intervenors' charges unsubstantiated. We think it amply clear from that example that the Board's rejection of intervenors' contentions on fault length was soundly based.

The Board's evaluation of the other issues underlying the conclusion that 7.5M is the largest magnitude earthquake likely on the Hosgri also faithfully reflects the record made before it.[38] Without implying that the Board resolved every dispute perfectly, we are satisfied that it weighed the evidence fairly and arrived at the correct result.

We close this point with a reminder. The resolution of issues of fact in favor of one side suggests neither bias nor error on the tribunal's part; without more, the appropriate inference is that the evidence of the prevailing party was the more persuasive. Be that as it may, we reiterate that in administrative hearings as in court cases, rulings and findings made in the course of a proceeding are not in themselves sufficient reasons to believe that the tribunal is biased for or against a party.[39]

III

THE SEISMIC REANALYSIS OF DIABLO CANYON

A. Introduction

As we noted (p. 913, *supra*), no party has challenged the Licensing Board's determination that the appropriate safe shutdown earthquake for the Diablo facility is a 7.5M event on the Hosgri Fault. The next step is translating the ground motion induced at the plant site by the SSE into information useful in analyzing the plant's ability to withstand such a seismic event. Earthquake motion can be described in terms of displace-

[38]We do think it worth noting, however, that a basis for assigning earthquakes as large as 7.5M to the Hosgri has weakened rather than strengthened since the initial hearing below. USGS witnesses testified in the Licensing Board hearings that a 7.3 earthquake occurred on that fault in 1927. LBP-79-26, 10 NRC at 485. At the reopened hearing before us, however, witnesses from that agency acknowledged that more recent studies tend toward the idea that the 1927 earthquake occurred on another fault, the Lompoc (R.Tr. 943-65). Those witnesses could not, however, rule out all possibility that the Hosgri was the source of that event. See Devine, R.Tr. 950-51 (See *NOTE*, p. 910, *supra*).

[39]*Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear 1), ALAB-224, 8 AEC 244, 246, *rehearing denied*, ALAB-227, 8 AEC 417 (1974), *reversed sub nom. Porter County Chapter* v. *AEC*, 515 F.2d 513 (7th Cir.), *reversed summarily and remanded sub nom. Northern Indiana Public Service Co.* v. *Walton League*, 423 U.S. 12 (1975), *affirmed on remand*, 533 F.2d 1011, (7th Cir.), *certiorari denied*, 429 U.S. 945 (1976).

ment (the distance the ground moves at any given point during an earthquake); velocity (the speed of that ground movement); and acceleration (the rate at which that velocity changes expressed in terms of "g," the acceleration of gravity). In order to assess earthquake effects, a building or mechanical system may be conceived of as a damped, harmonic oscillator having a particular frequency. When such oscillators are subjected to the vibratory motion induced by an actual or postulated earthquake, their maximum reactions in terms of displacement, velocity, and acceleration can be predicted by means of a "response spectrum."[40] The spectrum can then be used both to design and to analyze structures, components, and systems for their capability to withstand earthquake induced stresses.[41] The development of such response spectra is required by the governing regulations. 10 CFR Part 100, App. A, § VI(a).

For the reanalysis of the Diablo Canyon facility the applicant and the NRC staff each prepared a basic response spectrum to characterize the motion at the Diablo Canyon site assuming a magnitude 7.5 earthquake on the Hosgri Fault. Both took a value of 0.75g[42] for the high frequency anchor

[40]More definitively, a response spectrum is the result of an analytical procedure whereby a number of one-degree-of-freedom harmonic oscillators, each having the same degree of damping but with different natural frequencies, are driven by the time-dependent motion characteristic of a real or postulated seismic event. For a particular event and degree of damping there will be a time-dependent response which varies for oscillators of the different frequencies. The maximum values of the response of the oscillators in terms of acceleration, velocity and displacement, may be plotted as a function of the frequency of the oscillators being excited. Such a plot can be produced for any one of the three parameters taken individually. Because of the relationship among acceleration, velocity and displacement under harmonic motion, a tripartite plot showing the maximum responses in acceleration, velocity and displacement as a function of oscillator frequency may also be prepared (see, *e.g.*, Regulatory Guide 1.60, Figure 1).

The term "damping", as it pertains to the response of a simple harmonic oscillator, relates to internal, friction-like processes by which the initial kinetic and potential energy of the oscillating system are transformed into heat, thus reducing the amplitude of the oscillation. The analysis of the motion of the harmonic oscillator system proceeds under the assumption that the motion is in the linear or elastic range (*i.e.*, the restoring force is directly proportional to the displacement). See also the discussion of response spectra in the opinion below, LBP-79-26, 10 NRC at 486, 493 and in the testimony of Dr. Blume, fol. Tr. 6099 at pp. 5-7, and Dr. Frazier, Tr. 6607 *et seq.*

Response spectra tend to have jagged peaks and valleys. For engineering analysis and design purposes these can be evened out either (1) by drawing a smooth curve enveloping the peaks (or by averaging the peaks and valleys), or (2) by statistically combining individual spectra derived from similar earthquakes. When so smoothed they are sometimes called "design response spectra." See NRC Regulatory Guide 1.60 at p. 1.60-3 (Rev. 1, December 1973).

[41]See, *e.g.*, Tr. 8641.

[42]The expression 0.75g indicates an acceleration equal to 75% of the acceleration due to gravity ($g = 980\ cm/sec^2$).

point acceleration for the spectrum.[43] Variations of these spectra, modified to reflect specific effects believed to be active at that site, were then used to provide the basis for the seismic reanalysis of the Diablo Canyon facility.

In the proceedings below, joint intervenors contended that the choice of 0.75g was an inappropriate value for the 7.5M earthquake on the Hosgri. See LBP-79-26, 10 NRC at 457 (Contention 3). In other contentions they challenged the validity of the response spectra that were used, the methods for generating them, and the mechanisms utilized to justify a reduction in the motion predicted by the basic spectra. *Id.* at p. 492. After hearing evidence on these issues, the Licensing Board concluded that the response spectra employed by the staff and applicant, as anchored at 0.75g and as modified, were appropriate. *Id.* at pp. 490, 493-97. Whether the Board was correct in doing so is a central issue raised by intervenors' appeal.[44] In addition, we allowed the Governor of California (who did not participate in the Licensing Board hearings) to file an *amicus curiae* brief with us.[45] The *amicus* brief[46] questions the trial board's conclusions on the focusing of seismic motion and the existence of high stress drops along the fault,[47] subjects which ultimately pertain to the appropriateness of the response spectra used to analyze seismic ground motion[48] at the Diablo site.

The intervenors and the Governor criticize the Licensing Board's treatment of the response spectra evidence in three broad categories:

(a) The maximum horizontal ground motion to be associated with a 7.5M Hosgri event and the use of "effective acceleration" to fix the anchor point at 0.75g;

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[43]See, *e.g.*, LBP-79-26, 10 NRC at 486. The most commonly measured characteristic of earthquake motion, obtained using a seismograph, is the time-dependent acceleration of the ground (or some other foundation of the seismograph) during the earthquake (Tr. 5495). In an earthquake, a hypothetical very rigid structure (*i.e.*, one with very high natural frequencies) would shake in phase with the motion of the ground itself — and the ground motion would not be amplified in the building. For this reason, the high frequency or "zero period" portion of the response spectrum provides a convenient point from which to scale the standard spectrum; hence the high frequency end of the spectrum is called the anchor point. In Regulatory Guide 1.60 the staff indicates that a building whose natural frequency is 33 hertz or greater will move with the acceleration of the ground. The natural frequencies of nuclear facility buildings lie in the range of 1 to 10 hertz and, in that range, structures will experience some motion amplification (see fn. 336, *infra.*).

[44]See Joint Intervenors' Brief in Support of Exceptions, December 7, 1979, pp. 16-55 (hereinafter "J.I. Br. at 000").

[45]See p. 8, *supra.*

[46]Brief *Amicus Curiae* of the Governor of the State of California, December 7, 1979, *passim* (hereinafter "Gov. Br.").

[47]These subjects are discussed *infra* at pp. 944 *et seq.*

[48]Unless otherwise specified, references to seismic ground motion in this opinion and in the underlying record are to *horizontal motion* (*e.g.*, motion in the plane of the earth's surface). Earthquakes also generate motion in the vertical direction (see, *e.g.*, pp. 957-962, *infra*). When vertical motion is being considered it will be explicitly identified.

(b) The use of a "tau-effect" to reduce the high frequency portions of the response spectra; and

(c) The use of a 7 percent "damping ratio" for steel and concrete structures.

The evidence introduced before us at the reopened hearing to consider new data from the 1979 Imperial Valley earthquake (see p. 912, *supra*) essentially bears on the first two of those three points. But it also brings up an additional issue: the adequacy of the Diablo redesign in relation to the *vertical* ground motion of a 7.5M seismic event and, hence, the appropriateness of the vertical motion response spectrum used for that purpose.

Before we may address those issues, however, it is necessary first to look closely at the special characteristics of seismic ground motion phenomena with which this case is principally concerned — motion in the "near field" of a strong earthquake.

B. General Considerations of Effects of a Hosgri Earthquake in the Diablo Canyon Area

The Diablo Canyon seismic reanalysis started from the assumption that the 7.5M SSE would occur on the Hosgri at the point closest to the nuclear facility, a distance of about 5.8 kilometers.[49] A site this close to a rupturing fault is considered to be in the "near field" of earthquake motion.[50] The characteristics of near-field motion (and their reflection in design response spectra) are manifestly important in determining the potential stresses that the Diablo facility would have to withstand and considerable evidence on the point was adduced before the Board below.[51] Because there are relatively few seismographic records of motion close to rupturing faults, much of that testimony was based on information from recording instruments distant from the actual region of energy release, that is to say, beyond the near field. This reason (among others) persuaded us that joint intervenors were correct in asking to have the record reopened for the IV-79 data. Those data were not only derived from a strong seismic event, but included extensive information on near-field ground motion that could be compared with predictions made in the Licensing Board hearings.[52]

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[49]See, *e.g.*, Blume, fol. Tr. 6100, at 12; Blume, IV-79 testimony, at I-3; Newmark, fol. Tr. 8552, Reference A, at C-2; SER Supp. 7, fol. Tr. 8183, at 1-3. ("IV-79 testimony" refers to prefiled, direct testimony presented at the reopened hearing. Unlike that of witnesses for the intervenors and staff, prefiled testimony of the applicant's witnesses was not bound into the transcript.)
[50]Smith, Tr. 5916-17; Bolt, Tr. 5927; Brune, Tr. 8023-24, 8088-89. See also p. 928, *infra*.
[51]See, *e.g.*, Smith, Bolt, Frazier and Hamilton, Tr. 5859-97, 5915-48, 5990-6002, 6031; Brune, Tr. 7934-8129; Newmark, Tr. 8596-8630; see also LBP-79-26, 10 NRC at 487-89.
[52]See ALAB-598, *supra*, 11 NRC at 876-881.

The record reveals a general agreement among the experts about the basic nature of fault-related earthquakes and the resulting motion in the near field. It also provides a sound basis for describing the events associated with the strong tremors that affect structures close to a rupturing fault. Understanding those phenomena is needed not only to place in context the Licensing Board's findings and conclusions on vibratory ground motion and design response spectra, but to appreciate the challenges made to those findings and conclusions.[53] Accordingly, we first describe the seismic phenomena giving rise to near-field motion, then explain certain significant conclusions which in our judgment follow from those phenomena,[54] and, finally, in the light of those discussions, examine the specific issues raised by the appellants.[55]

1. **Fault-related earthquakes.** In simple terms, a fault-related earthquake occurs as the result of an increase in stress across the fault. (This stress buildup can have various causes; *e.g.*, subsidence of rock on one side of a fault or, in the case of a plate boundary, deep earth motions causing the plates themselves to shift positions.) Rocks strained by this increase in stress finally rupture and release the stored energy in the form of seismic waves. These waves have a wide range of frequency and may exert compressional or shear forces on the earth through which they pass.[56] As the energy radiates outward from its source in the form of seismic motion, the higher frequency components are attenuated by the earth more rapidly than the low frequency portions.[57]

The magnitude of an earthquake — the amount of energy released — can be related to the "stress drop"[58] occurring when the rock fractures, the style of faulting involved, *i.e.*, thrust or normal[59], and the rupture's size. Rupturing on faults has both longitudinal and vertical dimensions; in terms of the magnitude of energy release, however, the longitudinal component is the more significant.[60] As the record reveals, the rate of energy release

[53]See pp. 925-926, *supra*.
[54]Our conclusions do not necessarily represent the consensus of all the expert testimony; where there are conflicts, we explain why we have adopted one position over another.
[55]We cover the broad issues raised on the initial appeal (see pp. 925-926, *supra*) at pp. 962-978 and 962-982, *infra*, taking into account the IV-79 data received at the reopened hearing. The matter of vertical ground motion we treat separately at pp. 957-963, *infra*.
[56]Tr. 5501-06; 6023.
[57]Tr. 5499.
[58]Smith, fol. Tr. 5490 at 17. "Stress drop" may be defined as the change (decrease) in the rock stresses on either side of a fault before and after an earthquake.
[59]Under *normal* faulting conditions the rocks are in tension — that is, being pulled apart. *Thrust* faulting results from the failure of rocks in compression — being squeezed together. The stress associated with thrust faulting is generally greater than in normal faulting (*Smith*, fol. Tr. 5490 at 9).
[60]Tr. 5586-87; Tr. 5685; R.Tr. 755.

along a rupturing fault is not uniform but highly irregular, corresponding generally to regions of high and low stress concentration in the fracturing rock.[61] The picture which evolves for large earthquakes of fault origin is essentially one of energy being released over an extended portion of the fault, and there have been correlations made between an earthquake's measured magnitude and the length of the ruptured fault.[62]

Another observable characteristic of fault-induced earthquakes is that generally the faulting and energy release take place deep (*i.e.*, on the order of 10km in California) beneath the earth's surface.[63] Such a rupture may also be expressed on the earth's surface when the relative vertical or horizontal motion across the sides of the fault produce cracking and visible ground displacement along the line of the fault.

2. **"Near-field" motion characteristics.** As we explained, p. 926, *supra*, the term "near field" describes locations on the earth's surface close to the rupturing rocks involved in the seismic motion. A number of more or less qualitative definitions of "near field" were proposed during the hearing.[64] For our purposes, however, it is sufficient to treat the near field as that strip on the earth's surface about 10km on either side of the fault.[65]

There is a phenomenological basis for defining a near-field region. Because the energy release on rupturing faults takes place at some depth, locations on the earth's surface within this region are at nearly the same distance from the source of energy release. Hence, the attenuation of seismic motion that is a function of distance from that source (*i.e.*, the reduction attributable to the resistance of the earth itself) would be nearly the same everywhere in the near-field region. Only beyond the near field

[61]Tr. 5938-39; Tr. 8629; R.Tr. 1385; J.I. Exh. 66 at 3-14; Tr. 8505.

[62]Tr. 5685-88. While the scatter of the data included in these correlations is considerable, because other factors also determine the total energy release, it is not seriously disputed that large magnitude earthquakes are generally associated with rupturing along a considerable length of a fault. See also Renner B. Hofmann testimony on Contention 2, fol. Tr. 8522, Figures 2-2 and 2-3.

[63]Smith, fol. Tr. 5490 at 18; Tr. 5536; Tr. 6641.

[64]*See, e.g.*, Tr. 5927, 8023-24, 5916, 8088-89; R.Tr. 677.

[65]For faults oriented vertically to the earth's surface this definition of near field would result in a strip 20km wide centered upon the surface expression of that fault. For faults with planes at some angle with respect to the earth's surface, the region of energy release during faulting at depth would not lie directly beneath the surface expression of the fault. In such a case, the midline of the near field would be a region on the earth's surface directly above the rupture at depth. R.Tr. 366.

does distance on the earth's surface away from the fault significantly increase the overall distance to the region of energy release.[66]

3. **Magnitude saturation.** A consequence of the release of seismic energy over an extended area is that peak ground motion in the near field "saturates." Saturation, as we use the term here, refers to an observed physical effect which occurs when two variables or parameters normally directly proportional to each other lose that proportionality at some point in their range. Saturation occurs when, as one of the variables continues to increase in value, the other increases less rapidly and finally may become constant. In the graph below, A saturates with respect to B.[67] Magnitude saturation begins when the peak ground motion to be expected in the near field of a strong earthquake is no longer primarily dependent upon the total release of seismic energy. Rather, it is more strongly influenced by the nature of the faulting along the rupture adjacent to where the ground motion is measured.[68] To understand the concept of saturation one must appreciate the full scope of an earthquake movement. In the stress relief

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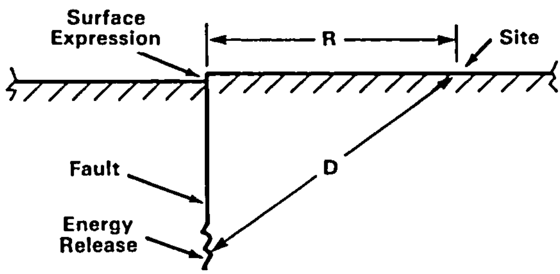
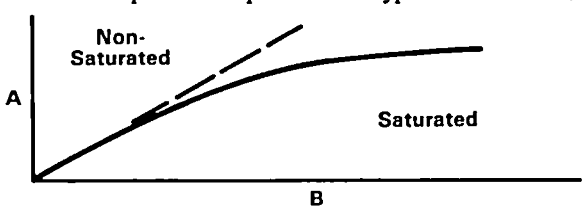
[66]Tr. 8637. This simply reflects the geometry of the situation. If, for a vertical fault, energy is released at 10km below the earth's surface, the total distance (D) of a point on the earth's surface to the energy release is related to the distance (R) to the surface expression of the fault in the following way:

R	D
0	10km
3	10.4
7	12
10	14
15	18
30	31
100	101



Within R = 0 to 10, there is only a modest change in D; For R greater than 10, R and D become nearly the same. For a very short rupture (*i.e.,* the length of rupture much less than R or D), the values R and D represent the epicentral and hypocentral distances, respectively.

[67]



[68]Bolt, Tr. 5920, 5876-79; Smith, Tr. 5916-17, 5470, 5889-90 (quoting Hanks-Johnson, see reference fn. 78, *infra*); Brune, Tr. 7928 (quoting Ambraseys); Trifunac, Tr. 8993; Smith, fol. Tr. 5490 at 9, R.Tr. 1261, 1272; Trifunac, fol. R.Tr. 1138 at p. II-2. *But see* Brune, fol. R.Tr. 601 at pp. 4-5; Newmark, fol. R.Tr. 534 at pp. 11-12.

process which causes the earthquake oscillations, a wide range of wave motion frequencies[69] is produced in the earth. The higher the frequency, the greater is the attenuation as the waves travel through the earth. While extremely high frequencies are present very close to the fault they are quickly attenuated, and waves in the range of 1 to 10 hertz (which are most likely to damage large buildings)[70] tend to predominate in the near-field area.[71] The lower frequency waves (less than 1 hertz) are attenuated very slowly and can be measured at great distances.[72]

Magnitude saturation of peak ground motion occurs at a near-field site because the higher frequency waves from distant portions of the rupturing fault are attenuated before they can contribute to motion at that site. Lower frequency waves from the entire rupturing fault may contribute to site motion, but they are not critical with respect to structural damage.[73]

(a) It is the low frequency waves (about 0.05 hertz) measured at great distances that are used to determine the surface wave magnitude (M_S) of an earthquake, and these measurements are indeed an indication of the total or integrated energy release along the entire length of the ruptured fault.[74] Closer to the source (within 600km), magnitude measurements are made using the M_L, or Richter, scale and are largely influenced by waves of frequency greater than a few cycles per second. This scale is considered to give the most appropriate measure of earthquake size for engineering purposes.[75]

The testimony of Dr. Enrique Luco submitted for the IV-79 reopened hearing provides us with a graph of values of the M_S of historical earthquakes plotted against values of M_L for the same earthquakes.[76] This figure shows that for smaller earthquakes, *i.e.*, below M_S or M_L of about 6, there is little difference between those values. But for high M_S values — that

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[69]These frequencies can vary from a low of 1 oscillation over many seconds to 100 or more oscillations per second.

[70]See D-LL 42, Figure 42-A — A figure on which the natural frequencies of the major structures at Diablo Canyon are shown to lie in the range 1-10 hertz. Throughout this opinion reference will be made to a series of reports which were prepared by the applicant on specific topics and submitted as a part of Appendix D to amendments 50 and 53 to the "Seismic Evaluation for Postulated 7.5M Hosgri Earthquake," (hereinafter cited as "Hosgri Report"), which is part of the Final Safety Analysis Report (FSAR). The reports are identified by the prefix D-LL, and a number. Citations to reports of this series shall be, D-LL xx, where xx refers to the specific report number (see Blume, fol. Tr. 6100, at 30-31). The FSAR was admitted below as part of App. Exh. A.

[71]Tr. 5528.

[72]See C.F. Richter, *Elementary Seismology*, p. 27 (1958).

[73]Tr. 5970, 5877; Trifunac, fol. R.Tr. 1138, p. II-2.

[74]Smith, fol. Tr. 5490 at pp. 12-13; Tr. 5523-32.

[75]*Ibid.*

[76]Luco, (follows Trifunac testimony fol. R.Tr. 1138), Fig. 1-1. (This figure is taken from an article by Kanamori, 69 Bull. Seism. Soc. Am. 1645-70 (1979)).

is for very severe earthquakes involving faulting over great lengths (M_S values as high as 8.2) — the measured M_L values are no longer equivalent to those of M_S but appear to reach a maximum in the $M_L = 7.2$ to 7.5 range. To be sure, there is some scatter in the data presented by Dr. Luco in this figure, but the existence of the saturation effect displayed by those data cannot be gainsaid.[77]

(b) Additional support for the concept of near-field motion saturation is provided by data presented in J.I. Exh. 47, a paper by Thomas C. Hanks and Dennis A. Johnson, referenced by several of the experts who testified.[78] The gist of the Hanks-Johnson report is that in the near field (which the authors defined essentially as we have done here) the high frequency acceleration from earthquakes is a function of physical processes related to the fault region near the measuring point and is not dependent upon earthquake magnitude.[79]

Figure 1 of the Hanks-Johnson paper plots measured peak near-field accelerations as a function of earthquake magnitude for a number of seismic events.[80] The figure suggests that the peak accelerations measured are at least to some extent dependent on the magnitude of the earthquake below M = 4.5. The authors conclude, after acknowledging that there is considerable scatter in the data available to them, that the data indicate little or no similar magnitude dependence above M = 4.5. With one exception (a measurement made at Pacoima Dam, February 9, 1971 for the 6.4M San Fernando Earthquake) of 1.15g,[81] the plot shows no peak acceleration values greater than approximately 0.8g — and such values were measured for earthquakes below as well as above M = 5.

In his testimony below, applicant's witness Dr. H. Bolton Seed presented a modified version of Hanks-Johnson Figure 1 (App. Exh. 61) on which he had plotted peak acceleration data for five additional earthquakes measured in the near field.[82] (These data were also presented in tabular form in App. Exh. 62.) Dr. Seed's data include a measurement at 5km (well within the near field) for the severe 1978 earthquake at Tabaz, Iran, of magnitude $M_S = 7.8$. Dr. Seed's additional information fully supports Hanks and Johnson's conclusion on saturation. The maximum acceleration of the new

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[77]Tr. 5970.
[78]J.I. Exh. 47, T.C. Hanks and D.A. Johnson, "Geophysical Assessment of Peak Accelerations," Bull. Seism. Soc. Am. 66, pp. 959-968 (1976).
[79]Tr. 5889-90.
[80]Both vertical and horizontal peak accelerations are plotted in this figure.
[81]Values of the peak acceleration at Pacoima Dam in this record vary from 1.15g to 1.25g (see, *e.g.*, Hanks and Johnson, fn. 78, *supra*, Table 1; Newmark, fol. Tr. 8552, Reference A, p. C-6; Hofmann, fol. Tr. 8522, p. 5).
[82]Tr. 10,104-07.

data was 0.95g — but this was measured close to an earthquake of magnitude 5.5; the highest acceleration otherwise measured was 0.8g.

At the reopened hearing, applicant's witness Dr. Blume presented evidence on magnitude saturation derived from IV-79. Figure 1-10 in Dr. Blume's testimony is a further refinement of the Hanks-Johnson figure. The new figure plots data obtained in the near field of the $M_S = 6.9$ IV-79 event.[83] In this case there were a large number of acceleration measurements made in the near-field region; again, the maximum acceleration measured in that region reached only 0.8g.

(c) A number of the witnesses (and the authorities they cite) supported the idea of magnitude saturation.[84] None, however, was willing to claim that the peak ground motion measurements displayed by the original Hanks-Johnson figure and its various refinements necessarily reflect the maximum peak motion possible in the near field.[85] There is agreement, though, that peak high frequency ground motion in the near field is primarily dependent upon the nature of fault rupturing in the vicinity of the measurement site.[86] There cannot be total assurance that the measurements made in the near field to date sample all conditions that might result in large local values of acceleration. Applicant's witness Dr. Bolt expressed the view, however, that the physical properties of the fracturing rocks themselves limit the seismic energy locally releasable.[87]

(d) Joint intervenors' witness Dr. James Brune did take a different view of the concept of magnitude saturation. He testified that "all statistical correlations available in literature indicate an increase in average peak accelerations, velocities and spectrum of ground motion with magnitude, with decreasing slope for larger magnitude."[88] Citing uncertainties in the

[83]R.Tr. 70; Joint Intervenor's Proposed Findings at p. 10. Staff witness Dr. Rothman reports that the USGS assigned values of $M_S = 6.9$ and $M_L = 6.6$ to the IV-79 event (fol. R.Tr. 536, pp. 4-5). These are the same values quoted in Staff Exh. R-1 (pp. 1, 4). Dr. Brune for the intervenors felt that the M_L value was in the range 6.2 to 6.6 (R.Tr. 757-58). Dr. Luco agreed with Dr. Brune that there may be some uncertainty with regard to the M_L value, and he also ascribed a range $M_S = 6.5$ to 6.9. (Luco, IV-79 testimony at pp. 1-2, 1-3.) Both of these witnesses, however, testified that the USGS used $M_S = 6.9$. Applicant's witness Blume used the value $M_L = 6.6$ (Blume, IV-79 testimony, p. I-2). We note also that the recently published USGS Open-File Report 81-365 (March 1981) also uses $M_L = 6.6$ for IV-79 (see p. 994, *infra*).

[84]Newmark, fol. Tr. 8552, Reference B at pp. 5-6; Smith, Tr. 5916-17, 5941; Bolt, Tr. 5920; Smith, quoting Hanks-Johnson, Tr. 5889-90; Brune, quoting Ambraseys, Tr. 7928; Trifunac, Tr. 8993; Newmark, fol. R.Tr. 534, pp. 11-12; R.Tr. 547.

[85]See, *e.g.*, Tr. 5895; Tr. 5940.

[86]Fn. 68, *supra*.

[87]Tr. 6026.

[88]James N. Brune, Testimony on Behalf of Joint Intervenors, fol. R.Tr. 601 at p. 7 (under Item D, "Extrapolation to M = 7.5").

data for large events, Dr. Brune went on to conclude that, on the average, peak acceleration values would be greater for M = 7.5 than for M = 6.5.[89] While Dr. Brune's position certainly is not an adoption of saturation, in actuality his testimony differs little from the ideas expressed by the other witnesses.

(e) We find that the physical description that has been developed for the nature of the earthquake motion in the near field and the data on peak ground motion that have been presented in these proceedings (*i.e.*, the Hanks-Johnson figure and its modifications) provide a convincing case for the concept of magnitude saturation. Put another way, we find that in the near field, peak high frequency ground motion is largely independent of earthquake magnitude.

This finding does not discount the possibility that future ground motion records may exceed those previously measured. Nor do we ignore the fact that larger magnitude earthquakes give rise to a greater probability of high peak measurements. (The latter point was in fact made in the Hanks-Johnson paper.[90]) As applicant's witness Dr. Smith explained, this is in part due to the sampling effect — there is a greater chance of having a recorder in the near-field, high-peak acceleration region of a large earthquake than a small one because the near field of a large event simply covers more ground.[91] He further explained that the higher probability was also due in part to physical processes, such as focusing and local inhomogeneities in the fault zone.[92]

The significant factor in our finding accords with the principal observation of Hanks and Johnson — that peak motion in the near field is determined primarily by magnitude-independent, fault-related processes.[93] This circumstance, we conclude, makes it appropriate to use measurements of severe, high frequency ground motion in the near field of smaller earthquakes to infer properties of such motion in the near field of larger seismic events. The reason for doing so is to obtain broader-based and hence more reliable data for the seismic evaluations that must be made here.

4. **Distance saturation.** Another form of peak ground motion saturation is associated with the earthquake types we have under review. As we have mentioned (p. 928, *supra*), the release of earthquake energy generally occurs

[89]*Ibid.* Dr. Brune also testified that the effects of focusing and high stress drop could result in peak ground acceleration values of 2g or higher (Tr. 7923-28). For further consideration of Dr. Brune's ideas in these areas, see pp. 944-951, *infra.*
[90]J.I. Exh. 47 at 964.
[91]Tr. 5938-39.
[92]*Ibid.*
[93]J.I. Exh. 47 at 963.

at some depth beneath the earth's surface (*e.g.*, 10km). At surface locations within the near field, therefore, there is little difference in the distance between the measurement site and the region of rupture. Thus, within the near-field region, the magnitude of peak ground motion should not be strongly affected by the distance to the surface expression of the fault (hence, distance saturation).[94]

In this connection, we asked the parties at the reopened hearing to provide us with peak acceleration data from IV-79 plotted with the empirical predictions of peak motion versus distance introduced previously before the Licensing Board.[95] Such graphs allow a comparison between experimental data and the empirical curves for predicting the intensity of earthquake motion as a function of fault distance.

The responses contained basically two general methods for plotting earthquake motion with respect to distance. The applicant's figures[96] and those in USGS Publication 795[97] (referred to in our question to the parties) both plot the data from various locations as a function of the distance to the *nearest surface expression of the rupturing portion of the fault.* Graphs of this type accept the concept that the significant distance for determining the attenuation of high frequency motion is that to the nearest portion of an extended region of energy release.

Board witness Dr. Mihailo Trifunac and intervenors' witness Dr. James Brune, however, preferred to plot the peak acceleration data as a function of the distance to a particular point. In Dr. Trifunac's graphs,[98] that point was the postulated seismic event's "epicenter," *i.e.*, the point on the earth's surface directly above the portion of the fault where the rupture initiated.[99] Dr. Brune's graphs were made with respect to a hypothesized "zone of energy release," a region of limited spatial extent believed by him to be somewhat removed from the epicenter (illustrated in J.I. Exh. R-12).[100]

Figure I-1 of Dr. Blume's IV-79 testimony and Figure I-2 of Dr. Seed's IV-79 testimony on Board Question 1 are comparisons of the IV-79 data

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[94]See fn. 66, *supra;* Tr. 5922, Tr. 8637.

[95]See ALAB-598, *supra,* 11 NRC at 888; Board Question 1.

[96]Blume, IV-79 Testimony on Question 1, Figures I-1, I-5, I-6, I-7, I-8, I-9; Seed, IV-79 Testimony on Question 1, Figure I-2.

[97]J.I. Exh. R-1.

[98]Trifunac testimony fol. R.Tr. 1138, Figures I.1, I.2, I.3.

[99]See Bolt, *Earthquakes,* fn. 24, *supra,* at 226.

[100]Dr. Brune's graphs (J.I. Exhs. 13, 14 and 15 for identification) were not admitted into evidence because of the uncertainty of the points plotted with respect to the unknown center of energy release. See R.Tr. 877-887.

with predictions based on earthquake models constructed by Dr. Blume (SAM-V)[101] and Dr. Seed, respectively. Both models predict distance saturation in the near field, and in both cases there is reasonable agreement between the IV-79 data and the model predictions. Dr. Smith testified that the IV-79 data demonstrate the validity of plotting peak ground motion data with respect to the nearest point on the ruptured fault as the technique that best accounts for the known physical nature of the faulting process.[102]

On the other hand, the graphs of measured peak acceleration as a function of distance to a point (*e.g.*, epicenter or zone of energy release) display behavior inconsistent with the general characteristics of the attenuation of earthquake motion predicted by physical principles.[103] Both Dr. Trifunac's and Dr. Brune's graphs contain collections of data points having a wide spread of acceleration values associated with a narrow range of distance.[104] These graphs appear to be disproportionately influenced by the particular arrangement of the seismographs aligned across the Imperial Fault rather than to reflect an accurate representation of physical attenuation processes at work.

In sum, our review of the IV-79 data displayed in the various graphs confirms the view that, in a major fault-related earthquake, energy is released at depth along an extensive length of the fault. The data support the concept that peak ground accelerations in the near field saturate and are not strongly dependent on distance.[105] This is consistent with the present understanding of physical earthquake mechanisms.[106] It follows that empirical models of earthquake motion incorporating this concept are apt to be reliable tools for developing response spectra suitable for designing structures to be located in a potential near-field site.

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[101]SAM is an acronym for the Site-Acceleration-Magnitude procedure formulated by Dr. Blume for estimating the relationships of site, materials, horizontal peak accelerations, magnitude, and epicentral distance. SAM-V is the latest modification of the original SAM procedure. See D-LL 11B.

[102]R.Tr. 52-59.

[103]R.Tr. 59-61; App. Exhs. R-3 and R-4; R.Tr. 110.

[104]R.Tr. 568.

[105]An obvious exception, of course, would be at locations directly upon the surface expression of the fault itself. Here ground motion associated with the displacement of the fault may occur. Tr. 6027.

[106]Dr. Blume's IV-79 testimony on Question 1, Figure I-6, shows the IV-79 peak acceleration data displayed on Figure 4 of USGS Circular 795, J.I. Exh. R-1. The latter provides prediction intervals (± 70%) of peak motion data for three magnitude categories (5.0-5.7, 6.0-6.4, and 7.1-7.6). Prediction intervals for magnitude categories are shown for defined distance ranges. Only the intervals for the 5.0-5.7 category are defined within the near field (*i.e.*, less than 10km). The IV-79 data for distances within 10km fall within the 5.0-5.7 prediction interval. This observation is consistent with the concept of magnitude saturation in the near field.

C. Methodology Used to Predict Ground Motion for the Diablo Canyon Site and the Development of the Response Spectrum in the Hosgri Reanalysis

The basic parameters of the ground motion at the Diablo Canyon site as a result of a 7.5M earthquake on the Hosgri Fault were established by the USGS, whose recommendations in this area were generally adopted by the NRC staff. The USGS recommendations included a table of peak ground accelerations for various magnitude earthquakes. For the Diablo Canyon reanalysis, however, the staff and applicant adopted a ground response spectrum anchored at a high frequency acceleration lower than the highest peaks shown in Table 2 of USGS Circular 672 (J.I. Exh. 45).[107] Intervenors complained that, at least, the USGS peak acceleration value should have been used to anchor the response spectrum.[108]

As discussed earlier, the device used for the design of a structure to be subjected to earthquake tremors is a response spectrum generated for free-field ground motion. This provides a measure of the maximum ground motion over a frequency range encompassing the natural frequencies of systems and structures designed to withstand seismic loading. For nuclear power plants sited beyond the near field, the usual methodology is to assign a peak high frequency ground acceleration.[109] This peak acceleration then becomes a scaling factor by which a standardized response spectrum[110] may be adjusted to reflect ground motion conditions anticipated at the site.[111] The standard response spectrum was developed by averaging over individual response spectra generated from ground motion records taken at a number of sites in the United States and is given in Regulatory Guide 1.60, a document which also outlines how this methodology is to be applied.[112]

Although the standard response spectrum of Regulatory Guide 1.60 was generated conservatively, the data that went into it was recorded mainly at far-field locations.[113] The Diablo Canyon site, however, is situated in the

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[107]See discussion of this at pp. 936-938, *infra.*

[108]LBP-79-26, 10 NRC at 486; J.I. Br. at pp. 16 *et seq.*

[109]Applicant's witness Dr. Frazier in his oral testimony (Tr. 10,127-10,129) pointed out that high frequency ground motion (frequencies greater than 2 hertz) is attenuated very rapidly in the shallow layers of the earth. This attenuation can be as high as a factor of 10 in one kilometer of near-surface travel for a wave with a frequency of 10 hertz. However, these high frequency waves do travel with low attenuation in the deep, more competent rock and reach the surface at fairly steep angles by diffraction from the low lying rock strata. Thus, some high frequency motion may be present at a large distance from the source but except in the very near field where horizontal components are still unattenuated, the high frequency waves found at the surface arrive at steep angles.

[110]See fn. 40, *supra.*

[111]Tr. 8328.

[112]Tr. 8583-86.

[113]*Ibid.*

near field of the Hosgri Fault and on a rock foundation. Because of this, the staff and the applicant concluded that the Regulatory Guide 1.60 spectrum might not be the best representation of the ground motion to be expected at the Diablo site from a 7.5M event on the Hosgri. They therefore elected to adopt a methodology for developing a spectrum to be used in the Hosgri reanalysis that, in their judgment, would correspond to the actual Diablo Canyon site conditions.[114]

The decision to elect a different methodology was neither impermissible nor inconsistent with Regulatory Guide 1.60. The guides, advisory rather than obligatory,[115] explain on their face that they "are issued to describe and make available to the public methods acceptable to the [NRC] Regulatory staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission."

As we mentioned, the event specified to control the seismic response of the Diablo Canyon facility is a 7.5M earthquake on the Hosgri Fault adjacent to the facility. This magnitude was suggested by the USGS, despite the opinions of the applicant and the staff that Hosgri probably would not be the site of an earthquake of this size.[116]

In addition to specifying the magnitude of the earthquake to be expected on the Hosgri Fault, the USGS advised the NRC staff with regard to the peak ground acceleration that might be expected in the near field of such an event. Its recommendation is included as Appendix C to the Safety Evaluation Report (SER) Supplement 4, issued May 1976. In conclusion 7 (p. C-16), the USGS states:

> Consequently, we feel that an appropriate earthquake for this site should be described in terms of near-fault horizontal ground motion. A

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[114]Newmark, fol. Tr. 8552, Reference B at p. 6; Tr. 8639.

[115]*Porter County Chapter* v. *AEC*, 533 F.2d 1011, 1016 (7th Cir.), *certiorari denied*, 429 U.S. 945 (1976).

[116]Stepp, fol. Tr. 8484, p. 31; Blume, fol. Tr. 6100 at p. 16. The previous discussion regarding the saturation of earthquake ground motion with respect to earthquake magnitude leads at most to a weak correlation between magnitude and peak acceleration. Thus, the contest over the size of the earthquake to be expected on the Hosgri loses some of its significance. The magnitude of the event is important, however, to the extent that it effects the likelihood that the Diablo Canyon site would be in the near field of that portion of the fault which ruptures during the earthquake. Clearly, the Diablo Canyon facility is more apt to be adjacent to a long rupture along the Hosgri Fault than to a short one.

technique for such a description is presented in the Geological Survey Circular 672 entitled "Ground Motion Values for Use in the Seismic Design of the Trans-Alaska Pipeline System" [1972].[117]

In particular, Table 2 of Circular 672 is labeled "Near-fault horizontal ground motion." In that table are presented values of peak absolute horizontal accelerations that might be expected for earthquakes of varying magnitudes.[118] For the ground motion record that might be generated by a 7.5M earthquake, Table 2 indicates a high peak of acceleration of 1.15g; a second highest peak of 1g; a fifth largest peak of 0.85g; and a tenth largest peak of 0.65g. Presumably, then, the table could be used to generate a simulated ground motion record of a 7.5 magnitude earthquake.

In discussing how the ground motion values of Table 2 could be used to determine a structural response spectrum (for the design of the Alaska pipeline), the authors of Circular 672 suggest that design values of motion be derived by modifying the ground motion values of Table 2 to allow for various mechanisms in the vibratory response of a structure. The USGS reply to the NRC recommending the use of Circular 672 (see pp. 937-938, *supra*) similarly explains that:

> It is our intention that the ground motion values as exemplified by Table 2 "Near-fault horizontal ground motion" of Ref. 4 [*i.e.*, USGS 672] for magnitude 7.5 be used to form the basis of a description of the earthquake postulated to have the potential for occurring on the Hosgri Fault at a point nearest to the Diablo Canyon site subject to the conditions placed on these values in Ref. 4. *The earthquake so described should be used in the derivation of an effective engineering acceleration* for input into the process leading to the seismic design analysis.

Ibid. (emphasis added).

Thus, the USGS recommended not only the magnitude of the earthquake, but also a maximum peak ground acceleration for such an event and the use of some form of "effective acceleration" as the basis for a response spectrum to be used in the reanalysis.[119] Although, as we have noted, the "effective acceleration" concept is one that the joint intervenors and the

[117] J.I. Exh. 45.

[118] The peak ground motion values displayed in USGS Circular 672 show a definite increase in expected peak ground motion as a function of earthquake magnitude, although there is a decreasing rate of increase for the larger magnitude events (*i.e.*, there is no strong saturation with magnitude indicated for earthquakes of magnitude up to 7.5).

[119] We note *inter alia* that in a concluding sentence of the section entitled "Design Approach" (p. 3), the authors of USGS Circular 672 state: "Finally, smoothed tripartite logarithmic response spectra are constructed from the design seismic motions by the general procedure of Newmark and Hall...."

Governor find objectionable, it is the approach suggested by the USGS both in its original recommendations to the NRC and again supported in the testimony of a USGS official before the Licensing Board.[120] On the other hand, the USGS made no suggestion regarding how that effective acceleration might be determined or what its magnitude might be relative to the peak ground motion values presented in Table 2.

Our review of the record below discloses that it was the Nuclear Regulatory Commission staff, guided by its consultant and primary witness Dr. Nathan Newmark, who developed the basic design response spectrum to be used for the Diablo Canyon reanalysis. This process is described in Dr. Newmark's direct testimony (fol. Tr. 8552, Reference A).[121]

It is not entirely clear how the anchor point acceleration of 0.75g ultimately settled upon for the basic response spectrum was actually obtained. Dr. Newmark referred to a relationship provided in a paper by N.C. Donovan relating ground motion to distance from an earthquake source which yielded a maximum ground acceleration of 0.75g.[122] On the other hand, further reading of Reference A reveals that he generated response spectra for the ground motion recorded at Pacoima Dam (a record which includes a peak acceleration of 1.2g, somewhat larger than recommended for a 7.5M event in Table 2 of USGS Circular 672). He then compared these spectra with an idealized design spectrum that was roughly equivalent to the standard spectrum appearing in Regulatory Guide 1.60. He then observed that the Pacoima Dam spectrum was virtually bounded by the idealized spectrum when the latter was anchored at the high frequency end at 0.75g.[123] Except for frequencies in excess of about 13 cycles per second and for very small excursions at several somewhat lower frequencies, the Pacoima Dam response spectrum lies below the idealized Newmark spectrum anchored at 0.75g.[124] Because the Pacoima Dam spectrum was generated using ground motion records having the largest peak horizontal acceleration measured,[125] Dr. Newmark explained that the idealized spectrum anchored at 0.75g was, in his opinion, a conservative upper limit of the motion that might be expected at Diablo Canyon and indeed was responsive to the USGS recommendations that the peak ground

[120] Devine, Tr. 8330-31.

[121] Reference A of Dr. Newmark's testimony was previously published as Appendix C to Supplement 5 of the Diablo Canyon Safety Evaluation Report (SER). Although in his direct testimony Dr. Newmark referred to an effective acceleration, under cross-examination he stated that he did not employ the concept. Tr. 9287, 9275-76, 9321; R.Tr. 544.

[122] Newmark, fol. Tr. 8552, Reference A, at C-3.

[123] *Id.* at C-4, also Figures 1A, 1B, 18 and 19.

[124] Tr. 9275-76, 8589-90.

[125] Newmark, fol. Tr. 8552, Reference A, at p. C-4.

motion of 1.15g be "associated" with a 7.5M earthquake on the Hosgri Fault.[126]

In his direct testimony, applicant's witness Dr. John Blume outlined a different approach which he used to determine the response spectrum for the Diablo Canyon reanalysis (fol. Tr. 6100 at pp. 39-41). Dr. Blume generated a response spectrum from each of several strong motion records obtained from relatively intense near-field records on rock sites. From these records he generated a smooth idealized average spectrum and scaled the amplitude of the resulting spectrum to a high frequency anchor point acceleration of 0.75g.[127]

The results obtained by Dr. Newmark for the staff and Dr. Blume for the applicant yielded basic design spectra (and spectra modified for the tau effect) not greatly different from one another. In employing those spectra for the Diablo Canyon reanalysis, the staff's approach was to use the one which gave the largest magnitude of motion for the particular frequency range and structure being considered; in other words, the staff adopted the highest magnitude portions of both the Newmark and Blume design spectra.[128]

The Licensing Board approved this methodology (LBP-79-26, 10 NRC at 486-89); we turn now to the joint intervenors' exceptions to that portion of the decision below.

IV

INTERVENORS' CHALLENGE TO THE APPROPRIATENESS AND VALIDITY OF THE BASIC DESIGN SPECTRUM DEVELOPED FOR THE DIABLO CANYON REANALYSIS

A. The Use of Effective Acceleration

The gravamen of intervenors' complaint involves the design spectrum used for the Diablo Canyon reanalysis. In their view, to represent the maximum vibratory ground motion associated with a 7.5 magnitude earthquake on the Hosgri, that spectrum should at least have been

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[126]Having thus established a basic response spectrum, Dr. Newmark then allowed for a reduction in the response in the higher frequency portion of this spectrum, the so-called "tau effect," for certain of the larger structures at the Diablo Canyon site. The tau effect reduction is also considered to be inappropriate by the intervenors; the entire question of its validity is discussed at pp. 962-978, *infra*.

[127]Dr. Blume seems to have accepted Dr. Newmark's recommendation of a 0.75g as an anchor point. He agreed that the concept of effective acceleration was valid, but added in his opinion that a lower value of effective acceleration would have been more appropriate for the Diablo Canyon site. Tr. 6495.

[128]Tr. 6836; Tr. 8594; Knight, fol. Tr. 8696 at p. 10. Also see pp. 975-976, *infra*.

anchored at the 1.15g peak ground acceleration value given in Table 2 of USGS Circular 672.[129]

Their first objection goes to the use of a 0.75 anchor point acceleration, a challenge which they base on both legal and technical considerations. First they assert this value does not satisfy the requirements of Appendix A to 10 CFR Part 100 ("Seismic and Geologic Siting Criteria for Nuclear Power Plants"). Section VI of that Appendix ("Application to Engineering Design") provides in pertinent part:

> (a) *Vibratory Ground Motion* — (1) *Safe Shutdown Earthquake.* The vibratory ground motion produced by the Safe Shutdown Earthquake shall be defined by response spectra corresponding to the *maximum vibratory accelerations* at the elevations of the foundations of the nuclear power plant structures * * *. (Emphasis supplied.)

The 1.15g figure is the value of peak acceleration for a 7.5M earthquake as given in Table 2 of USGS Circular 672. It does not necessarily follow that this figure corresponds to the "maximum vibratory acceleration" mentioned in Appendix A to Part 100. As we have explained before, "under any rule of reason, however, that requirement must be understood to have reference to *effective* maximum acceleration."[130] In the *Seabrook* hearing, "Dr. Newmark testified without contradiction that the highest acceleration peaks are associated with the highest frequency ground waves. These high frequency waves would be fully recorded by the relatively small and compact seismographs, but yet would have no significant effect on the large massive structures of a nuclear facility."[131] In short, the scientific validity of the effective acceleration concept to one side, the Board below correctly distinguished between measurements of peak acceleration and effective acceleration. See LBP-79-26, 10 NRC at 486.

A more fundamental objection is the argument, advanced by Dr. Enrique Luco, that there exists no physical basis for a reduction from the peak measured acceleration to an effective acceleration.[132] Testifying for joint intervenors in the Licensing Board hearing, Dr. Luco offered the following criticism in response to a question by intervenors' counsel (Tr. 8893):

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[129] J.I. Exh. 45.

[130] *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), ALAB-422, 6 NRC 33, 63 (1977) (emphasis in original), *reversed on other grounds*, CLI-80-33, 12 NRC 295 (1980).

[131] *Ibid.*

[132] Tr. 8867-8895. Another witness, Dr. Mihailo Trifunac, expressed similar doubts about "the concept of effective acceleration." Tr. 8973.

Q. * * * Could you indicate what your problem is with the use of the effective acceleration values that have been designated for the response spectra? I think in this case the effective acceleration which serves as a zero period limit for the Newmark spectra is 0.75g.

A. The main problem I have with this reduction for effective acceleration is that no sound physical basis has been given for that reduction. The only argument that I could consider — and it has been mentioned — is that in many cases the structures seem to, or appear to be stronger than they were designed for. Or that the structures that in the paper are supposed to fail do not fail.

Expert witnesses who testified on behalf of the staff and applicant, however, strongly supported the use of an effective acceleration. Applicant's witness Dr. Blume explained that peak ground motion records often reflect spikes of short duration. He stressed that these have little energy associated with them and simply have no structural significance or impact on a response spectrum.[133] He backed his statements with detailed analyses showing that "clipping" high acceleration peaks from time history of motion records has but minor effect on the response spectra developed from those records.[134] He also relied on empirical evidence to support his point that high acceleration spikes are not significant from the standpoint of building damage. Dr. Blume referred to observations of a number of structures that survived in the near field of earthquakes despite peak acceleration measurements at levels where damage would have been expected were those spikes truly indicative of effective acceleration.[135] Applicant's witness Dr. Bruce Bolt agreed with Dr. Blume that the very high frequency peak accelerations have little significance for structural design.[136]

Moreover, in the Pacoima Dam record, where the recorded peak accelerations include an acceleration spike of 1.2g, Dr. Newmark's response spectrum (anchored at 0.75g) virtually enveloped the response predicted by the measured ground motion,[137] indicating that the high peak acceleration had little or no effect on the calculated response to the Pacoima Dam event. (Those calculations also provide the quantitative basis for the assignment of the anchor point or effective acceleration at 0.75g.)[138]

[133]Blume, fol. Tr. 6100 at p. 19.
[134]*Ibid;* see also D-LL 30.
[135]Blume, fol. Tr. 6099 at pp. 21-25. Dr. Newmark made similar observations at Tr. 8638-39.
[136]Tr. 5846-48
[137]See p. 939, *supra.*
[138]See pp. 939-940, *supra.*

Finally, the USGS itself accepts the use of effective acceleration in seismic design. The concept was included in that agency's recommendation to NRC that the peak acceleration values of USGS Circular 672 be employed in the Diablo Canyon reanalysis.[139] And at the Licensing Board hearings, a USGS witness called by the staff, Dr. James Devine, expressly represented that his agency approved the concept of effective acceleration as an accepable approach to deriving response spectra for nuclear power plants.[140]

In the circumstances, for the reasons discussed we find that the use of an anchor point or effective acceleration lower than the peak ground acceleration is a physically valid and acceptable procedure for structures in the near field. In our judgment, this finds particular confirmation in the fact that the Newmark spectrum, anchored at 0.75g, virtually envelopes the response spectra generated by ground motion records containing an acceleration spike of 1.2g.

B. Use of the Pacoima Dam Record

Board witnesses, Drs. Luco and Trifunac, objected to Dr. Newmark's use of the Pacoima Dam record as the basis for his redesign response spectrum. They pointed out that the 1971 San Fernando earthquake (which produced the Pacoima Dam record) had a magnitude of 6.5 whereas the Hosgri event specified for the Diablo Canyon SSE is an earthquake with a magnitude of 7.5.[141] In their view, the spectrum developed using the Pacoima Dam record would be appropriate only if the plant were to be designed for a 6.5M event.[142] (Dr. Trifunac, however, added his opinion that 6.5 rather than 7.5 is the appropriate magnitude to expect at Diablo Canyon from a Hosgri earthquake. On this basis he accepted the basic Newmark design spectrum — *i.e.*, uncorrected for the tau effect — as adequate for the plant.[143]) Both witnesses expressed the belief that, for a 7.5M event, the response spectrum should be anchored at the 1.15g peak acceleration value given in USGS Circular 672.[144]

We discussed in the previous section (pp. 938-940, *supra*) the rationale for using the Pacoima Dam record to characterize a 7.5M event. That ground motion record has the highest peak horizontal ground acceleration

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[139]See pp. 938-939, *supra.*

[140]Tr. 8332-33. Dr. Devine did not endorse the precise anchor point chosen by Dr. Newmark. He indicated that as USGS geophysicists it was their duty to provide engineers with ground motion data, without presuming how these data would be used to anchor ground motion spectra (Tr. 8331).

[141]See J.I. Br. at pp. 27-28.

[142]Licensing Board Exh. 2C (Dr. Luco's comments to the ACRS dated May 30, 1978) at p. 1.

[143]Tr. 8971, 8985.

[144]Tr. 8872-77 (Dr. Luco); Tr. 8974-75 (Dr. Trifunac).

ever measured, even though there have been several recordings made in the near fields of larger earthquakes.[145] Our previous findings regarding near-field saturation of peak acceleration with magnitude (see pp. 92?-933, *supra*) suggest that the Pacoima Dam record is characteristic of the strongest horizontal motion in the near field of any large earthquake. If anything, the Pacoima Dam record is widely believed to overstate the maximum ground motion in the 1971 San Fernando event because the record was taken on a ridge which, acting like a structure, tended to magnify the acceleration.[146] Finally, the San Fernando earthquake was generated by thrust faulting, which causes near-field motion of greater magnitude than would an event triggered by a strike-slip fault like the Hosgri.[147]

We find these factors sufficient to justify the use of the Pacoima Dam record as the basis for a near-field ground response spectrum for a 7.5M event and that the procedures followed on that basis took proper account of the USGS' recommendations.[148]

C. Focusing and High Stress Drop

Joint intervenors also contend that, even were the Diablo Canyon response spectrum anchored at 1.15g, it would not be conservative. They assert that in some regions, "focusing,"[149] "high stress drop," or both

[145] Tr. 10,103-105.

[146] See Smith fol. Tr. 5490 at p. 26; Tr. 8529; Tr. 10,085-88, 10,093-94; Tr. 10,104-05; Int. Exh. R-1 (USGS 795) at p. 25; D-LL 12.

[147] Tr. 8617-18. In response to a question on cross-examination Dr. Newmark explained the physical basis for expecting greater motion from a thrust fault than from a strike-slip fault like the Hosgri. Tr. 8624. See also p. 933, *supra*.

[148] Witnesses for the applicant and the staff testified that the USGS Circular 672, Table 2 peak acceleration values were extremely conservative and, in their view, included an unjustifiably strong magnitude dependence of peak acceleration (see, *e.g.*, Bolt, Tr. 5874-78; Smith, fol. Tr. 5490 at pp. 27-28; Tr. 5470; Newmark, Tr. 9287-89; Hofmann, Tr. 8538).

IV-79 provided a statistically significant set of near-field data for evaluating the Circular's predictions; we reviewed them for that purpose. Table 2 of USGS 672 presents values of expected peak near-field horizontal acceleration for earthquakes of various magnitudes (presumably M_S magnitudes because the scale goes up to M = 8.5; see pp. 930-931, *supra*.) For M = 6.5 and M = 7.0, the predicted highest peak accelerations set forth in the table are 0.9g and 1.05g respectively.

Figure I-1 of Dr. Blume's IV-79 testimony on Question 1 is a plot of 35 values of uncorrected peak accelerations recorded during IV-79 within 10km of the fault. The highest peak, recorded at the Bond's Corner station (3km), is 0.81g. The next highest peaks shown are one at 0.71g (1km) and two at about 0.65g (3-4km). Nine of the 35 values exceed or equal 0.5g. Thus for 35 near-field recordings of peak ground motion for an earthquake of magnitude M = 6.5 - 7.0 (see fn. 83, *supra*), not a single peak acceleration falls into the range 0.9g to 1.05g predicted by Table 2 of Circular 672. The IV-79 data thus lend support to the position that Table 2 predictions of near-field peak horizontal accelerations are conservative, *i.e.*, they tend to predict overly high results.

[149] Focusing or directivity (we believe the terms may be used interchangeably) refers to an

together could produce accelerations of 2g or more. Intervenors' case on this point was presented principally through their witness Dr. James N. Brune.[150] The Licensing Board, however, rejected Dr. Brune's evidence as speculative. Noting that focusing "is not a new phenomenon," the Board stressed *inter alia* that only two instances of acceleration in excess of 1.15g have ever been recorded and neither approached the 2g level hypothesized by intervenors' witness. LBP-79-26, 10 NRC at 489.

In addition, the Governor (in the reopened hearing) asserts that the focusing and stress drop phenomena are important new safety issues raised for the first time by Dr. Brune. He therefore argues that, under our *Indian Point* decision,[151] the Licensing Board erred in not seeking further evidence to resolve them.

1. **Focusing.** In his prepared testimony before the Licensing Board, Dr. Brune stated that focusing "can lead to accelerations and velocities amplified by more than a factor of 2 in a sector of about ± 5° from the direction of fault propagation."[152] He added that, as a result of the direction of strike in the Hosgri Fault northwest of the site, "[e]nergy released about 20km up the fault could be focused nearly directly at the Diablo Canyon site."[153] (Because the Hosgri changes direction as it approaches closer to the site, seismic energy released within 20km of Diablo Canyon would presumably be directed in a focusing zone which does not include the facility. See p. 919, *supra*, Fig. 3.)

Dr. Brune's direct testimony on focusing was essentially theoretical. It included several analytical and laboratory studies by himself and others in which the phenomenon had been demonstrated.[154] As actual examples of focusing, however, Dr. Brune cited only the 1971 San Fernando (Pacoima Dam), and 1940 Parkfield earthquakes. He testified that, in those events, focusing enhanced the velocity of the seismic ground motion.[155]

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enhancement of vibratory earthquake motion along the direction of fault propagation. Staff witness Dr. Robert L. Rothman explained in the reopened hearing that "[t]he focusing effect results from constructive interference of signals whose velocity is close to that of the rupture propagation velocity." Fol. R.Tr. 536 at p. 13. His testimony contains an illustrative figure taken from an original work on the subject. *Id.* at Fig. 4. The effect was also described as a seismic Doppler effect. Frazier, IV-79 testimony at p. VII-2. See also Tr. 5878-82.

[150]J.I. Exh. 66 at 3-10 *et seq.* In the reopened hearing on IV-79, Dr. Brune appeared as the Governor's witness on this issue, fol. R.Tr. 601.

[151]*Consolidated Edison Co. of New York* (Indian Point, Units 1, 2 & 3), ALAB-319, 3 NRC 188 (1976).

[152]J.I. Exh. 66, at 3-2.

[153]*Id.* at 3-13.

[154]*Id.* at 3-10 to 3-13.

[155]*Id.* at 3-10. As we have mentioned, (fn. 70, *supra*), the evidence indicates that the important structures and systems at Diablo Canyon have relatively high natural frequencies. At high frequencies, the critical seismic motion parameter is not velocity or displacement but acceleration. Blume, IV-79 testimony at pp. III-2 to III-3. Neither point is in dispute.

During questioning at the hearing below,[156] Dr. Brune acknowledged that in the cases of the earthquakes at Parkfield (Ms 6.3) and 1940 Imperial Valley (Ms about 7), seismic recordings within 20 km of the rupturing faults and in the direction of their propagation revealed no unusual values of *acceleration* due to focusing.[157] Dr. Brune testified that the phenomenon of seismic focusing had been demonstrated, but he added that it was not known how effective it is in actuality.[158] He agreed that focusing was more apt to be noticed for waves of low frequencies than for high frequencies.[159] Finally, he asserted that additional field evidence is needed to demonstrate the significance of focusing and that further analytical modeling of the phenomenon is called for.[160] On redirect examination, Dr. Brune admitted that the model he used as one basis for his focusing predictions could not be claimed to be reliable and that the calculations should be made more realistically.[161]

With regard to the IV-79 earthquake (which we reopened the record to consider), the Imperial Fault ruptured right through an array of seismographs. Dr. Brune agreed that there was no clear evidence of focusing at this seismic event.[162] He suggested, however, that further analyses of the data and rupture mechanism might show that the IV-79 earthquake would be "better represented as a sequence of multiple events than as a continuous rupture," and, if such were the case, "that focusing from a more continuous rupture would have led to even higher accelerations."[163]

The applicant and staff dispute Dr. Brune's position on the importance of focusing. In their view, to the extent that focusing has a significant effect in actual events, it is already accounted for as part and parcel of the seismic records that already exist.[164] Their witnesses testified, moreover, that no

[156]Tr. 7941-8084, 8114-8129.

[157]Tr. 8030-8039; see also J.I. Exh. R-6, Table 3-3 and Table 3-4. Applicant's witness Dr. Frazier agreed that focusing was recorded at Parkfield, where the peak acceleration at the station near the fault (#2) was 0.5g (IV-79 testimony, p. VII-3).

[158]Tr. 8011-12; see also Tr. 8028-30, 8041-42.

[159]Tr. 8016-17.

[160]Tr. 8028-29, 8042, 8091, 8104.

[161]Tr. 8135-36. At this point Dr. Brune also volunteered the modeling technique of Archalita and Frazier as being capable of helpful calculations. Tr. 8138.

[162]Brune, fol. R.Tr. 601 at p. 8.

[163]*Ibid.*

[164]Tr. 5878-81 (Bolt); Rothman, fol. R.Tr. 536 at p. 13; Edwards, IV-79 testimony, VII-3; Smith, fol. Tr. 5490 at p. 26. While Dr. Brune agrees that focusing is a part of all earthquakes, he holds to the opinion that, to this date, records may not have been made in the maximum direction of focusing. Tr. 8075.

The matter of focusing in the Pacoima Dam record has led to considerable controversy (see, *e.g.*, Gov. Proposed Findings, pp. 56-57). The subject was considered in detail before the Licensing Board (cross-examination of Drs. Bolt and Frazier, Tr. 5880-5887). A review of that discussion reveals that a study by T.A. Eaton, with which those applicant witnesses appeared accelerations as high as those expected by Dr. Brune have ever been measured, adding their judgment that none is reasonably likely to occur.[165] They further stressed that the position of Diablo Canyon site with respect to the Hosgri Fault was such that focusing effects are not to be expected.[166]

The question whether the Diablo Canyon site is situated so as to experience the focusing of a rupture on the Hosgri Fault was strongly contested. Under cross-examination before the Licensing Board, Dr. Brune concluded that the probability of focusing at any particular site is low.[167] At the reopened hearings, he suggested that his testimony before the Board had been misconstrued — in particular, that the restriction of focusing to a ± 5° sector was too stringent and that different maps of the fault yield different distances between the Diablo facility and that portion of the fault aligned for focusing toward the site.[168] Our review of the record on this question leads us to conclude that focusing of earthquake motion due to a rupture on the Hosgri Fault does not present a credible likelihood of exceeding the Diablo Canyon seismic design spectrum. We are guided to this result primarily by the fact that the focused motion must travel some 20km[169] to reach the site and that the damaging higher frequencies of this motion will be preferentially attenuated in traveling this distance.[170]

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to agree, hypothesized that in the early stages of the 1971 San Fernando event, there was rupturing toward Pacoima Dam with considerable focusing, which caused a large velocity pulse (a "fling") in the record (Tr. 5884). The peak acceleration associated with this pulse was 0.4g (Tr. 5886). The large high frequency peak acceleration of about 1.20g, for which the Pacoima record is best known, occurred later in the ground motion record and was not related to the focusing (Tr. 5885-87; see also J.I. Exh. 47, fn. 78, *supra*).

In developing a response spectrum from the Pacoima Dam record, the motions included in the entire record were utilized in the procedure (Newmark, fol. Tr. 8552, Reference A, pp. C-3 to C-10; Reference B, pp. 5-6; also see pp. 939-940 *supra*). Thus the spectrum incorporates the effects of both the velocity pulse due to focusing and the high frequency peak acceleration.

[165]Tr. 5878; see also J.I. Exh. 47.

[166]Smith, fol. Tr. 5490, p. 27; Edwards, IV-79 testimony, pp. VII-3 to 4; Frazier, IV-79 testimony, p. VII-12.

[167]Tr. 8140-44. See also Hamilton-Jahns, fol. Tr. 4459 at Fig. 44.

[168]R.Tr. 619-623.

[169]See p. 945, *supra* (Dr. Brune's own estimate).

[170]Edwards IV-79 testimony, p. VII-4. At the IV-79 hearing, Dr. Brune testified that motion from an apparent short splay of the Hosgri Fault, ending some 3.8km from Diablo Canyon, could be focused directly at the site (R.Tr. 623). Applicant's witness Mr. Hamilton offered the opinion that such a splay would not participate in a major earthquake on the Hosgri (R.Tr. 321-22). Dr. Frazier for the applicant was of the opinion that the splay would be capable of generating little more than a magnitude 3 event (R.Tr. 317). And Dr. Devine of the USGS, testifying for the staff, said that he did not believe the feature existed, and thus did not consider it "capable" (R.Tr. 939-40). We do not believe the splay represents a reasonable source of earthquake motion beyond those already being considered in the plant's design.

At the reopened hearing, both the staff and the applicant presente(testimony (and Dr. Brune agreed) that the peak horizontal acceleratio values of IV-79 show no indication of focusing.[171] The applicant's principa witness on focusing phenomena in general was Dr. Gerald Frazier.[172] Dr Frazier outlined his analytical simulation of earthquake motion on th Hosgri Fault. He used a model developed by the TERA Corporation tha followed procedures he helped to develop that are described by him i published papers.[173]

Dr. Frazier's calculated results for values of various types of ruptur along the Hosgri indicate that some focusing effects are possible at Diabl Canyon. For the hypothetical ruptures considered, however, at least two o which are greater than 7Ms,[174] the mean peak horizontal accelerations h calculated never exceeded 0.58g, and the values of peak acceleration wer relatively independent of the length of fault assumed to be involved in th postulated event.[175]

Although Dr. Frazier's calculational model was of the type that Dr Brune had himself suggested to the Licensing Board be used,[176] Dr. Brun testified before us that "[u]nfortunately, the results presented in Dr Frazier's testimony do not represent the results for reasonable variations i model parameters which will indicate conservative conclusions as recom mended in my testimony."[177] A major dissatisfaction of Dr. Brune was tha there was no calculation run similar to "Case G" in the San Onofr analysis.[178] For the reasons noted in the margin, however, we find tha

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[171]See, *e.g.*, Brune, fol. R.Tr. 601, p. 8 (Section E); Frazier, IV-79 testimony, p. VII-4 Rothman, fol. R.Tr. 536, pp. 13-14 and Figure 5. As shown in Figure 5, several recordin stations are within the sector that Dr. Brune considered susceptible to focusing. The absence o focusing was strongly supported by applicant's witness Dr. Edwards (IV-79 testimony o Board Question 7).

[172]See Frazier, IV-79 testimony on Board Question 7.

[173]J.I. Exh. R-6 at pp. 2-1 *et seq.* J.I. Exhs. R-6, R-7, R-8, and R-9 were a Final Report an Supplements I, II, and III, produced by TERA Corporation in support of the San Onofr Nuclear Power Plant licensing proceeding. The Final Report sets out in detail the model used and the results of testing it, using the 1966 Parkfield, 1940 Imperial Valley, and 1976 Brawle earthquakes; Supplement I presents computations of earthquake motion at San Onofre Supplement II is a test of a modified version of the model against the 1933 Long Beach an 1971 San Fernando earthquakes; and Supplement III is a comparison of predictions of th model with data taken at IV-79 (see R.Tr. 1357-64). The relevant portions of these exhibit were admitted into evidence. R.Tr. 1133-34.

[174]R.Tr. 1371.

[175]Frazier IV-79 testimony at pp. VII-14 and 15.

[176]See fn. 161, *supra*.

[177]R.Tr. 632.

[178]R.Tr. 629. A review of the parameter variations employed by Dr. Frazier in his calculation indicates that twelve different cases were studied. Frazier IV-79 testimony, pp. VII-9 and VII-10.

'Case G" is neither relevant to the situation before us nor illustrative of 'ocusing.[179]

The staff, while it agrees that focusing is not likely to cause ground notion accelerations of concern at Diablo, essentially rests on the ground hat Dr. Brune's speculations are unpersuasive. However, it also takes the position that the calculated results from the applicant's new model are nsufficiently tested to be useful as quantitative predictions of earthquake notion.[180]

Dr. Frazier's modeling — while not yet perfected — does appear to be a ophisticated tool useful in analytically predicting seismic motions caused it points by a rupturing fault.[181] Several instances are portrayed in the exhibits in which the technique is tested against results of actual earthquake ecords.[182] Given the difficulties inherent in attempting to portray analytically the result of an event which encompasses as many variables as an earthquake, the Frazier model does rather well.[183] We are of the opinion hat this type of effort is of value for the future. However, we agree with the taff that, at its present stage of development, the model's results are not persuasive evidence; we have accordingly given them minimal weight.

We have reviewed carefully all the evidence on the enhancement of eismic motion caused by the phenomenon of focusing. The Licensing Board might have elucidated further its reasons for rejecting Dr. Brune's position, but we cannot criticize its conclusion. The evidence does not permit acceptance of the postulations that seismic focusing might cause enormously enhanced ground motion at Diablo Canyon. As our preceding liscussion indicates, Dr. Brune could give no example of acceleration 'alues even remotely approaching those he hypothesized; it is at best loubtful that Diablo Canyon lies within a "focal zone" on the Hosgri in any event; and, to the extent earthquake modeling techniques have been

[179]See J.I. Exh. R-7, pp. 6-3 and 6-4. Case G in the San Onofre analysis has an epicenter 10km outh of the site along the fault and assumes that the rupture propagates southward away from he site. In that case, according to Dr. Brune (R.Tr. 629-30), "even lower response spectra" would result at the reactor site because of focusing in the opposite direction. But, as the upture starts south of the site and continues in that direction, obviously at no point is it ever djacent to the site and, hence, it has no real relevance to the Diablo Canyon analysis.

[180]Staff IV-79 Proposed Findings, p. 72, fn. 52.

[181]See the series of Dr. Frazier's modeling studies presented in J.I. Exhs. R-6 to R-9 as noted in n. 173, *supra*.

[182]In its initial test against the IV-79 results, the model indicates too much focusing, *i.e.*, a ypothetical recording station along the fault is calculated to have an acceleration 2 to 3 times he acceleration actually recorded at an analogous position (J.I. Exh. R-9, Fig. 4-19). At the ame time, stations further from the fault were calculated to be lower than actually measured ccelerations.

[183]See R.Tr. 1372-77.

perfected, their results cut against Dr. Brune's theory. His hypothesis wa correctly rejected as without evidentiary support and speculative.

2. **High stress drop.** This is a shorthand reference to what happens whe a short fault segment ruptures under very high forces. The resultant energ release is concentrated in a small area and can cause ground motion tha accelerates rapidly. The potential for a stress drop of this kind is accepted i seismology and high values for this factor are known to exist in som areas.[184] Because high stress drops can influence the magnitude of energ release in earthquakes, they are one of the factors that make correlation between earthquake magnitude and ruptured fault length inexact.[185]

Intervenors contended below that this phenomenon might act to produc unusually high ground acceleration at the Diablo Canyon site, relying o testimony of Dr. Brune.[186] The Board below, however, agreed with th applicant and staff[187] that in terms of design significance, the hig acceleration values which Dr. Brune postulated — on the order of 2g[188] — were entirely speculative, stressing that he could give no examples of th measurement or existence of such values occurring anywhere.[189] LBP-79-26 10 NRC at 489.

As with his focusing testimony, we believe the Board did not err i disregarding Dr. Brune's position on stress drop as speculative. Knowledge able witnesses testified that there are no indications of high stress dro regions on the Hosgri Fault, emphasizing (among other things) that were i to rupture, the fault is expected to exhibit a strike slip-dip slip motion rathe than a thrust motion, the latter being the accepted cause of the highes stress drop values.[190] The evidence introduced below provided no reason t believe that unusually high ground acceleration would be experienced a the Diablo Canyon site as a result of this phenomenon.[191]

Nor has anything presented as a result of IV-79 changed this picture. Dr Brune's postulation that much of the energy in that event was released in a small zone is simply not supported by the complete set of motion records o that event.[192] For example, Dr. Brune advanced the theory that as much a

[184]Tr. 5833-37.
[185]See pp. 927-928, *supra;* Tr. 5685-88.
[186]See, *e.g.*, Tr. 7930-31; J.I. Exh. 66, pp. 3-14 to 3-16.
[187]See, *e.g.*, Tr. 5846-47; Stepp, fol. Tr. 8484 at pp. 32-34.
[188]J.I. Exh. 66, p. 3-16.
[189]See *id.* at pp. 3-14 to 3-16. Dr. Brune cited numerous examples of instances of high stres drop, but none resulted in high measured accelerations except the Pacoima Dam event. Eve there, the highest measured acceleration was only 1.2g, and other causes for that unusua measurement have been suggested that are more persuasive than the possibility of high stres drop. See p. 944, *supra.*
[190]Smith, fol. Tr. 5490 at pp. 9-11; Hofmann, fol. Tr. 8522 at pp. 2-3; Newmark, Tr. 8617-18.
[191]See, *e.g.*, Tr. 5469; J.I. Exh. 66 at 3-17.
[192]See R.Tr. 1379-86; App. Exh. R-6; Rothman, fol. R.Tr. 536, Fig. 5.

half the total energy release of 6.9 Ms could be due to stress drop in a relatively small part of the rupturing fault.[193] Intervenors point to the high acceleration (0.81g) recorded at one station (Bond's Corner) as evidence of this.[194] But if, as Dr. Brune postulated, the energy release at IV-79 was at 6-10km depth (R.Tr. 613), there are several other recording stations at roughly the same hypocentral distance to the high stress drop zone as Bond's Corner.[195] None, however, showed acceleration values approaching that of Bond's Corner — a fact that undercuts the idea that a great portion of the energy was released in a small zone of high stress drop.[196] Moreover, rather than evidencing unusually high peak acceleration, the Bond's Corner measurement of 0.81g is commensurate with measured near-field values reported for earthquakes above 5 Ms in other recent studies.[197] In short, Dr. Brune's theory is inconsistent with the evidence of record and, accordingly, we find the position of the intervenors and the Governor on high stress drop not well taken.

D. Adequacy of the Newmark Spectrum in Light of Actual Ground Motion Records

The joint intervenors and the Governor press the further argument that the Newmark free-field ground motion spectrum is not an adequately conservative representation of a 7.5Ms event on the Hosgri Fault. They point out that the Newmark Spectrum has been exceeded by response spectra developed from ground motion records obtained during two less severe earthquakes, the Pacoima Dam record from the 1940 San Fernando event (6.5Ms) and the Bond's Corner record from the 1979 Imperial Valley earthquake (6.9Ms).[198] The point is not disputed but is illustrated in Dr. Newmark's own discussion of the Pacoima Dam record,[199] as well as in the testimony of Dr. Pao-Tsin Kuo for the staff[200] and Dr. George Young for the Governor[201] who addressed the Bond's Corner record. In each instance

[193]R.Tr. 853-54; see also J.I. Exh. R-12.
[194]J.I. IV-79 Proposed Findings, 17-22.
[195]Fn. 148, *supra*.
[196]We note in passing that Dr. Brune himself acknowledged that his "small zone" of energy release could have been some 10km in length. R.Tr. 854.
[197]See J.I. Exh. 47 (Hanks-Johnson paper); App. Exhs. 61 and 62.
[198]J.I. IV-79 Proposed Findings at pp. 24-27; Governor's IV-79 Proposed Findings at pp. 22-27. The design basis event on the Hosgri is a 7.5Ms (Luco, fol. R.Tr. 1138, p. 1-3); the USGS value for the 1979 Imperial Valley is 6.9Ms. R.Tr. 758. Intervenors argue the latter could be in the range of 6.5 to 6.9Ms, IV-79 Proposed Findings at 15; see fn. 83, *supra*. In light of our finding that, in the near field, high frequency earthquake motion appears to saturate with magnitude, the difference is not significant.
[199]Newmark, fol. Tr 8552, Reference A, p. C-4, Figs. 1a and 1b.
[200]Kuo, fol. R.Tr. 538, Attachment 1.
[201]Young, fol. R.Tr. 608, Figure 7.

the Newmark Spectrum, though bounding the others over most of the range of frequencies, is in fact exceeded for certain narrow ranges.[202]

We have already outlined how the Pacoima Dam record, which includes the highest recorded ground acceleration, was used by Dr. Newmark in creating his basic response spectrum for Diablo Canyon (pp. 939-940 *supra*).[203] Dr. Newmark testified that the narrow ranges in which the Pacoima Dam record exceeded the boundary of the Hosgri design spectrum are not significant from the standpoint of any structural behavior,[204] a position in which the staff concurred.[205]

As we previously observed, the Pacoima Dam record is not considered representative of a typical earthquake in its size range; the geological features of that site resulted in a record that magnified the actual ground motion. (See p. 944, *supra*.) Dr. Newmark elected to use the Pacoima Dam record for comparison purposes essentially because it was one of the few available that approached the high acceleration value recommended by the USGS for a 7.5M event.

No similar geological attributes of the site explain the magnitude of the record made at Bond's Corner during the 1979 Imperial Valley event.[206] The evidence presented by Dr. Blume for the applicant at the IV-79 hearing does, however, underscore the large disparity between that one record and the many others obtained contemporaneously under similar circumstances. Blume, Figure II-1 is a plot of the response spectrum developed for the Bond's Corner record and spectra developed for 23 other near-field (within 11km of the fault) records obtained during IV-79. Over most of the high frequency range, the Bond's Corner record of peak ground acceleration exceeds those other records roughly by a factor of 2. If the Bond's Corner record does not reflect some sort of geological or recording station anomaly,[207] it certainly represents a record at the boundary of empirically

[202]Joint intervenors additionally argue that this is also a violation of the Commission's seismic rules (set forth in Appendix A of 10 CFR Part 100 that the plant be designed to accommodate maximum expected ground motion. J.I. Proposed Findings at p. 25. For the reasons explained at p. 941, *supra*, we reject this argument as resting on a misconstruction of those regulations.

[203]Tr. 8618-20.

[204]Newmark, fol. R.Tr. 534 at pp. 10, 13-14; Tr. 8590, 8593.

[205]Kuo, fol. R.Tr. 538 at pp. 5-7.

[206]Staff witness Dr. Rothman observed in his prepared testimony (at p. 6) that there might be something unusual about the Bond's Corner site or station as the records obtained there at IV-79 were high relative to surrounding stations, and the same behavior was noted during a subsequent earthquake. Under cross-examination, Dr. Rothman admitted that he had done no studies of the Bond's Corner geological structure (*i.e.*, such as had been done at Pacoima Dam) and had no reason to discount the Bond's Corner recordings other than their anomalous magnitude. R.Tr. 563-64.

[207]See R.Tr. 185.

determined statistical expectations for accelerations at that distance from the 1979 rupture of the Imperial Fault.

Dr. Blume, in his testimony below[208] and at the reopened hearing,[209] described his method of obtaining a design spectrum by averaging over a number of individual spectra, obtained for the type of site being considered, and normalized to some common high frequency anchor point.[210] Spectra obtained in this way, in Dr. Blume's view, represent the proper way of incorporating the available data, whereas the use of a single high record such as Bond's Corner would statistically misrepresent the data.[211]

For the reasons we have discussed, neither the Pacoima Dam nor the Bond's Corner records typify the motion in the near field of 6.5Ms events but, rather, represent distorted responses. Even so, the Newmark Hosgri Design Spectrum is approximately equivalent to the raw spectra developed from those records — two of the strongest near-field ground motion records ever measured. The Newmark Spectrum is thus fairly comparable to them both. Taken with our earlier finding that the magnitude of near-field ground motion is not strongly dependent on earthquake size, the Bond's Corner and Pacoima Dam records are not cause to reject the Hosgri Spectrum as an insufficiently conservative representation of SSE ground motion at Diablo Canyon.

E. Variation of Motion on Rock versus Soil

Diablo Canyon has been described as a rock[212] or a soft rock site;[213] on the other hand, the Imperial Valley is a deep, soft alluvium (*i.e.*, soil) site.[214] This difference gave rise to our question about the relevance of data obtained at IV-79 to the prediction and description of seismic motion at

[208]Fol. Tr. 6100, pp. 13-15; Tr. 6680-84.
[209]Blume, IV-79 testimony, p. II-1 & 2.
[210]*Ibid;* see also fn. 40, *supra.*
[211]R.Tr. 185-189.
[212]R.Tr. 192.
[213]Applicant's IV-79 testimony, Exh. 1 (TERA Corporation Report), p. 2-4.
[214]R.Tr. 192.

Diablo Canyon.[215] Joint intervenors and the Governor contend that in the near field of earthquakes of comparable magnitude, high frequency ground motion would be greater at a rock site than at a deep soil site.[216] They rest their assertion largely on the testimony and publications of Dr. H. Bolton Seed,[217] one of applicant's witnesses, as well as on statements of the staff's witness Dr. Newmark[218] and Board witnesses Drs. Luco and Trifunac.[219] Intervenors and the Governor argue accordingly that the Bond's Corner reading of 0.81g in IV-79 would have been even higher had it been obtained on a rock site. They put this forward as an additional reason why, in their judgment, the Newmark Spectrum and anchor point acceleration (0.75g) are not conservative.

Two compilations of earthquake data were put into evidence (USGS 795[220] and the TERA Corporation Report[221]) and a third (NUREG/CR-1175[222]) was referred to extensively by the Governor's witness Dr. Young. We note that the authors of all these documents conclude — contrary to intervenors' position — that for a comparable seismic event, peak ground

[215]Appeal Board Question 3 (ALAB-598, *supra*, 11 NRC at 889):

> "We are told that IV-79 data are not relevant to the Diablo Canyon seismic analysis because that plant is a 'rock' site, whereas the Imperial Valley data were obtained on soil sites. (Rothman - Kuo Affidavit at 3; Blume Affidavit, Para. 8.) What is the significance of this difference in view of the conclusion of the authors of USGS Circular 795 (based on an analysis of data provided in that document) that, for comparable earthquake magnitude and distance, there are no significant differences between peak horizontal accelerations measured on soil or rock? (USGS Circular 795 at pages 1, 17, and 26.) This question should be considered in light of statements by applicant's witness Blume to the effect that acceleration, rather than velocity or displacement, is the critical parameter in the design of Diablo Canyon (Blume Affidavit, Para. 9; Testimony fol. Tr. 6099, at 33)."

[216]See, *e.g.*, Gov. IV-79 Proposed Findings at pp. 12-13; J.I. IV-79 Proposed Findings at p. 23; R.Tr. 863-64.

[217]Cross-examination of Dr. Seed, R.Tr. 212-220, 232-253; J.I. Exhs. R-3 and R-4.

[218]Newmark, fol. R.Tr. 534 at p. 17. Dr. Newmark indicates here that the rock-soil acceleration ratios would be sensitive to the intensity of motion. At higher intensities (0.6g or above), the acceleration at a rock site would exceed that on a soil site. This is in agreement with Dr. Seed's findings illustrated in Fig. 9 of J.I. Exh. R-4 at p. 1337. Dr. Seed's near-field conclusions (within about 20km or for accelerations greater than about 0.3g) are extrapolations based on data taken at greater distances and on the author's judgment (*id.* at p. 1334).

[219]Trifunac, fol. R.Tr. 1138 at pp. III-1 & 2; Luco, fol. R.Tr. 1138 at pp. 3-2 to 3-6. (Note: Luco testimony follows Figure 18 of App. A to App. VIA of Trifunac testimony.)

[220]J.I. Exh. R-1.

[221]Exh. 1 to applicant's prefiled directed testimony at the reopened hearing on Appeal Board Question 1.

[222]NUREG/CR-1175 - Statistical Analyses of Earthquake Ground Motion Parameters; Shannon & Wilson, Inc., and Agbabian Associates, December 1979.

accelerations measured on rock would be generally equivalent to or less than those measured on soil,[223] although they qualified their findings as resting on a relatively small data base for the near field.[224] Applicant's witness Dr. Blume reached the same conclusions using extensive near-field data compiled during nuclear weapons testing.[225]

In answer to our questions, Dr. Seed explained that no theoretical basis underlay his conclusion that accelerations on rock would exceed those on soil, but that it was a result of the way in which he separated the existing experimental data into rock and soil sites.[226] The TERA Corporation Report, of which applicant's witness Dr. Stewart Smith was a primary author, also segregated ground motion data according to site geology, but into categories of recent alluvium (soil), pleistocene deposits (soil), soft rock, and hard rock.[227] Although no specific results are shown, that report concludes that for the "soft rock" category (into which it places the Diablo site), peak accelerations would be lower than those recorded on either hard rock or soil sites.[228]

In light of the apparent discrepancy, we examined with particular care the TERA Corporation Report (1980) data base and that given in Dr. Seed's two papers[229] (both published in 1976). Our review revealed that most of the near-field data used by TERA was not available to Dr. Seed.[230] The evidence before us thus favors the concept that near-field soft rock and soil accelerations should be expected to be about equal. We accept the conclusions drawn from the TERA Report as resting upon a more recent and more comprehensive data compilation. We do not, however, believe that a finding in this regard is crucial to the resolution of the issues raised before us.

Intervenors and *amicus* have urged that the Bond's Corner (deep soil) record and response spectrum be adopted as a basis for Diablo Canyon, a suggestion we have already rejected on statistical grounds (p. 953, *supra*). We have not, however, used the IV-79 data directly to substantiate any of the Diablo Canyon analyses. Rather, the new data corroborate the

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223USGS 795, p. 1. In discussing near-field motion, however, the authors of USGS 795 opine that "[a]t sites other than rock sites accelerations might be less because of the limited strength of near-surface materials." *Id.* at 26. See also TERA Corporation Report at p. 1-7; NUREG/CR-1175 at p. 5-9.
224See, *e.g.*, J.I. Exh. R-1 at p. 25.
225Tr. 6650-51 (referencing Fig. 1, D-LL 11B).
226R.Tr. 250-51.
227See TERA Report (fn. 173, *supra*), Table 2-1.
228*Id.* at p. 5-18.
229An investigation of the data base used by the TERA Corporation to reach their conclusion reveals that, for distances within 10km of the fault rupture surface, their recordings included records from 36 soil sites, 4 soft rock sites and 3 hard rock sites (*id.*, Table B-I).
230J.I. Exhs. R-3 and R-4.

concepts of distance and magnitude saturation (pp. 931-933, 934-935 *supra*) and illustrate that focusing and high stress drop effects were not apparent at IV-79. But this does not require the use of absolute acceleration values from Imperial Valley at Diablo Canyon.[231]

We agree with the witnesses for the applicant and the staff who cautioned that it would be inappropriate to use the IV-79 data to predict the absolute value of ground motion at Diablo Canyon because of the geological differences between the two sites.[232] We note in passing, however, that Dr. Luco's suggestion — that IV-79 data should be scaled up to account for the rock-soil difference were it to be used to support the Diablo Canyon reanalysis — was taken up by Dr. Seed, who thereby obtained a peak acceleration appropriate for Diablo Canyon design conditions equal to 0.71g.[233] While we do not believe it necessary to accept Dr. Seed's computations for our decision, they do suggest a basic consistency between various methods used to approach the ground motion prediction.[234]

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[231]We did note that the highest peak acceleration at IV-79, 0.81g, was in general agreement with the peak values tabulated in the Hanks-Johnson paper (p. 966, *supra*).

[232]Blume, IV-79 testimony at pp. III-1 to III-3; Rothman, fol. R.Tr. 536 at pp. 3-4, 11-12.

[233]R.Tr. 1407-15. Using IV-79 data regressed (App. Exh. R-18), Dr. Seed determined that mean peak acceleration 5.8km from the fault was 0.39g, and the mean plus one standard deviation ($M + \sigma$) was 0.52g. He increased this for the rock-soil difference by the factor 1.2, obtaining a ($M + \sigma$) for a rock site (Diablo Canyon) of 0.62g. He then multiplied this value by the factor 1.15 to scale it from 6.9Ms (IV-79) to 7.5Ms (Hosgri SSE), with a resulting peak ($M + \sigma$) acceleration value of 0.71g. (These operations are outlined in App. Exh. R-19.) Dr. Seed also calculated a ($M + \sigma$) design spectrum using two scaling methods and the IV-79 mean spectrum calculated by Dr. Blume (IV-79, Fig. II-2). Both spectra were encompassed by the Newmark Design Spectrum at all frequencies of interest for Diablo Canyon structures. App. Exh. R-20.

[234]Dr. Luco also devised a response spectrum using scaled IV-79 data. Fol. R.Tr. 1138 at Fig. 2-1, p. 2-5. However, he first normalized Dr. Blume's raw IV-79 mean spectrum to a 0.75g anchor point (*i.e.*, scaling the Blume spectrum up by a factor of 0.75/.36 = about 2), and then applied a factor of 1.50 to obtain a mean plus one standard deviation spectrum. Applicant's witness Dr. Seed pointed out, however, that a 0.75g anchor point value is in fact already a mean plus one standard deviation, and thus Dr. Luco was using a standard deviation factor twice (R.Tr. 197, 208-209).

Similarly, Governor Brown's witness Dr. Young introduced a response spectrum utilizing IV-79 data (Gov. Exh. R-14), normalized to a mean peak acceleration of 0.75g and scaled upward to a mean plus one standard deviation at the anchor point of 1.08g. Applicant's witness Dr. Blume correctly pointed out, however, that if the spectrum were properly anchored at a ($M + \sigma$) of 0.75g, all points of design significance would fall below the Newmark Spectrum (R.Tr. 1353.

We believe there is ample evidence to show that the expected mean peak acceleration at Diablo Canyon for a 7.5M event on the Hosgri is about 0.5g. Hence, the objections noted above to the Luco and Young spectra are well taken. (TERA Corporation Report, Table 4.1 at pp. 4-4; R.Tr. 245-47; Blume, IV-79 testimony, Fig. I-9; Tr. 6067). As discussed earlier (p. 943

In closing this point, we stress that the real value of the Imperial Valley data to these deliberations lies in the large number of strong motion recordings obtained there. These allow a coherent view of earthquake motion near a fault and some check on assumptions underlying the seismic reevaluation. That usefulness does not, however, depend upon the precise motion values measured, nor does it require resolution of the rock-soil question beyond that indicated in the foregoing discussion.

F. Vertical versus Horizontal Accelerations

One issue that has arisen solely as a result of the data obtained at IV-79 is the magnitudes of peak vertical accelerations in the near field relative to those of peak horizontal acceleration. To outline this matter, we quote below our Question 4 of ALAB-598 (11 NRC at 889-90). The question also points to an apparent inconsistency between the Diablo Canyon Hosgri Reanalysis and normal NRC staff requirements as set forth in Regulatory Guide 1.60:

> 4. The magnitudes of vertical and horizontal acceleration values measured at IV-79 are generally comparable. (Mean values calculated at a distance of 5.8km from the fault are virtually identical.)[36] The response spectra developed for vertical motion within 11km of the Imperial Fault during IV-79 appear to show generally equivalent values of vertical and horizontal response for periods less than about 0.2 seconds (*i.e.,* frequencies in excess of 5 cps).[37] Finally, in some instances the higher frequency portions of the IV-79 response spectra for vertical motion exceed comparable portions of the Diablo Canyon Design Response Spectrum.[38]
>
> Observations made of the IV-79 data and response spectra appear to be consistent with the criteria set forth in NRC Regulatory Guide 1.60.

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supra), anchoring the design response spectrum at an acceleration value lower than that measured by an instrument in the near field, *i.e.,* at the "effective acceleration," is entirely justifiable.

*[36] Blume Affidavit, Table 1, Figures 1 and 2.

* Footnote included in our Question 4.

*[37] Rothman-Kuo Affidavit, Figures.

*[38] *Ibid.*

These require that vertical accelerations in the higher frequency range be equal to horizontal accelerations. As the guide states:

> It should be noted that the vertical Design Response Spectra are 2/3 those of the horizontal Design Response Spectra for Frequencies less than 0.25; for frequencies higher than 3.5 they are the same, while the ratio varies between 2/3 and 1 for frequencies between 0.25 and 3.5.[39]

The references to vertical motion made in the Diablo Canyon record, however, indicate that a 2/3 ratio between vertical and horizontal motion was apparently utilized at all frequencies.[40] The parties should address this apparent inconsistency and explain it, if possible. Should there be substantive and relevant analyses suggesting that vertical motion records do not reflect the true vertical motion, these should be provided.[41]

We note one other factor before addressing the responses to our question. In addition to the IV-79 near-field data's general trend toward equal vertical and horizontal accelerations, the peak vertical acceleration during that event, measured at El Centro Station number 6, was 1.74g (uncorrected) and 1.52g (corrected by the USGS). This is the highest ground acceleration ever measured anywhere.[235]

We turn first to the apparent regulatory inconsistency. Dr. Kuo's testimony for the staff sets this matter straight.[236] Diablo Canyon was indeed analyzed on the basis of a vertical motion spectrum two-thirds the magnitude of the basic Newmark horizontal spectrum (*i.e.,* with an anchor point acceleration of 0.5g rather than 0.75g). Regulatory Guide 1.60 was published in its present form in 1973. In July 1976, however, the staff adopted a "branch position" allowing applicants in the western part of the

*39 We note that elsewhere in the Regulatory Guide frequencies are presented with accompanying units of cycles per second (cps), and assume that these units are inadvertently omitted in the portion we have quoted.

*40 SER Supplement 7, at 3-18; Knight Testimony, at 13, fol. Tr. 8697, Ghio Testimony, at 1, fol. Tr. 6993. Blume Testimony, at 41, fol. Tr. 6099.

*41 See, for example, Newmark Testimony, fol. Tr. 8552, Reference B at 4, 5; Tr. 9349.

235 USGS Open-File Report 79-1654 (included with Board Notification of December 17, 1979); Frazier, IV-79 testimony at pp. IV-1 and IV-6; Int. Exh. R-9, Table 2-2 at p. 2-6. There was little damage from IV-79 (Frazier at p. IV-8).

236 Kuo, IV-79 testimony fol. R.Tr. 538 at pp. 8-9.

United States the option of using such a two-thirds ratio for peak acceleration in lieu of the Regulatory Guide 1.60 criteria.[237] It is to be recalled that the Regulatory Guides do not lay down mandatory directives but delineate problem-solving techniques the staff deems acceptable from past experience. See p. 937, *supra*. The changes were not made for this case. Rather, they reflect Dr. Newmark's conclusions as the staff's general consultant in this area from a study of extensive compilations of ground motion records.[238]

We need only add that the earthquake data compilations called to our attention in this case (to the extent that they deal with vertical motion) confirm that within the distance range of their measurements, the two-thirds vertical to horizontal ratio provides a reasonably conservative measure of vertical motion.[239]

The most extensive response to Question 4 was the testimony of applicant's witness Dr. Frazier.[240] Briefly summarized, he explained the general vertical-horizontal motion behavior in terms of the Imperial Valley's geologic structure. The Valley is a deep alluvial basin that tends to amplify — as an echo chamber — all vibratory motion. Compression waves (P-waves) are dispersed (attenuated) relatively slightly within the soils and they emerge steeply at the earth's surface, giving rise to high peaks of vertical acceleration. The primary source of horizontal motion, however, is shear waves (S-waves). These travel more slowly than P-waves[241] and are more heavily attenuated in the deep soil of the Imperial Valley. Dr. Frazier's testimony provided a graphic description of the early arriving, sharp (high frequency) vertical peaks that contrast with the broader, later peaks of acceleration in the horizontal plane.[242] These motion records lend corroboration to Dr. Frazier's attribution of the vertical peaks to P-waves.

According to Dr. Frazier, the Imperial Valley geology that gives rise to the enhanced P-wave vertical accelerations is not characteristic of Califor-

[237]The Hosgri Reanalysis took place after July 1976. SER Supplement 5, containing the Newmark Design Spectrum, was published in September 1976.
[238]Kuo, IV-79 testimony fol. R.Tr. 538 at pp. 8-9.
[239]For instance, USGS 795 (J.I. Exh. R-1), without comment, displays curves for vertical acceleration that are lower by a factor of about 2 than comparable horizontal motion curves. Compare Figures 1, 2, and 3, with 14, 15, and 16. In NUREG/CR-1175, cited by the Governor's witness Dr. Young, Figure 3-31 shows for all magnitudes and for distances to about 10km, peak vertical acceleration less than two-thirds of horizontal. This report also concludes (at p. vii) that the Regulatory Guide 1.60 position on vertical acceleration is "very conservative." For high frequencies in the near field, Dr. Young disagrees with this statement and believes more study is needed. Young, fol. R.Tr. 608 at pp. 28-29.
[240]Dr. Frazier's IV-79 testimony, pp. IV 1-10.
[241]It is a well-accepted aspect of earthquake phenomenology that the P (compression) waves travel more rapidly than the shear waves. This fact is used to determine the location and magnitude of earthquakes. See, Bolt, *Earthquakes*, 30, 31, 96, and 105. See also Tr. 5621-23.
[242]Frazier IV-79 Testimony, Figures IV-4, IV-5, and IV-6.

nia earthquake structures. In the typical pattern, peak vertical acceleration would be caused by vertically polarized shear waves generally arriving in phase with the horizontal shear waves and having a vertical-to-horizontal peak acceleration ratio of less than one.[243] Dr. Frazier testified that the typical situation would be characteristic of the Diablo Canyon site.[244] Dr. Frazier also presented calculations he performed with a simplified version of his earthquake model (see p. 948, *supra*). The results of those calculations support his proposition that the Imperial Valley earth structure enhances vertical accelerations with respect to horizontal,[245] something that would not be the case at Diablo Canyon.

Dr. Frazier also offered a geologic explanation for the very large vertical accelerations recorded during IV-79 at El Centro Array, Station Number 6.[246] He pointed out that this instrument is on the end of a wedge-shaped section of land formed by the intersection of the Imperial and Brawley Faults and has recorded acceleration peaks during other events that exceeded those measured at neighboring stations.[247] He attributed the peculiar behavior of Station Number 6 to the fact that the wedge of land on which it is sited has lower wave velocity than the surrounding area. Extensive geologic investigation indicates that similar conditions are not present at Diablo Canyon.[248]

Both Dr. Newmark and Dr. Blume observed that the way in which seismographic instruments are mounted, whether on the ground or in a structure, can give rise to spuriously high indications of vertical accelerations, and our attention was called to papers addressing the overregistration of vertical ground motion from such causes.[249] Dr. Newmark noted, however, that he had made no study of the particular instruments and their mountings at Imperial Valley.[250] Dr. Brune for intervenors[251] and Dr. Young for the Governor[252] testified that they had no personal knowledge that seismographic records of vertical motion would be suspect. Dr. Young

[243] *Id.* at p. IV-2; see also R.Tr. 282-86.
[244] R.Tr. 284-86.
[245] Frazier IV-79 Testimony, p. IV-15, Table IV-2.
[246] *Id.* at pp. IV-6 to IV-8.
[247] At IV-79, for the three stations, Number 7 (1km southwest of Number 6, and west of the Imperial Fault); Number 6 (east of Imperial but west of Brawley), and Number 5 (3km northeast of Number 6 and east of both faults), the peak, uncorrected vertical acceleration recordings were: Number 7, 0.65g; Number 6, 1.74g; and Number 5, 0.71g. *Id.* at pp. IV-6 to IV-7; p. IV-16, Figure IV-1.
[248] *Id.* at pp. IV-6 to IV-8.
[249] Newmark fol. R.Tr. 534 at pp. 8-9; fol. Tr. 8552, Reference B, pp. 4-5; Tr. 9349; Blume IV-79 testimony at p. IV-5.
[250] R.Tr. 593.
[251] Brune, fol. R.Tr. 601 at pp. 8-9.
[252] Young, fol. R.Tr. 608 at p. 29.

further added that he was aware of no reservations in this regard being expressed by the record processors at USGS or the California Institute of Technology.[253]

Dr. Luco had reservations about the validity of the Frazier calculations, suggesting that the results for the ratio of vertical to horizontal peak acceleration appear to be very sensitive to the assumed earth structure.[254] Dr. Frazier appeared to share this concern at least to some degree. He observed that the variation of ratios over the range of 3 to 10 reflects some uncertainties in the earth structure models used for the Imperial Valley. He went on to testify, however, that the same level of uncertainty was not present in the Diablo Canyon calculations, the earth structure there being better known. Thus he asserted that calculational results at Diablo would be less sensitive to such variations.[255]

Dr. Luco[256] and Dr. Trifunac[257] both referred to Dr. Trifunac's earlier analysis in which a coefficient, developed for an empirical equation to describe earthquake motion, portrays higher vertical than horizontal accelerations for motion frequencies exceeding 10hz. Both witnesses believed that further investigation was needed of high peak vertical acceleration as a near-field phenomenon of concern to Diablo Canyon.

In response to our question on the near-field nature of the high ratios of vertical to horizontal peak acceleration, Dr. Frazier testified that, in his view, the ratio would increase in the near field because it is controlled by P-waves that decrease in importance with distance from the fault.[258]

We are satisfied from the record that the high peak accelerations at Imperial Valley are due to P-waves of high frequency, arriving well in advance of the peaks in horizontal motion.[259] We also believe that Dr. Frazier has made a reasonable case for the proposition that this behavior is due in part to earth structures at the Imperial Valley that have no

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[253]*Ibid.* In this regard, we have reviewed the USGS data for IV-79 in *uncorrected* form (USGS Open-File Report 79-1654, see fn. 235, *supra*) and in *corrected* form (Int. Exh. R-9, Table 2-2). For the data within 11km of the Imperial Fault, with one exception (Station Number 6, 140°), the corrections in horizontal data were reductions in magnitude of about 10% or less. The corrections to vertical peak accelerations were reductions ranging from 14 to 60 percent, with an average correction of about 35%. We believe this comparison at least implies some USGS sensitivity to vertical instrumental over-registration.

[254]Luco, fol. R.Tr. 1138 at pp. 4-6 and 4-7.

[255]R.Tr. 439-441.

[256]Luco, fol. R.Tr. 1138 at p. 4-5.

[257]Trifunac, fol. R.Tr. 1138 at p. IV.1 (citing Appendices IVA, IIA and IIB of his testimony). We note, however, that Appendix IIIA of the Trifunac testimony is a paper by Dr. Trifunac and A.G. Brady that correlates peak acceleration data with earthquake intensity. The data for all intensities show peak horizontal acceleration greater than vertical by about a factor of 2 (Table 5).

[258]R.Tr. 289-90.

[259]See pp. 959-960, *supra*.

counterpart at Diablo Canyon. We are aware, however, that the calculations supporting this conclusion are very sensitive to the assumed earth structure at the Imperial Valley.[260] Nevertheless, we conclude that even if, due to its proximity to the fault, Diablo Canyon were to experience high vertical P-wave accelerations comparable to the horizontal peaks, they would not cause stresses exceeding design values. This is because any such vertical acceleration would be high frequency peaks having little energy associated with them[261] which, moreover, would occur well out of phase with the peak horizontal accelerations. (On the latter point, the Hosgri analysis, in conformance with NRC practice, added the acceleration in both horizontal directions and the vertical, as if the peaks occur simultaneously.[262] This assumption would clearly be invalid for the early P-wave, vertical peaks.)

A final observation regarding the significance of vertical acceleration at Diablo Canyon is warranted. Before the Licensing Board, applicant's witness Dr. Hanusiak testified on cross-examination that the contribution to the stress in the containment shell due to vertical seismic excitation was about 1.9% of the total.[263] Put another way, for the point under discussion, the total stress was 60.31 KSI, the total seismic stress 40.70 KSI, and the stress due to vertical seismic loading, 0.79 KSI. This testimony was not challenged. It indicates that, even were the vertical accelerations equal to those in the horizontal direction (*i.e.*, increased by 50 percent over the design value), the resulting increase in the total calculated stress would, for the case in point, be only about 1 percent.[264]

For the reasons stated, we conclude that the vertical motion phenomena we have described will have no significant consequences for the Diablo facility.

V

THE TAU EFFECT

A. General Discussion

The "tau effect" is a phrase which was employed during the course of these hearings to symbolize a phenomenon by which the higher frequencies of earthquake motion are reduced in large structures. ("Tau" is simply the

[260]See pp. 959-961, *supra*.
[261]See, *e.g.*, p. 942, *supra*. Also Blume IV-79 testimony, p. IV-6; Frazier IV-79 testimony, p. IV-8.
[262]Frazier IV-79 testimony, p. IV-9; Blume IV-79 testimony, p. IV-5.
[263]Tr. 7045, 7048-51; J.I. Exh. 62 and 63.
[264]Tr. 7050.

Greek letter "t.") The term was defined by Dr. Nathan Newmark to represent the time needed for a seismic wave propagating horizontally to cross the effective width of a building.[265] The applicant and the staff used the tau effect in the Hosgri reanalysis. Doing so reduced significantly the higher frequency portions of the response spectra.[266] It is the validity of that use and that reduction which drives the debate on this point.

Joint intervenors objected to the adoption of these response spectra reductions. Relying essentially on testimony of Drs. Luco and Trifunac, they argued to the Licensing Board that the tau effect had in general been insufficiently proven and its application to the Diablo Canyon site in particular inadequately demonstrated. The Board, however, found the tau effect reductions in question to have been both justified and conservative.[267] Joint intervenors renew their arguments on appeal. In light of their assertion that their position has been fortified by information derived from the 1979 Imperial Valley earthquake, we posed two questions related to the tau effect for the reopened hearings on IV-79.[268]

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[265]Newmark, fol. Tr. 8552, Ref. A, p. C-14.

[266]FSAR Amendment 50, Appendix D-LL 10 at page 10.4. The fractional reductions in the Newmark and Blume response spectra due to the tau effect are presented here for different structures and for several frequencies. For example, a tau value of 0.04 was assigned to the Diablo Canyon containment building. This had the effect of reducing the anchor point acceleration of the Newmark Spectrum by 20%, *i.e.*, from 0.75g to 0.6g. The reduction is 22% at 5 hertz, generally decreases as frequencies are reduced and disappears at about 1.7 hertz.

[267]LBP-79-26, 10 NRC at 494-96.

[268]ALAB-598, *supra*, 11 NRC at 890-91 (footnotes omitted):

> 5. Peak horizontal acceleration values measured at the base of the Imperial Valley Services Building during IV-79 exceed those measured in the free field 103 meters away from the building. The motion records are described as showing similar amplitudes but greater low frequency motion in the building than in the free field. No response spectra for the two recording locations have been provided. The acceleration data, however, may be taken to indicate that no reduction in building motion due to the tau effect was realized in this instance.
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> Based on these observations, intervenors question the validity of the tau concept as well as its use to reduce the higher frequency portions of the Diablo Canyon Design Spectrum. The staff and the applicant answer that, because the Imperial County Services Building was supported on piles in a deep soil structure, these observations are irrelevant to the use of a tau effect in the seismic reanalysis of Diablo Canyon, which is built on a rock site. Staff witness Newmark, however, used recorded earthquake motions at the Hollywood Storage Building to demonstrate the use of a tau effect analysis. The Hollywood Storage Building itself is built on piles in soil. Thus, the "built-on-piles" rationale appears insufficient to explain why no tau effect was evident at the Imperial Valley Services Building.
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> One feature distinguishing the two buildings that no party commented upon is that the Hollywood Storage Building has a basement and the Services Building does not. Intervenors' witness, Dr. Luco, used this fact to explain in part why he believes the Hollywood Building should have a large tau value. Rojahn and Ragsdale's discussion

Before turning to those matters, we note preliminarily that the tau effect apparently encompasses (if not combines) several different and technically complex physical phenomena. While the Governor joins intervenors in taking the position that its utilization is not justified,[269] in addition the witnesses for the staff and applicant themselves differ about the role of various individual physical phenomena in the net effect. Finally, perhaps because of the complexities involved, the staff's and the applicant's presentations to us were less than pellucid. As a result, certain bases upon which a quantification of this effect is established tended to be obscured and our review of the subject became a difficult undertaking.

In applicant's direct testimony, the reduction in seismic motion due to the tau effect was analogized to the responses of vessels in a choppy sea. Large ships "iron out" many of the ocean's waves and wave motion does not toss them about the way it does smaller craft.[270] Thus, the excellent performance of the foundations of large buildings in earthquakes was cited as evidence of the tau effect.[271] Physically, this effect results as the large rigid foundation[272] averages out the motion of the higher frequency (short wave length) ground oscillations, which may be subjecting various portions

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implies that to some extent ground level instrumental responses within the Imperial Valley Services Building may have been influenced by the response (and failure) of the building itself.

In any event, given the apparent similarities between the structural foundations of the two buildings, the explanations provided thus far for a seeming lack of a tau effect at the Imperial Valley Services Building are inadequate. The parties should provide additional information on this point and relate their analyses to both geologic and structural conditions prevailing at the Diablo Canyon site.

6. Throughout the Licensing Board hearings, parties stressed the role of soil-structure interactions as a mechanism that would reduce the magnitude of structure motion relative to ground motion (*e.g.*, Tr. 8878; 8947-53). Staff and applicant's arguments (in response to intervenors' suggestion of the apparent lack of tau effect during IV-79) point to soil structure interactions as the reason for building motion exceeding that of the ground (Blume Affidavit, Paragraph 10; Rothman - Kuo Affidavit, page 7). (a) Describe and explain the circumstances in which soil-structure interactions produce enhanced or reduced structural response. (b) Discuss the relevance and applicability for such interactions to the seismic response assumed for Diablo Canyon.

[269] Gov. IV-79 Proposed Findings at pp. 35-45.
[270] Blume, fol. Tr. 6100 at p. 42.
[271] *Ibid.*
[272] The validity of the tau effect includes a necessary assumption that the building foundation be rigid, and hence that all portions of it would move generally together (in phase) rather than would be the case for a flexible foundation. Under the ship analogy, a large ship with a relatively rigid hull structure would iron out wave motion. On the other hand, in a collection of small boats tied loosely together (a flexible array), although they might cover the same area, each would feel more of the wave motion than the large ship.

of the structure simultaneously to different and competing (*e.g.*, up and down, left and right) phases of the motion.[273]

Applicant presented a generalized analytical treatment of the tau effect, using harmonic seismic waves of different frequencies incident at arbitrary vertical directions upon a slab foundation.[274] In this paper, the averaging of idealized seismic motion over the rigid foundation is represented mathematically by a frequency-dependent motion filter that, for higher frequencies, reduces input motion to the building relative to ground.[275] The filter in this analysis represents a structure of given size subjected to horizontally propagating seismic waves. It is used analytically to modify a ground motion record in order to generate response spectra characterizing the reactions of such a structure. For increasing values of the tau parameter, the resulting spectra display progressively decreased spectral acceleration at the higher frequencies[276] (tau in these calculations being defined as the length of the building in the direction of wave propagation divided by the wave velocity).

The calculations in the paper are illustrative of the analytical technique but have no direct bearing on the Diablo Canyon plant and were performed for purely horizontally moving waves, although the technique would allow consideration of vertical incidence. (See, p. 968, *infra.*)

The most definitive exposition of the tau effect phenomenon was provided by Dr. Newmark and appears in Appendix C of SER Supplement 5.[277] In that document, Dr. Newmark presents response spectra produced from motion records taken in the Hollywood Storage Building (HSB) for two earthquakes (San Fernando 1971 and Kern County 1952), along with the spectra from "free field" ground motion records for those events obtained in a parking lot 112 feet from the building.[278] For the San Fernando event (about 35km north of the building site), starting at a frequency of 1-2 hertz, response spectra for motion records in both the

[273] A further discussion of the physical basis of the tau effect appears at pp. 967-969, *infra.*

[274] D-LL 39A, D. Ray and D.P. Jhaveri, "Effective Seismic Input through Rigid Foundation Filtering," p. 4 (presented to a 1977 conference).

[275] *Id.* at 1, Figures 2 and 3.

[276] *Id.* at Figures 5a and 5b.

[277] This Appendix is also Reference A of Dr. Newmark's direct testimony before the Licensing Board. See fn. 121, *supra.* Dr. Newmark's definition of the travel time, tau, is slightly different from that of the applicant. He states that he obtained the travel time parameter tau by dividing the building width by the seismic wave velocity (*id.* at C-13). This would yield somewhat lower values of tau than would the applicant's definition.

[278] Newmark, fol. Tr. 8552, Ref. A, pp. C-10 through C-13, Figures 10 through 15, and Figure 8 (a drawing of HSB indicating the location of accelerographs). The two earthquakes for which records were available were Kern County on July 21, 1952 (7.2M_L, 107km from HSB); and San Fernando on February 9, 1971 (6.4M_L, 35km from HSB) [Earthquake data from Applicant's IV-79 testimony, Exh. 1 (TERA Corporation Report), Table 3-3, p. 3-11].

north-south and east-west directions show a distinct reduction in building motion as compared to the parking lot. At higher frequencies, that difference increases to a factor of about 2. The reduction for the Kern County event (107km to the north) is very much less but still evident in the frequency range 2-20 hertz. In the lower frequency range, however, the spectra for the parking lot and the building are virtually identical.

For both earthquakes, Dr. Newmark's response spectra employ travel time (*i.e.,* tau) corrections.[279] The spectra generated using the tau of 0.08 seconds and 0.12 seconds agree in shape and magnitude with the spectra obtained from the building motion records in the north-south and east-west directions respectively (the tau = 0 spectra are merely those generated from free-field records).[280] From the results displayed in these figures, Dr. Newmark concluded that a tau correction properly characterizes the high frequency motion reduction phenomenon, with tau established as the effective building width divided by the seismic wave velocity, *i.e.,* the wave transit time.[281] Dr. Newmark then defined a reduction factor to be applied to a standardized response spectrum based on the value of tau for a given building.[282]

Intervenors and the Governor argue that the tau effect as introduced by Dr. Newmark requires the ground motion to be horizontally propagating

[279]Newmark, fol. Tr. 8552, Ref. A, Figures 12 through 15. Dr. Newmark's testimony does not indicate by what method he obtained the tau-reduced spectra. He references papers by H. Yamahara and R.H. Scanlan, both of which include analytical techniques comparable in general to those of Ray and Jhaveri (fn. 274, *supra*); we may fairly assume that he followed such procedures.

[280]Similar results are obtained for the Kern County event for tau of about 0.08 sec.

[281]Newmark, fol. Tr. 8552, Reference A at p. C-13.

[282]From Newmark, Reference A at pp. C-14 and C-15:

$$\text{Reduction Factor} = R = \frac{\text{Acceleration in Foundation}}{\text{Acceleration in Free Field}}$$

$$R = 1 - 5 \times \text{tau.}$$

For the Diablo Canyon Containment Building,

$$\text{tau}_{DC} = \frac{\text{Building Width}}{\text{Wave Velocity}} = \frac{160\text{ ft.}}{4000\text{ ft. sec}} = 0.04.$$

Therefore,

$$R_{DC} = 1 - 0.2 = 0.8,$$

and the foundation acceleration for the building is reduced from the free-field value of 0.75g to 0.8 x 0.75 = 0.60g. Different values of tau apply to different structures at the Diablo site (see fn. 266, *supra*).

waves (*i.e.,* the motion reduction is solely due to a wave passage) and the effect would not be apparent for vertically propagating waves. The basis for this objection is apparently Dr. Newmark's definition of the tau factor itself. We believe this complaint is poorly founded. For the record shows that, while the tau factor is defined on the basis of horizontal wave passage, the effect itself as viewed by Dr. Newmark encompasses both wave passage and wave inhomogeneity effects.

In Reference A, where the tau effect is introduced on the basis of horizontal wave transit time, Dr. Newmark nevertheless observed that the parameter is "more closely associated with the averaging of accelerations over the area of the structure than it is with an actual wave transit time."[283] In other portions of his direct testimony, Dr. Newmark specifically associated the tau effect with all variations of ground motion over the area of a foundation rather than with simply wave passage time, citing phase differences due to ground inhomogeneities and scattering (incoherence).[284]

We believe that despite the confusion associated with the definition of tau in terms of wave passage, it is quite clear that the tau effect as discussed by Dr. Newmark was intended to include as well spatial inhomogeneities in the wave motion over the foundation surface, a characteristic of virtually all seismic motion.[285] For example, an important source of data on the tau effect was the work of Dr. Yamahara, in Japan, to which Dr. Newmark made frequent reference not only in his prefiled direct testimony but on cross-examination.[286] Dr. Yamahara displayed a number of ground motion records for a building with associated near-field measurements. A review of his paper reveals that Dr. Yamahara indeed identifies a tau-like parameter to use in a high frequency filter analysis to explain his observations (similar

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[283]Newmark, fol. Tr. 8552, Ref. A at p. C-14.

[284]Newmark, fol. Tr. 8552, Ref. B at pp. 11-12; see, also Tr. 8567-68. We have observed that the tau effect reduction comes about as a result of averaging ground motion over the area of a building foundation. Clearly, horizontally propagating pure harmonic wave motion can cause different portions of the foundation area to be subjected to different phases of the motion if the wave lengths of the harmonic waves are short compared to building dimensions (*i.e.,* for higher frequencies). A similar effect will also result if the ground motion is inhomogeneous — *i.e.,* chaotic — rather than harmonic. (See Trifunac, fol. R.Tr. 1138, Fig. V2, for an illustration of such motion.) In the case of chaotic wave motion, phase differences across the area of a building foundation occur at random and are not dependent upon the direction of wave propagation, as in the case of pure harmonic waves.

[285]Tr. 9279, 9294, 9349, 8565-66. Intervenors suggest that the basis for the tau effect has changed with the passage of time (J.I. IV-79 Proposed Findings at p. 33, fn. 16, citing Luco and Trifunac) with the notion of wave motion incoherence coming later. We believe the discussion above shows clearly that this insinuation is not correct, and that wave motion incoherence has always been a part of the explanation of observed motion reduction in large buildings. Intervenors may have been misled by the use of the tau parameter to characterize these effects.

[286]Newmark, fol. Tr. 8552, Ref. A at p. C-18 (Ref. 14). H. Yamahara, "Ground Motions During Earthquakes and the Input Loss of Earthquake Power to an Excitation of Buildings," Soils and Foundations, Vol. 10, No. 2, pp. 145-161 (1970). See, also Tr. 9296, 9346.

to that of Ray and Jhaveri, fn. 274, *supra*). But his discussion of the physical basis for the reduced motion in large buildings focuses almost exclusively on the incoherencies observed at high frequency in ground motion records.[287] Dr. Yamahara also indicated that an assumption of vertical rather than horizontal wave propagation yields the best interpretations of his observed results.[288] Simply in light of his repeated references to Dr. Yamahara's work, only a very crabbed reading of Dr. Newmark's testimony could assume that he did not appreciate tau in all its ramifications.

As we also noted (at pp. 965-966, *supra*), records taken at the Hollywood Storage Building during the 1971 San Fernando earthquake played a significant role in Dr. Newmark's quantification of the tau effect. Of concern was the fact that Diablo Canyon, close to the probable source of strong seismic motion, might receive a larger component of vertically propagating seismic motion than did the Hollywood Storage Building, some 37km from the San Fernando event. This raised questions about the validity of transferring observations made at the HSB to Diablo Canyon.[289] Our concern in this regard was relieved by the unchallenged rebuttal testimony of Dr. Frazier. He called our attention to the results of investigations indicating that, even for sites far from the source, the high frequency components of earthquake motion are likely to arrive in the vertical direction.[290] This, he explained, is based on the fact (previously noted) that high frequency waves are rapidly attenuated in the surface layers of the earth.[291] This portion of seismic motion is therefore transmitted laterally in the rock deep below the surface and emerges in a generally vertical direction even at distances from its source. Dr. Frazier testified accordingly that such waves emerged at least as steeply at the Hollywood Storage Building as they would at Diablo Canyon from an event on the Hosgri.

We conclude *en passant* that there is a reduction in the high frequency ground motion transmitted to the rigid foundations of large structures — an effect more pronounced at higher frequencies — because the different parts of such structures are affected by vibrations that are out of phase.[292] Analytical studies of such motion demonstrate that, in these circumstances, the foundation *as a unit* vibrates less than individual points on the earth underneath it. Spatial variations in ground motion can result from wave

[287]Yamahara, fn. 286, *supra*, at pp. 145, 146, 160.
[288]*Id.* at 160.
[289]See, *e.g.*, Gov. IV-79 Proposed Findings at p. 43, fn. 17.
[290]Tr. 10,128-36. Dr. Luco uses Dr. Frazier's work in this area as the basis for some of his own conclusions (Tr. 8879-80; see also Tr. 10,127.
[291]See fn. 109, *supra*.
[292]See fn. 284, *supra;* Smith, IV-79 testimony at p. V-3.

passage effects and inhomogeneities in the seismic motion itself. The quantification of the tau effect for the Diablo Canyon reanalysis, rather than being derived from a simple wave passage model, appears to have been influenced strongly by actual observations of the effect at the Hollywood Storage Building. As we noted, those observations find confirmation in similar studies conducted independently in Japan by Dr. Yamahara.

B. Specific Challenges

To this point, the objections we have considered were bottomed essentially on allegations that tau gave inadequate consideration to vertical components of seismic motion at Diablo Canyon. We found those objections unfounded because the tau effect reductions do indeed encompass the effects of inhomogeneous and vertically propagating waves. A number of objections to the use of the tau effect were also made on other grounds; we turn to them in the succeeding pages.

1. There would be no appreciable tau effect at Diablo Canyon because of its proximity to the Hosgri Fault.[293]

Dr. Newmark testified that wave inhomogeneities can result from a number of causes; for example, passage through imperfect (inhomogeneous) soil and rock and the reflection of vertical waves at the earth's surface.[294] The only available direct evidence on earthquake motion inhomogeneity in the near field are data from the El Centro Differential Array obtained in the IV-79 event. These ground motion records were obtained at five seismic stations within 214 meters of one another and located about 5km from the rupturing Imperial Fault. Applicant, through its witness Dr. Smith, put into evidence several different analyses of these data that demonstrate incoherence effects increasing with frequency.[295] Briefly, those analyses showed that incoherence effects were small for frequencies below about 5 hertz. But in the case of two stations located about 50 meters apart, they also revealed that only about 80 percent of the seismic power between 5 and 15 hertz was linearly related (*i.e.*, coherent).[296] This is of significance because the Imperial Valley is composed

[293]This point is raised in joint intervenor's original appeal (Brief on Exceptions at 46) as well as in their IV-79 Proposed Findings (at pp. 35-36); it is also pressed by the Governor (Proposed Findings at p. 40) and was supported by Dr. Luco's testimony below. See, *e.g.*, Tr. 8887-88.

[294]Tr. 9294.

[295]Smith IV-79 testimony at pp. V-1 through V-6; R.Tr. 1386-96; App. Exhs. R-13, R-14, R-15, and R-16.

[296]R.Tr. 1395; App. Exh. R-16.

largely of relatively homogeneous soil,[297] a geological condition less apt to cause inhomogeneities than the rock formations underlying Diablo Canyon.[298]

Both Dr. Newmark[299] and Dr. Frazier[300] testified that seismic records obtained during blast testing display near-field inhomogeneities. Because such tests provide a single seismic motion source, they are more likely to produce homogeneous motion at a measurement point than an earthquake, which represents a multitude of seismic sources releasing energy over a considerable distance.[301]

In short, the evidence presented supports the concept that the near-field, high-frequency seismic motion at the Diablo Canyon site would be incoherent or chaotic in nature because it will be generated at many locations along a rupturing fault and must pass through deformed rock strata to reach the site.[302] Thus, there is no reason to discount the tau effect because of Diablo Canyon's closeness to the Hosgri; the conditions underlying that effect are not diminished by the proximity. This conclusion is underscored by the reminder that it was observations of the survival of large buildings in the near field of earthquakes that led to the investigation of the tau effect in the first instance.[303]

2. The foundations of the Diablo Canyon facility are insufficiently rigid for a tau effect reduction.[304]

There appears to be no dispute that the tau effect is operative only in large buildings with rigid foundations.[305] Joint intervenors' objection based on the rigidity *vel non* of the Diablo facility stems primarily from a study made by Dr. Trifunac.[306] In that study, he was not able to correlate data for reduced building motion with building size. This led him to conclude that some of the buildings studied had foundations of insufficient rigidity to register any tau effect. Be that as it may, this record establishes that the Diablo foundations are all rigid ones.[307] In the absence of any contrary evidence at all we find no merit in the objection. Indeed, we raised this

[297]Seed IV-79 testimony at V-4.
[298]*Id.* at V-6; Hamilton IV-79 testimony at V-9.
[299]Tr. 8568-69; Tr. 9329-30.
[300]Tr. 10,134-35.
[301]R.Tr. 1296, 1329.
[302]See fn. 298, *supra.*
[303]Newmark, fol. Tr. 8552, Ref. B at 11; Blume, fol. Tr. 6100 at 42.
[304]This point is raised in Joint Intervenor's Brief on Exceptions at p. 45 and their IV-79 Proposed Findings at p. 43.
[305]See fn. 272, *supra.*
[306]Trifunac, fol. R.Tr. 1138 at pp. V-3 to V-7.
[307]Blume, IV-79 testimony at V-4 to V-5; R.Tr. at 1301-1302; R.Tr. 690-691.

point ourselves in Question 5, which sought the reason for the apparent lack of a tau effect at the Imperial Valley Service Building during IV-79.[308] Witnesses for the applicant, the staff, and Dr. Luco testified that the Services Building was in general poorly designed for earthquake resistance and in particular lacked a rigid foundation.[309] In this circumstance, the structural failure of the Imperial Valley Services Building sheds little (if any) light on the tau effect controversy.

3. The observation of motion reduction at the Hollywood Storage Building is not sufficient to justify the use of tau effect at Diablo Canyon.[310]

As we mentioned earlier, the tau effect was quantified essentially from ground motion recordings made at the Hollywood Storage Building during the Kern County and San Fernando earthquakes.[311] In studying response spectra developed from those data, Dr. Newmark discerned a frequency-dependent reduction in motion that he associated with the tau parameter.[312] But it is not accurate to assume that the tau effect is justified solely on this single study. Other sets of records also indicate motion reduction at high frequencies. For example, as we mentioned earlier, Dr. Newmark also relied on the work of Dr. Yamahara in Japan. Dr. Yamahara presented actual motion records that demonstrated a lack of high frequency motion in a large building compared to nearby free-field stations. There is no mistaking the incoherence of free-field motion at high frequencies in that study; the coherency of lower frequency motion and the fact that near-field and building motions are comparable at lower frequencies are similarly clear.[313] And, like Dr. Newmark, Dr. Yamahara adopted a parameter analogous to tau in substantiating the analytical explanation of his results.[314]

In our judgment, the data and analyses presented in Dr. Newmark's Reference A[315] (including the referenced works by Yamahara and Scanlan), provide a justification for the tau effect reduction that goes well beyond the data presented for the Hollywood Storage Building in that document. The use of tau cannot be fairly criticized as based solely on the reaction of that one structure.

[308] See ALAB-593, *supra*, 11 NRC at 890-91.
[309] Newmark, fol. R.Tr. 534 at pp. 14-16; Blume, IV-79 testimony at pp. V-2, V-4 to V-5; Luco, fol. R.Tr. 1138 at p. 5-18.
[310] See J.I. IV-79 Proposed Findings at pp. 39-45.
[311] See pp. 965-967, *supra*.
[312] Newmark, fol. Tr. 8552, Ref. A, Figures 10 to 15.
[313] Yamahara (fn. 286, *supra*) at p. 147, Figures 2, 3 & 4.
[314] *Id.* at p. 148, Figures 5 & 6.
[315] Newmark, fol. Tr. 8552.

We stress again that the structural failure of the Imperial Valley Services Building is not evidence that the tau effect is inadequately proven. That Building did not meet the prime requirement for displaying a tau effect — that of a rigid foundation.[316] Moreover (as distinguished from the Hollywood Storage Building), the response spectra obtained for the IVSB bear no similarity to those for the adjacent free-field at any frequency.[317] At the Hollywood Storage Building, the free-field and building spectra coincide at low frequencies. At higher frequencies, they differ in magnitude — but very little in shape. This behavior of the HSB spectra is reproduced very well by applying, as Dr. Newmark, did, a tau correction to the free-field data. We conclude, therefore, that the behavior of the IVSB does not affect the validity of the tau concept. Rather, it illustrates the absence of conditions predicate to that phenomenon.

Joint intervenors also argue that there must be soft soil for the tau effect to take place, a condition met at the Hollywood Storage Building but not at Diablo Canyon.[318] However, the testimony they cite (Tr. 8879 and 8975) is not directed to that point. Rather, Drs. Trifunac and Luco were there referring to conditions pertinent to a calculation of motion reduction resulting from a soil-structure interaction. They were simply not addressing the type of motion reduction that would be caused by wave passage and wave incoherence effects. (We discuss the soil-structure interaction at pp. 977-978, *infra.*)

That to one side, Dr. Newmark noted that the data supporting the tau effect comes (in part) from measurements made during weapons tests indicating less damage than expected for both rock and soil foundations.[319] In Dr. Newmark's view, the tau effect is a result of high frequency wave propagation and incoherence that exists even for steeply emergent waves and not a function of whether the site and conditions are soil or rock.[320] The record supports Dr. Newmark on this point.

4. If there is a tau effect, there will be a concomitant torsion excitation of the structure which has not been accounted for in the Diablo Canyon analysis.

Joint intervenors also contend that any tau effect reduction associated

[316]See pp. 970-971, *supra.*
[317]Blume, IV-79 testimony, Figures V-5 and V-6; see also pp. V-8 and V-9. Dr. Blume observes that the building spectra are affected by the fact that the building structure is failing during the IV-79 event (pp. V-10 and V-11). Other reasons for distinguishing the IVSB situation from that of the HSB insofar as tau is concerned appear in Dr. Blume's IV-79 testimony at pp. V-1 to V-4.
[318]J.I. Br. at pp. 44-45, citing Tr. 8879, 8975.
[319]Tr. 9229-30.
[320]Tr. 9279, 9293-4.

with wave passage will be accompanied by a "torsion excitation"[321] and complain that the latter effect was neglected in the Hosgri reanalysis.[322] The record, however, does not support intervenors on this point. For example, in describing the tau effect, Dr. Newmark explained that:

> [c]onsistent with the concept of a wave motion of earthquake deformation, there are torsions and tiltings of a building foundation. Both effects are less on rock than on soil. The torsional effects are taken account of in current codes by assuming an eccentricity of horizontal seismic force of 5 percent of the width of the structure. This effect is less, however, for a very large structure, and the tilting effect is even smaller. Account should be taken of these effects in design.[323]

And Dr. Blume, testifying for the applicant, not only touched on the matter of torsional forces, but stressed that the structures of Diablo Canyon are such that those effects would probably not be noticed. He added, however, that an assumed eccentricity was in fact considered in the Hosgri reanalysis.[324] The direct testimony of the NRC staff similarly deals with this subject. Among other things, it describes in considerable detail the torsion analysis criteria used for the Hosgri reanalysis of Diablo Canyon and indicates just how these factors were applied to each major structure at the plant.[325]

As we perceive it, the intervenors' position on the torsion response stems essentially from the testimony of Dr. Luco, who was critical of the techniques used by the applicant and the staff to determine these effects.[326] Later, however, Dr. Luco explained that his testimony had not intended to convey the idea that torsional forces were ignored but, rather, that the subject was quite complex and in his judgment could have been better analyzed. He further testified that some of the particular points he had in mind when he wrote his prefiled testimony "have been corrected."[327] And, on cross-examination, Dr. Luco candidly acknowledged that motion records obtained at the corner of the Hollywood Storage Building (which

[321]Torsion excitation is an earth movement that tends to rotate or twist the affected structure.
[322]J.I. Br. at 46.
[323]Newmark, fol. Tr. 8552, Reference A at p. C-16. Dr. Newmark also described in his direct testimony his analytical developments of the torsional effects which accompany a tau effect. *Id.*, Reference B at pp. 14-17.
[324]Blume, fol. Tr. 6100 at 44-45.
[325]Knight, fol. Tr. 8697 at pp. 13-24.
[326]Tr. 8890-92.
[327]Tr. 8930.

were used in the reanalysis) would have included torsional motion were it present,[328] a fact Dr. Newmark had also mentioned.[329] Finally, as Dr. Frazier explained, much of the motion at Diablo would be propagating vertically and therefore cause minimal torsional excitation,[330] a phenomenon attributed primarily to horizontally propagating waves.

In sum, the record confirms that torsional forces were adequately considered in the Diablo reanalysis; intervenors' complaint that they were not is simply ill-founded.

5. The motion reductions upon which the tau effect is based are frequency-dependent, whereas Dr. Newmark's correction for the tau effect is essentially uniform down to about 1 hertz.[331]

Intervenors and Governor Brown note that the reduction in building motion due to the tau effect is "frequency-dependent." That is, the magnitude of the correction generally increases as the frequency increases. This dependence is particularly displayed by the incoherence data obtained during IV-79 from the El Centro Differential Array.[332] It is also borne out by a number of the analyses undertaken to determine the magnitude of the effect that were called to our attention in this proceeding.[333]

Applicant's analytical approach to determining the tau effect produced response spectra in which the tau-correction began at a threshold frequency of about 2 hertz and gradually increased to a maximum value determined by the value of tau. This is best illustrated in the Hosgri reanalysis by Figure 10A of Report D-LL 10. The basic Blume response spectrum is presented there together with tau-corrected versions. That figure (modified for illustrative purposes) is reproduced as Figure 4 of this opinion (*infra*, p. 976). Figure 4 also displays the tau-corrected Newmark spectra, in which the correction begins at frequencies between about 1.7 and 1.5 hertz, depending upon the value of tau; above this, the correction (reduction) is more or less constant and independent of frequency. The Newmark form of the tau correction represents a consistent interpretation of the empirical results obtained from the Hollywood Storage Building. Data from that source displayed the reduction in motion in the building compared to the

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[328]Tr. 9162.
[329]Fol. Tr. 8552, Ref. B at p. 13; Tr. 9277.
[330]Tr. 10,168.
[331]J.I. IV-79 Proposed Findings at 35-39; Gov. IV-79 Proposed Findings at 39-42.
[332]See pp. 969-970, *supra*.
[333]See, *e.g.*, the works by Ray and Jhaveri (fn. 274, *supra*); R.H. Scanlan, "Seismic Wave Effects on Soil-Structure Interaction," 4 Earthquake Engineering and Structural Dynamics, 379-88 (1976) (see fn. 279, *supra*); and Yamahara (fn. 286, *supra*).

free field to be only slightly frequency-dependent above an inception frequency of roughly 2 hertz.[334]

For purposes of the issues before us the intervenors' criticism is wide of the mark. It is to be recalled that when the Diablo Canyon facility was reanalyzed to take account of the Hosgri Fault, calculations for the responses both of structures and equipment were performed not only using Dr. Newmark's spectra, but also Dr. Blume's. The latter's spectra do in fact employ a frequency-dependent tau factor; where they exceeded Dr. Newmark's values Dr. Blume's results were deemed controlling.[335] Indeed, as Figure 4 shows, in the 2.5 to 8.0 hertz frequency range (a region of major concern[336]) the tau-corrected Blume spectra have the highest values of spectral acceleration and therefore were employed.[337]

The differential array data obtained at IV-79 demonstrate the existence of incoherence, but these data were (obviously) not used to determine the tau reduction at Diablo Canyon. As noted, the tau-corrected spectrum selected for the Hosgri reanalysis was the highest composite of the Newmark and Blume spectra. So used, these spectra fairly account for the frequency dependence of the tau effect motion reduction in an appropriately conservative fashion.

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[334]Newmark, fol. Tr. 8552, Reference A, Figs. 10 and 11.

[335]See p. 940, *supra;* Blume fol. Tr. 6100 at p. 41.

[336]The natural frequencies of the major structures and systems for Diablo Canyon are discussed in D-LL 42, and depicted in Figure 42-A of that Appendix. The approximate values are:

Structure or System	Natural Frequency (hertz)
Turbine Building	1.4 (after modification)
Auxiliary Building	2.0
Containment Shell	4.5
Interior Containment Structure	10.0
Reactor Pressure Vessel	14.0
Westinghouse Piping Systems	2.9 - 16.0 (range)
Nuclear Steam Supply System	5.5 - 10.0 (range)

[337]We have drawn the heavy line on Figure 4 to indicate which of the two spectra — Dr. Newmark's or Dr. Blume's — is controlling in the various frequency ranges for tau = 0.04. We also added a frequency scale on the horizontal axis that does not appear on the figure from which this is taken.

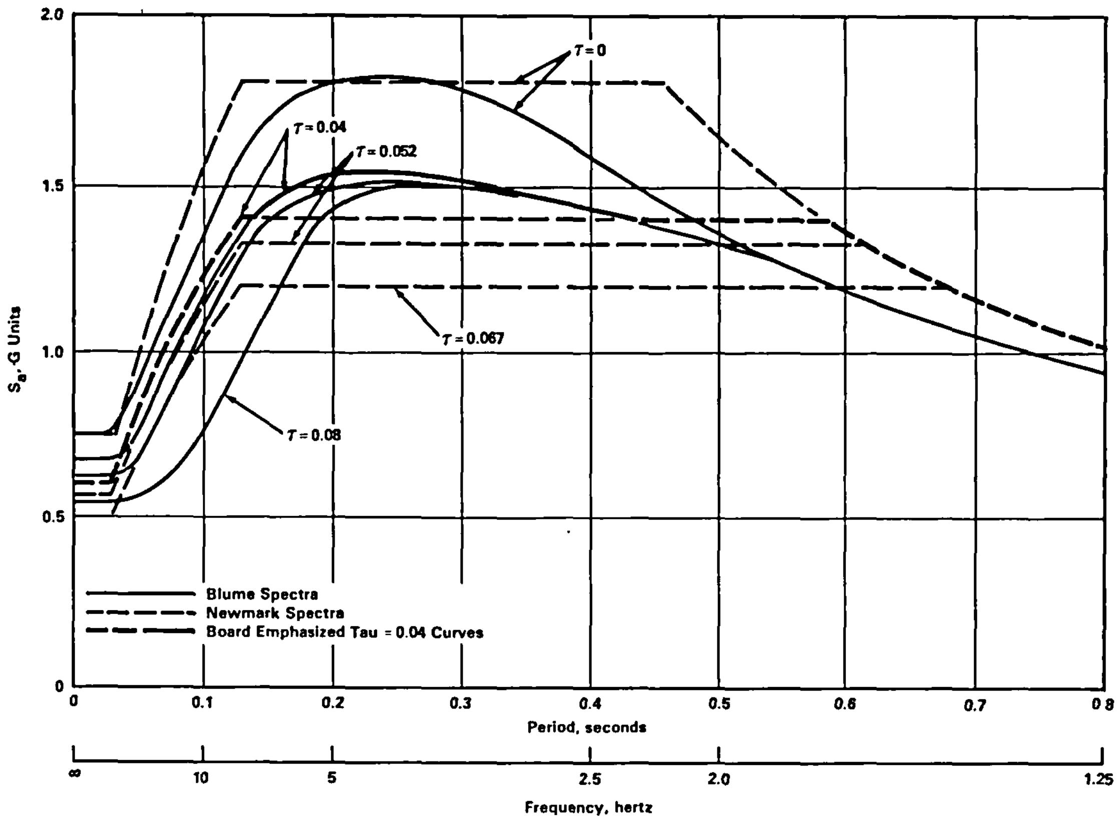

Figure 4 Diablo Canyon Units 1 & 2 Comparison of Hosgri 7.5M Blume & Newmark Spectra for Various Tau Values 7% Damping

6. Because the impinging motion at Diablo Canyon will be primarily vertical, the only basis for a reduced response in building motion is the interaction between the structure and the underlying soil. Such an interaction is insufficient to justify the response spectrum reductions used in the Diablo Canyon analysis.[338]

The "soil-structure interaction" phenomenon is associated with massive buildings that rest upon or are embedded in the earth. The earth can be deformed and therefore may absorb some of the impinging seismic wave motion with the result that the structure's response is different than that of the underlying soil.[339] The effect has been characterized as a third mechanism that may serve to reduce the response of a large structure to seismic motion.[340] (The other two are the wave passage and incoherence effects we previously discussed. See p. 967-969, *supra.*)

The question of soil-structure interaction has been controversial and a source of much misunderstanding. Dr. Luco appears to have been the first to inject it in these proceedings, expressing the view that the applicant and staff's use of tau could not be supported because that effect requires horizontally propagating waves and the incoming waves at Diablo Canyon from a Hosgri event would be vertical.[341] It was Dr. Luco's belief that the applicant's soil-structure interaction analyses were inadequate[342] and he therefore recommended elimination of the tau concept in favor of "a complete three-dimensional soil-structure analysis,"[343] a viewpoint shared by Dr. Trifunac.[344]

Soil-structure interactions and their consequential effect on the ability of large buildings to survive earthquakes are no doubt important issues in seismic engineering. But they are not central matters in this case. The record is clear that the tau-effect reductions in the Diablo Canyon response spectra were not formulated on the basis of such interactions. Rather, those reductions were derived from studies of actual motion records (such as the ones from the Hollywood Storage Building[345] and those reported by

[338]Gov. IV-79 Proposed Findings at pp. 46-51; J.I. IV-79 Proposed Findings at pp. 41, 45-46; J.I. Br. at pp. 46-47.
[339]Tr. 6773-74.
[340]See Seed, Tr. 10,162-66.
[341]Board Exh. 2C (July 7, 1978 Comments to ACRS, p. 1; May 30, 1978 Report to ACRS, pp. 3-4); Tr. 8881.
[342]Board Exh. 2C (May 30, 1978 Report to ACRS, pp. 6-8). This opinion preceded the Seed-Lysmer report (fn. 348, *infra*), however, a work with which Dr. Luco seems to have no quarrel, though he personally preferred other methods (Tr. 9033-35).
[343]Board Exh. 2C (May 30, 1978 Report to ACRS, p. 8).
[344]Board Exh. 2D (Comments to ACRS, April 1978), p. 6.
[345]In his IV-79 testimony (pp. 5-11 and 5-19), Dr. Luco argues that the Hollywood Storage Building motion records show a soil-structure interaction effect rather than a tau-effect. He then discusses analyses of the response of that building to the 1952 Kern County earthquake

Dr. Yamahara), and are attributed to causes discrete from soil-structure interaction, a conclusion we have found firmly supported in the record. See pp. 972, *supra.*

To lay the point to rest, we note that the NRC instructions governing the Hosgri reanalysis required that the earth underlying the Diablo facility be assumed rigid — *i.e.,* that in an earthquake the structures would move as one with the underlying ground.[346] This means that in developing the response spectra no credit was taken for soil-structure interaction. Thus, to the extent that there is some reduction in building motion fairly attributable to that cause, it represents a further conservatism and not a ground for criticism of the reanalysis.[347]

In sum, the tau-effect motion reductions are reasonable ones well supported in the record; any further reduction in building motion attributable to possible soil-structure interactions should be considered as a conservatism. If, as Dr. Luco has suggested, the Hollywood Storage Building record includes not only inhomogeneity and wave passage effects but soil-structure interaction as well, we are satisfied by the calculations performed by Dr. Seed that it is reasonable to expect comparable motion reductions at the Diablo Canyon site.[348]

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and concludes that "soil-structure interaction proper was not sufficient to produce the observed reduction and that the reduction was associated with scattering by the basement and piles" (p. 5-18). With due deference, we cannot reconcile Dr. Luco's initial statement with this conclusion. Dr. Luco also presented results of soil-structure interaction analyses (p. 5-11, Table 2). These, however, also show little or no difference between basement and free-field motion. Again, these results appear to undercut rather than support his argument that the Hollywood Storage Building records reflect soil-structure interaction.

[346]See Blume, IV-79 testimony at pp. VI-1 to VI-2.

[347]We note that Dr. Seed and Dr. Newmark testified that soil-structure interaction effects should not be lumped into an overall tau-effect but considered separately from the effects of wave passage and inhomogeneity. Tr. 9333; 10,167. Both also expressed the view that the effects of such interaction at the Diablo site would be negligible. Dr. Newmark put the reduction at less than 10 percent (R.Tr. 676-678); Dr. Seed quantified his estimate as a 20 percent reduction in seismic motion limited to the 4.0 to 25 hertz range (Tr. 10,149).

[348]Dr. Seed testified at the hearing below about calculations he and Dr. John Lysmer had performed for the applicant (Tr. 6770-74). These compared the ground and structural motion in a simulated Diablo Canyon containment structure resting on a rigid earth base with two cases of seismic motion for which the earth is considered deformable (a) for vertically propagating S and P waves and (b) horizontally propagating Rayleigh waves. In the latter case, the earth is assumed to have the actual properties of the rock underlying the Diablo Canyon site. Figure 13 of the Seed-Lysmer report (J.I. Exh. 58) illustrates their results for a point at the center of the building foundation. The response spectra calculated for the deformable base show reduced magnitudes in the higher frequency ranges, in general accord with Dr. Seed's estimates. This figure was also offered in the Seed IV-79 testimony as Figure VI-1. (We had asked that soil-structure interaction effects, with particular reference to the Diablo Canyon site, be addressed to the reopened hearing. See ALAB-598, *supra,* 11 NRC at 891, Question 6.)

VI

DAMPING

A vibrating system will slow and eventually cease oscillating because of unavoidable energy losses from friction or analogous phenomena operating to dissipate its original energy. The rate or degree of that energy loss varies with different materials and systems and is usually expressed as a percentage of "critical damping," *i.e.,* that amount of damping at which vibratory motion could not exist. The significance of damping for this case centers on the need to factor into the Diablo Canyon seismic response spectra appropriate damping values for the facility's bolted steel and reinforced concrete structures. Regulatory Guide 1.61 (October 1973) states that a 7 percent damping factor should be appropriate for that purpose in the absence of documented tests that would support a higher value. That value was in fact applied in the Hosgri reanalysis. The joint intervenors challenged the figure as too high, relying on testimony of Drs. Luco and Trifunac. The Board below, however, accepted the views of the applicant's and staff's experts and found the 7 percent figure to be both appropriate and conservative. LBP-79-26, 10 NRC at 496-97.

After capsulizing the testimony of the various witnesses, joint intervenors' brief on appeal asserts (at 54, reference omitted):

> Finally, we turn to damping, here again, the Licensing Board's decision is devoid of a fair explication of the evidence. The Licensing Board cites mostly general concepts, and fails to discuss the views of Trifunac and Luco.

From that argument (quoted in its entirety) the intervenors would have us conclude that "the evidence in this case does not demonstrate that . . . 7% damping values are appropriate and conservative" (*id.* at 55).

The Licensing Board devoted substantial attention to the damping phenomenon in its opinion. See 10 NRC at 494-96. Its discussion may not satisfy the intervenors, but the Board did indicate the basis and the reasoning that led to its conclusion. Unfocused objections of the kind intervenors raise here are singularly unhelpful in our review of the Board's decision. In *Black Fox,* where we were faced with similar generalized complaints, we explained:

> we may not "make an appellate determination on a clean slate without regard to the Licensing Board's opinion" and do not "weigh each piece of evidence *de novo.*" Rather, "the decision below is 'part of the record'; we may, indeed must, attach significance to a licensing board's

evaluation of the evidence and to its disposition of the issues." By neglecting to address their brief to the decision under review and by omitting adequate record citations, intervenors leave us (and the appellees) guessing about the precise nature of their arguments and ignorant of the evidence they rely on to support them.[349]

As we stressed there, the record need not be searched for unspecified error even in criminal cases.[350] In this case as in that one, the circumstances justify treating essentially unbriefed issues as abandoned.[351]

We have, nevertheless, elected to review the record on the point as best we can to assure ourselves of the soundness of the Licensing Board's decision. As summarized in the margin,[352] the staff and the applicant

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[349]*Public Service Co. of Oklahoma* (Black Fox Station, Units 1 and 2), ALAB-573, 10 NRC 775, 805 (1979) (citation omitted).

[350]*Id.* at 806 and authorities cited there in fns. 131 and 132.

[351]*Ibid.*

[352]The applicant initially used a value of 5% for damping as then required by Regulatory Guide 1.61. Prior to the Hosgri reanalysis, however, new data had caused the staff to revise Regulatory Guide 1.61 in October 1973 to allow the use of 7% damping in the analysis of bolted steel and reinforced concrete structures. Blume, fol. Tr. 6100 at pp. 14-15. The applicant's basis for adhering to the new value in the Hosgri reanalysis is described in the record in a paper by Dr. Blume and Ahmad R. Kabir, "Data on Damping Ratios," D-LL 9. From the results of (a) tests on two other nuclear reactor containment buildings, (b) damping determinations on 22 concrete building components, (c) tests on bridge piers, (d) tests on models of shear walls and (e) a series of tests on scaled buildings models, the authors concluded that "a 7% damping ratio for the main structures seems proper, even conservative for a very severe seismic loading such as M = 7.5 adjacent to the Diablo Canyon plant." *Id.* at p. D9.13. They emphasized that their summary results (shown in Table 9.4 of the paper) showed an increase in damping with increasing stress levels and "[i]f the strain level in the reinforcement exceeds about 70% of yield strain, 7% appears to be a realistic value for damping." *Id.* at p. D9.8.

Dr. Blume and Dr. Frazier elaborated on this report at the hearing, noting, *inter alia*, the application of this experience to bolted steel structures and reinforced concrete buildings (Tr. 6486-94; 6552-66), and emphasizing that the damping within reinforced concrete and bolted steel structures is in addition to the so-called "radiation" damping, which refers to soil-structure interaction. In rebuttal testimony, Dr. Blume pointed out that tests carried out by the Portland Cement Association on reinforced concrete shear walls show damping values of 7% or above at loadings comparable to those at Diablo Canyon, under conditions where there can be no radiation damping (Tr. 10,119).

Mr. Knight and Dr. Kuo testified for the staff concerning the 1973 change in Regulatory Guide 1.61, explaining that the new damping values were adopted after a review of all information known to them, including data that "were obtained from forced vibration tests on structures, including reactor buildings and commercial buildings, and from actual earthquake data where available, and were supported by laboratory tests of what I would call structural elements, that is a beam or a section of a wall" (Tr. 9819). Dr. Kuo emphasized that while much of the data included both building and soil damping (damping caused by soil-building interaction was termed "radiation damping" by most of the witnesses), all of them "showed an unquestionable trend toward higher damping as the strain rates increased." *Ibid.* Dr. Kuo also noted that damping measured in the total building-soil system is much higher than 7% and is presented both laboratory and actual building tests that show increasing damping as stress levels increase with the value of damping of up to 10% or more at high stresses for structures alone, and up to 20% for structures and soil combinations. Drs. Luco and Trifunac, on the other hand, relied on their belief that 5% damping would be more conservative and the unpublished tests at the University of California on masonry under stresses of one-half yield point or less (Tr. 8895-97), but ignored the staff's and applicant's test results showing higher damping with increased stress. In our judgment, the weight of the evidence supports the use of 7 percent damping. There is, therefore, no occasion to disturb the Licensing Board's finding that, as recommended by Regulatory Guide 1.61, 7 percent damping was appropriately used in the reanalysis.

VII

THE EFFECT ON THE EL CENTRO POWER PLANT OF THE 1979 IMPERIAL VALLEY EARTHQUAKE

A sizable electric generating station exists only 5km from the fault that ruptured in the 1979 Imperial Valley earthquake. The 174 MW El Centro Steam Plant is a gas or oil-fired, four-unit facility that serves the Imperial Irrigation District. The most recent unit (No. 4), built in 1968, has an output of 90 MW. At the time of the earthquake two of the earlier units were down for repair, leaving only Unit 4 and one other running when the event occurred. Both operating plants tripped off the line, but one was back on line in a quarter of an hour and the other in about two hours.[353]

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fact, Japanese forced vibration tests on their Tokai 2 reactor showed composite damping of about 20% (Tr. 9820). With respect to the building damping alone, Dr. Kuo pointed out that the tests applicant referred to showed that pure structural damping (for concrete and bolted steel buildings) was in the 10% range (Tr. 9821).

The joint intervenors rest on the testimony of Drs. Luco and Trifunac on this point. Dr. Trifunac claims that, while the soil-structure system damping "may be much larger than 7%, inadequate basis has been presented to justify 7% damping in structural systems only." M.D. Trifunac, "Comments on Seismic Design Levels for Diablo Canyon Site in California," (April 1978), Board Exhibit 2D at pp. 3-4. Dr. Luco agreed and referred to tests run at the University of California at San Diego on reinforced masonry (Tr. 8897). These showed a damping of 3% for strengths ranging from very low to about one-half of yield strength. Dr. Luco had not, however, seen results for higher strength at the time of his testimony. *Ibid.* The concerns of Drs. Luco and Trifunac were effectively answered in the rebuttal testimony of Mr. Knight and Dr. Kuo for the staff (Tr. 9818-26).

[353]The El Centro facility was called to our attention in the staff's response of May 5, 1980, to the "Joint Intervenors' Motion to Reopen," as well as in a prior "Board Notification" letter dated December 17, 1979, from Mr. Steven A. Varga of the staff.

Power plants have many similar components; the construction and operating systems of El Centro have at least some resemblance to those at Diablo Canyon. We therefore instructed the parties to examine the effects of the 1979 earthquake on the El Centro plant "to help confirm or refute the analytical techniques and assumptions used in the Diablo Canyon seismic analysis." ALAB-598, 11 NRC at 891-92 (Question 8). Only the applicant and the staff undertook to do so. They presented their results at the reopened hearing in the form of testimony, exhibits and proposed findings that focused mainly on the newest and largest of the El Centro units, Number 4. The Governor restricted his efforts in this regard to cross-examining the witnesses presented and submitting his own findings. (The joint intervenors essentially left this point to the Governor.)

The applicant's analyses were presented in the testimony of Dr. Blume and Mr. Willmer C. Gangloff. Dr. Blume's testimony described the structure of El Centro Unit 4 in some detail, pointing out its substantial differences from Diablo Canyon[354] and that it had been "designed for a 0.2g horizontal seismic load, using simple static analysis techniques, with no consideration of vertical acceleration.[355] On the basis of response spectra derived from ground motion records made during IV-79 at seismographic station less than a kilometer from the El Centro plant,[356] Dr. Blume analyzed both the El Centro steam and turbine buildings with procedures and techniques comparable to those employed in the Diablo Canyon Hosgri reanalysis. That analysis, Dr. Blume testified, "predicted much more damage than actually occurred" at the El Centro plant as a consequence of the 1979 earthquake.[357]

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[354]Dr. Blume observed, *inter alia*, that the four El Centro units are structurally independent and each contains three distinct structures:

> a steel frame and concrete turbine building, containing mechanical and electrical equipment as well as piping systems; a concrete pedestal supporting the turbine and located within, but structurally separated from, the turbine building; and a boiler structure which is a braced steel frame supporting a hanging boiler and structurally connected to the turbine building.

Blume, IV-79 testimony at p. VIII-1.

[355]*Id.* at VIII-2.

[356]The ground motion records used for these calculations were taken from the seismograph at the El Centro Differential Array (USGS No. 5165) 0.85km from the plant. Blume IV-79 testimony at VIII-1; Staff Exh. R-1 at p. 6. This instrument gave peak acceleration values of 0.49g (north-south), 0.35g (east-west) and 0.66g (vertical). Staff Exh. R-1 at pp. 29-30. The validity of using this record to represent motion at the steam plant is supported in Staff Exh. R-1 at p. 6.

[357]Blume, IV-79 testimony at VIII-3. We note in this connection that the El Centro plant though designed only for an 0.2g horizontal seismic load, actually withstood 0.5g, accelerations during IV-79. *Id.* at VIII-5.

Mr. Gangloff's testimony explained how the El Centro plant had been inspected and a computer model of Unit 4 structures and piping prepared. Using the methods employed for the Diablo Canyon reanalysis, the time history of ground acceleration during IV-79 was applied to the computer model. This method was used to predict the responses of various parts of the plant when subjected to a seismic event of the intensity of IV-79. According to Mr. Gangloff, the Diablo Canyon analysis methods "significantly overpredicted the actual [damage] results for the El Centro plant.[358]

The staff's analysis of the effect of IV-79 on the El Centro power plant was performed for it by the Lawrence Livermore National Laboratory.[359] This report also describes the damage incurred by El Centro Steam Plant Unit 4, and explains the Livermore method of modeling the plant to calculate the expected damage under the Diablo Canyon reanalysis criteria. The major difference between the Livermore model and the applicant's was the inclusion of soil-structure interaction.[360] By using high soil damping values, the overall model approximately matched the observed structural reaction of the El Centro plant.[361] The staff's approach and methodology were selected to emphasize the response of the mechanical *equipment* in the El Centro plant as opposed to the *structures* themselves, which were believed to be substantially different from those at Diablo Canyon.[362]

Their study of the El Centro plant's response to the Imperial Valley earthquake led the Livermore Laboratory to draw the following implications for the Diablo facility and other nuclear plants:

> From these data and others available in the literature, it can be concluded that the inherent seismic resistance of engineered structures, piping and equipment is greater than is assumed in both past and

[358]Gangloff, IV-79 testimony, *passim* (the quote appears at p. VIII-8). A specific example of this overprediction was brought forth in the questioning of Mr. Gangloff. For one particular pipe section (node 3230), depending upon the motion assumption used, the amount of displacement predicted by the analyses was from 12 to 21 inches; the actual displacement was 2.8 inches (R.Tr. 1347; Gangloff, IV-79 testimony, pp. VIII-6 and 7).

[359]Staff Exh. R-1, "Equipment Response at the El Centro Steam Plant During the October 15, 1979 Imperial Valley Earthquake" NUREG/CR-1665 (October, 1980).

[360]*Id.* at p. 25.

[361]*Id.* at p. xvi.

[362]Mr. Knight explained that (R.Tr. 1341):

> "In the published report we focused — I should say, sought to achieve a realistic picture of the loads which were actually experienced by the equipment. The analytical results for the structures were adjusted by applying relatively large amounts of damping to the soil-structure interaction analysis to minimize the loads that one would depict as having actually applied to the equipment.
>
> "In this way we felt we would get a conservative view of equipment performance. That is, we would give the least credit for the equipment ability to withstand loads above the design loads."

> current analysis and design procedures. Even facilities designed with very nominal seismic consideration, such as the El Centro Steam Plant, withstand severe seismic environments. It can be concluded that when even the most modest attention is paid in design to providing lateral load carrying paths, significant capability is rendered. In contrast, nuclear power plants are designed to very rigorous techniques. Therefore, it is reasonable to expect even higher inherent margins than are implied in this evaluation. Many of the factors which contribute to our conservative prediction of seismic response during earthquakes can be quantified in light of current knowledge. Other factors are largely unquantified at this time; however, this study and others demonstrate that they do exist.[363]

As we mentioned, neither the joint intervenors nor the Governor furnished any evidence of their own relative to the El Centro plant. The Governor did, however, cross-examine the applicant's and staff's witnesses. In his judgment, "the Steam Plant experience during IV-79 does not provide relevant data for Diablo Canyon" essentially for the following reasons: (1) Contrary to Appeal Board Question 8,[364] the Diablo Canyon and El Centro Power Plant structures are significantly different;[365] (2) the staff analysis takes into account soil-structure interaction and shows that the El Centro Power Plant was not overstressed;[366] and (3) the claim made by applicant's witness that the El Centro plant structure and piping was severely overstressed cannot be substantiated.[367]

With due deference, we cannot agree, for the Governor misapprehends the record. First, both the applicant and staff recognized — indeed emphasized — that despite the fact that there are many similarities in the equipment used, "the steam plant structures are substantially different from the Diablo Canyon structures."[368] Here again (see p. 957, *supra*), the significance of the IV-79 observations is not that they could be transferred directly to the Diablo facility, but that they provide, as it were, a laboratory test of methods used in the Diablo Canyon analysis.

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[363]Staff Exh. R-1 at pp. 40-41.

[364]Appeal Board Question 8 stated in pertinent part that: "In many respects, the structures and systems of [the El Centro Power Plant] resemble those of the Diablo Canyon plant. Their response to a severe, well instrumented seismic event can be analyzed to help confirm or refute analytical techniques and assumptions used in the Diablo Canyon seismic analysis." ALAB-598, 11 NRC at 891-92.

[365]Gov. IV-79 Proposed Findings at pp. 74-75.

[366]*Id.* at pp. 75-77.

[367]*Id.* at pp. 77-78.

[368]Blume, IV-79 testimony at p. VIII-2; Gangloff, IV-79 testimony at p. VIII-1; see also Staff Exh. R-1 at pp. 40-41.

Second, the staff expressly acknowledges that its analysis used very high soil damping; the Livermore report itself recognized that "the resulting composite model damping values shown in Table 6 [of that report] may be unrealistically high."[369] And in his reply to Board questions, Mr. Knight emphasized this deliberately assumed overdamping in their models.[370] Even using the large soil-structure interaction — which tends to reduce motion in the structure — the Livermore analysis found that the plant equipment at El Centro experienced forces 2 to 9 times their specified design load.[371]

Third, in testifying for the applicants, Dr. Blume stated clearly that the El Centro plant was designed to withstand an earthquake induced acceleration of 0.2g.[372] Dr. Blume stressed than an analysis of the El Centro turbine building *using the criteria of the Hosgri reanalysis* "demonstrated that the shear stresses in the operating floor diaphragm at each end of the turbine-pedestal opening exceeded the calculated ultimate stress by a factor of two. In addition, selected columns in the turbine building were found to be overstressed. Yet an inspection of the building resulted in no observation of the predicted damage."[373] Diablo-type analyses of piping and a control panel similarly predicted expected damage to be far greater than actually occurred.[374] The record thus amply supports Dr. Blume's statement that:

> In conclusion, the analyses done for El Centro predicted damage that did not occur. The analyses were done in a manner consistent with the Hosgri reanalyses for Diablo Canyon. The obvious result is that such analyses are conservative in that they theoretically predict damage where in reality none occurs.[375]

In short, the analyzed response of the El Centro Power Plant to IV-79 lends weight to the view that the methodology used in the Diablo Canyon Hosgri reanalyses is indeed conservative.

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[369] Staff Exh. R-1 at p. 26.
[370] Tr. 1347-49; see also fn. 362, *supra*.
[371] Staff Exh. R-1 at p. 39.
[372] Blume, IV-79 testimony at VIII-2.
[373] *Id.* at p. VIII-3 (as corrected at R.Tr. 1335-37).
[374] Blume, IV-79 testimony at pp. VIII-4 through 10.
[375] *Id.* at p. VIII-10.

VIII

THE APPLICANT'S TESTING PROGRAM

Joint intervenors level a broad-gauge attack on the applicant's program for testing the adequacy of the Diablo facility and its components. They assert (Br. at p. 56, capitalization omitted) that:

> The Licensing Board erred in finding that the applicant's program of testing and analysis demonstrated that structures, systems and components necessary to achieve safe shutdown and to maintaining a safe shutdown condition will perform their safety functions during the Hosgri earthquake and aftershocks.

In support of that claim, intervenors cite four items in the staff's analysis of the Diablo facility which they contend depart from normal requirements: (1) the use of the "tau-effect" to reduce the ground response spectra; (2) the use of actual rather than code-specified minimum material strengths; (3) the allowance made for structural ductility; and (4) the method of combining loads for testing purposes.

The tau effect has already been discussed; we found that the use of this device reasonably predicts the motion reduction to be found in large buildings. We turn here to the remaining points.

A. Use of Actual Rather than Code Material Strengths

In its reanalysis of the Diablo Canyon structures for their response to a 7.5M Hosgri earthquake, the applicant used the same basic procedures and analyses as it had used in its initial analysis with certain exceptions.[376] One was the use of "as built" or actual average material properties for the concrete and steel components in the facility structures rather than the engineering code minimum requirements.

Intervenors maintain that this method of calculation is an unacceptable "departure from normal Staff practice" and that "the evidence offered by the Staff and Applicant fails to justify an exemption from the requirements [of] 10 CFR 50.55a" (Br. at pp. 58-59).[377] It is the intervenors' thesis that this section mandates conformance to certain American Concrete Institute and American Institute of Steel Construction codes which require use of minimum specified material strengths for concrete and steel rather than average actual strengths.

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[376]See Hosgri Report (fn. 70, *supra*), Section 4.1 at p. 4.2.

[377]The brief literally refers to 10 CFR § 50.55(a), an obvious typographical error.

To begin with, the cited regulation, while it refers to other industry codes, does not address the ones to which intervenors refer. Be that as it may, both the applicant and the staff acknowledge that, in building design, minimum acceptable material strength values are used in establishing design values.[378] However, in this evaluation we are concerned not with the design of new buildings but with the evaluation of the strength of already constructed facilities; it is for this very reason that applicant used the actual properties of the building components.[379]

Intervenors maintain nevertheless that use of an average value of actual concrete strength is non-conservative. As they see it (Br. at 59), "use of the '*average* actual value' means that one-half are not expected to fall below the computed average." This may be so, but we do not believe it indicates a problem with the analyses that were carried out. The record indicates that (1) the measured variations in the concrete strengths were small (less than 10%);[380] and (2) the concrete strength was based primarily on tests made 28 days after pouring[381] but the actual values today are at least several tens of percent higher due to the aging factor that was ignored in the analysis.[382] The average yield strength of reinforcing steel is more than 10 percent higher than the specified value, and the margin between the measured yield strength and the ultimate strength is much larger for the measured average values than for the specified values.[383]

On the basis of the evidence presented, this exception is simply not well taken.

B. Allowance for Structural Ductility

Under this general heading the intervenors maintain that 10 CFR Part 100, Appendix A, Section VI(a) "permits stresses and strains beyond the yield point [for] 'some' safety related structures, systems, and components only where the safety functions are not impaired" (Br. p. 60). They do not dispute that an analysis of stresses beyond the yield points showed that material yielding would not impair safety functions. Intervenors' point is, we take it, that in their judgment those analyses were inadequate because

[378]Blume, fol. Tr. 6100 at pp. 26-27; SER Supp. 7, fol. Tr. 8183, Section 3.9.3.2(2), p. 3-49.

[379]Blume, fol. Tr. 6100 at pp. 26-27; Tr. 7211-13. See also Knight, fol. Tr. 8697 at pp. 13-14.

[380]D-LL 6, Appendix D-6A at p. D-6A.3 (Table 6A.1).

[381]A small percentage of the tests were made 60 days after pouring. *Ibid.*

[382]Tr. 7194.

[383]D-LL 6, Appendix D-LL 6B, at p. D-6B.2. Staff testimony in support of the applicant's assessments appears in SER Supp. 7, fol. Tr. 8183, Section 3.9.3.2(2) at p. 3-49; and Knight, fol. Tr. 8697 at pp. 13-14.

(a) aftershocks were not considered,[384] (b) no inelastic analyses were done and (c) the Licensing Board neither addressed the deficiencies in component design criteria nor insisted upon an analysis of potential system interaction due to fracture of some systems following a seismic event (Br. pp. 61-62).

Our review of the record convinces us, however, that proper analyses were in fact undertaken. While the Licensing Board did not spell these out, it referred in its opinion to those portions of the record which discuss those analyses in detail. See LBP-79-26, *supra*, 10 NRC at 499-507. Thus, for example, the Board specifically mentions Appendix F of the Hosgri Report, where much of the information in question can be ascertained. We see nothing to be gained in an already lengthy opinion by restating that material here. The Board also cited (*id.* at 504) the direct testimony of applicant's witness Thomas C. Esselman,[385] who discussed the analyses that show the effects of combinations of stresses on various components and systems, and refers as well to his cross-examination on the point.[386]

The Board below might well have articulated its reasoning more fully, but we believe that its path "may reasonably be discerned."[387] Particularly as the intervenors neither point to specific items in error in the referenced testimony nor offered any testimony of their own, we find the exception not well taken and on the basis of the record it must be denied.

C. Combination of Loads on Facility Components and Systems

Among other things, certain piping systems in a nuclear facility must be able to withstand not only normal operational stresses but the additional loads that they might be expected to receive in an earthquake. NRC Regulatory Guide 1.92 provides a method for making the necessary assessment to insure this requirement is met.

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[384]In support of this challenge, intervenors have cited Dr. Newmark's testimony fol. Tr. 8552 at p. 5. What he actually said there was:

> "**Contention 5** - Adequacy of the Dynamic Analysis
>
> The design spectra that I recommended in Ref. A are generally more conservative than those proposed by the applicant, and were intended to be applied without allowance for inelastic structural response except where proper justification could be made. Hence the design criteria were intended to and do cover aftershocks, since no or little permanent deformation would result from the main shock.
>
> "In my review of the structural design, I felt that my intentions were achieved by the applicant."

[385]Esselman, fol. Tr. 7548.

[386]Esselman, Tr. 7549-86. With regard to systems interaction, also see fn. 406, *infra*.

[387]*Bowman Transportation, Inc.* v. *Arkansas-Best Freight System, Inc.*, 419 U.S. 281, 286 (1974).

The intervenors complain that "[t]he methods used in combining seismic stresses with normal operating loads and stresses for the piping systems were not in accord with the method in Regulatory Guide 1.92" (J.I. Br. at 64). However, the Guide itself permits an applicant to propose an alternative method to demonstrate compliance. *Id.* at p. 1.92-4.[388] This was done here. The record reveals that the staff reviewed and accepted the applicant's method of combining stresses; intervenors have apparently overlooked the discussion of the piping system analyses in Supplement 8 to the SER (fol Tr. 8183). There, in section 3.9.3.4, the staff explained how the applicant's analyses compared favorably with the results obtained using the methods of Regulatory Guide 1.92, a point reiterated by the staff in Mr. Knight's prefiled testimony.[389] Intervenors did not challenge this testimony on cross-examination; accordingly, on the basis of this record the exception must be denied.

IX

THE OPERATING BASIS EARTHQUAKE

As we explained earlier, the Operating Basis Earthquake is the strongest seismic event considered likely to occur during the operating lifetime of a nuclear power plant.[390] To be licensed, such facilities must be designed and built to function through an OBE without creating undue risk to the public health and safety. 10 CFR Part 100, App. A, Section III(d).

By way of general background, from the plant design standpoint the distinction between the OBE and the more severe SSE[391] is in essence this: the SSE is the seismic design basis for safety-related or "Category I" structures and equipment and the OBE the benchmark for the balance of the plant.[392] (At Diablo Canyon, however, two structures that would normally fall into the latter category — the turbine building and the intake structure — have been designed with the capability to withstand an SSE without loss of function.[393]) A nuclear plant subjected to vibratory ground motion exceeding that contemplated by the OBE must be shut down until the licensee can "demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation

[388]As mentioned previously (p. 937, *supra*), the Guides are advisory, not mandatory.
[389]Knight, fol. Tr. 8697 at pp. 44-46.
[390]See p. 911, *supra*.
[391]The safe shutdown earthquake is discussed at pp. 910-911, *supra*.
[392]See generally Hoch, fol. Tr. 6879.
[393]Hoch, fol. Tr. 6879 at pp. 14-15; Ghio, fol. Tr. 6941 at pp. 5-7.

without undue risk to the health and safety of the public." 10 CFR Part 100, App. A, Section V(a)(2).

Calling our attention to the fact that the Diablo Canyon SSE is 0.75g and the OBE is 0.2g, the intervenors contend that the Licensing Board erred in holding that the OBE satisfied the regulatory requirements. Their argument rests on Section V(a)(2) of Appendix A, which provides 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.

Intervenors argue that the use of the word "shall" indicates a mandatory intent that the OBE be set at one-half the SSE, *i.e.*, at 0.375g and not at 0.2g for the Diablo plant.

We cannot accept that argument, which is bottomed on maxims of statutory construction. Regulations, like statutes, may neither be read in isolation nor interpreted piecemeal.[394] Intervenors have simply disregarded other relevant portions of the governing Commission regulations allowing departures from the general Appendix criteria where justified in a specific situation. Thus, Section II of Appendix A expressly provides that:

> Each applicant for a construction permit shall investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in these criteria. Additional investigations and/or more conservative determinations than those included in these criteria may be required for sites located in areas having complex geology or in areas of high seismicity. *If an applicant believes that the particular seismology and geology of a site indicate that some of these criteria, or portions thereof, need not be satisfied, the specific sections of these criteria should be identified in the license application, and supporting data to justify clearly such departures should be presented.*[395]

[394]"In expounding a statute, we must not be guided by a single sentence or member of a sentence, but look to the provisions of the whole law, and to its object and policy." *United States* v. *Heirs of Boisdore*, 49 U.S. (8 How.) 113,122 (1849) (per Taney, Ch. J.), quoted in *Philbrook* v. *Glodgett*, 421 U.S. 707, 713 (1975). The same rules of construction apply to administrative regulations. *Rucker* v. *Wabash R. Co.* 418 F.2d 146, 149 (7th Cir. 1969). We need only add that the use of "shall" in the regulation cited by intervenors is not controlling. That term can be merely directory; background, context and the general intent of the administrative agency are what govern. *United States* v. *Reeb*, 433 F.2d 381, 383 (9th Cir. 1970), *certiorari denied*, 402 U.S. 912 (1971); *In re Franklin National Bank Sec. Litigation*, 478 F. Supp. 210, 223 (E.D.N.Y. 1979) (holding that "shall," as used in Rule 56 of the Federal Rules of Civil Procedure, is permissive).

[395]10 CFR Part 100, App. A, § II (paragraph 3) (emphasis supplied).

Intervenors also fail to appreciate the very definition of the OBE itself. The OBE is characterized as "that earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant .... "[396] Put another way, the OBE is defined in terms of local seismic conditions. In this case the applicant followed the alternative means of meeting the criteria by studying the seismicity and geology of the Diablo Canyon site. From these studies it determined the likely occurrence of various size earthquakes and selected as the OBE the largest which could reasonably be expected to affect this facility. Intervenors' argument (J.I. Br. at 81) that "[i]n this case the Staff and the Applicant have substituted their own [OBE] standard for the Commission's," is therefore not well taken; the applicant has done nothing more here than the governing Commission regulation permits.

The procedure just described is not novel and our interpretation of the controlling regulation is not new. The record is uncontradicted that, previous to this case, the OBE's for the Byron, Braidwood, Clinton, Koshkonong, Marble Hill, and Phipps Bend facilities have been found justified and approved by the staff with "a maximum vibratory ground acceleration less than one-half of the Safe Shutdown Earthquake."[397]

Joint intervenors complain, however (Br. at 74), that:

> It is quite possible that the OBE for these plants [*i.e.,* the ones listed at p. 992, *supra*], if indeed they are less than one-half (1/2) the SSE, might be extremely close to being one-half (1/2) the SSE. Tr. at 6896; 6905. For example, for a plant with an SSE equal to 0.25g, OBE might be .10g; instead of .12g. That example would appear to have little bearing on this case — where the OBE is .20g and one-half the SSE is .375.

This argument is specious. As we have noted, Appendix A to 10 CFR Part 100 Section V(a)(2) states that the OBE "shall be at least one half ... " of the SSE. As we have seen, if the applicant wants any value of the OBE less than one-half of the SSE (how much less is immaterial), it must justify

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[396] *Id.* at § V(a)(2).

[397] Hoch, fol. Tr. 6879 at pp. 9-11. See also Allison, Tr. 8419-20, 8471-75; SER Supp. 7, pp. 2-4 and 2-5.

that value by the alternative allowed in 10 CFR Part 100, Appendix A, Section II.[398]

Turning to the specifics of the case before us, to fulfill the requirements of the alternate method of establishing the OBE value, the applicant performed a series of analyses

> which estimate exceedance probabilities and average return periods for various values of peak instrumental and peak effective acceleration at the Diablo Canyon site .... The analyses considered the factors of regional and local geology and specific characteristics of local subsurface material as required by Section III(d) of Appendix A to 10 CFR Part 100.[399]

The results of those analyses are in the record.[400] They establish that,

> [f]or a peak instrumental acceleration (maximum vibratory ground acceleration) at the site of 0.20g, the lowest average return period computed by and of the methods used in the analyses is 275 years. The corresponding exceedance probability for a 40 year plant lifetime is approximately 14.5 percent.[401]

This recurrence time is well beyond the staff minimum of 110 years.[402]

Dr. Trifunac (who was called by the Board at joint intervenors' request) also calculated a return interval for the OBE causing an acceleration of 0.2g. Depending on the model he used, Dr. Trifunac's preferred results showed a 30 percent probability (*i.e.*, a recurrence period of approximately 110 years),[403] that accelerations ranging from 0.18g to 0.24g would be exceeded at least once in a 40-year period.

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[398]It is public information that the NRC Safety Evaluation Reports (SER's) for some of the plants mentioned by Mr. Hoch give the following SSE and OBE values:

Plant	SSE	OBE
Byron	.20g	.08g
Braidwood	.20g	.09g
Marble Hill	.20g	.08g
Phipps Bend	.25g	.09g

[399]Hoch, fol. Tr. 6879 at p. 11.

[400]D-LL 11, D-LL 28, D-LL 41, and D-LL 45.

[401]Hoch, fol. Tr. 6879 at p. 11.

[402]*Id.* at pp. 11-12; Mr. Allison (who is the NRC project manager for Diablo Canyon) pointed out that Dr. Newmark did an independent study for the staff of the recurrence probability of the 0.2g seismic event at Diablo Canyon and agreed with the applicant's conclusions. Tr. 8423-24; see also SER Supp. 7, Ref. 4, App. D.

[403]Board Exh. 2F, M.D. Trifunac, "Diablo Canyon Consultants Report," June 8, 1978. See also, J.I. Br. at p. 75.

Dr. Trifunac was not questioned directly about his OBE calculations in this proceeding.[404] A staff witness, however, characterized Dr. Trifunac's position on the OBE as being essentially in agreement with the staff's, making the observation that (Tr. 8424):

> Dr. Trifunac took his probability study and answered the ACRS as to what he thought the return period was, and he felt it was within our criteria as well. So he was with a different number, but, you know, he's in agreement with our conclusion.

Not only was this unchallenged by joint intervenors, their counsel expressly agreed "that Dr. Trifunac endorsed the selection of the OBE" (Tr. 9235). In all the circumstances, we conclude that the OBE approved by the Licensing Board is both permissible under the regulations and sustained by the record.

Intervenors' brief calls for one further comment. It states at page 80 that "where a nuclear plant is located in an area of high seismicity, as is the case with DCNPP, an exemption to the regulatory requirements [such as those in 10 CFR Part 100, Appendix A, Section II] is not justified." The record, however, does not bear out the claim that the Diablo Canyon site is one of "high seismicity." The term refers to the frequency of seismic events. Drs. Anderson and Trifunac plotted for the years 1950 through 1974 the known epicenters in the region, centered around Diablo Canyon, between 33° and 37° north latitude and 119° to 123° west longitude.[405] That plot, and the calculated low recurrence rate of an earthquake of the magnitude assigned

[404]The applicant made several estimates of the 0.20g return period (D-LL 41, Table 41.5 at p. 41.22 gives 275 years, 600 years and 860 years). These are obviously far longer than Dr. Trifunac's. Applicant provided several critical analyses of Dr. Trifunac's return period estimates (D-LL 24) concluding generally that, while the methodology was correct, assumptions made along the way yielded return periods that were too short. Dr. Luco, citing Dr. Trifunac's work, was in turn critical of the applicant's analyses and found that their return periods too long (Board Exhibit 2C at pp. 8-12).

We have reviewed all of these papers as well as what the oral examination of the various witnesses brought forth. While we need not decide between the methods because both give return periods which meet the NRC's OBE criteria, we are inclined to believe Dr. Trifunac's predictions overly conservative, *i.e.,* they yield too short return periods. Our inclination stems from two principal factors. First, Dr. Trifunac assigns much of the higher magnitude activity in the region chosen for the data base to activity on the Hosgri Fault. His choice of the distribution of seismic activity does not appear consistent with the record of epicenters shown in Figures 1, 2 and 3 of his paper (Board Exhibit 2J). Second, Dr. Trifunac uses the Trifunac and Brady correlation to define motion attenuation relationships. We believe that the record before us shows that these predictions are likely to yield peak acceleration values that are too high in the near-field (see, *e.g.,* Blume IV-79 testimony, p. I-5 and Figure I-8) — the region that would be affected by activity on the Hosgri Fault.

[405]Board Exh. 2J, Fig. 2.

the OBE, indicate that the region is at most one of low to moderate seismicity. See also Smith, fol. Tr. 5490 at p. 14.[406]

X

JOINT INTERVENORS' MOTION TO REOPEN THE RECORD FOR A SECOND TIME

The joint intervenors have moved us to reopen the record again, this time to receive into evidence USGS Open-File Report No. 81-365 dated March 1981, entitled, "Peak Horizontal Acceleration and Velocity from Strong-Motion Records Including Records from the 1979 Imperial Valley, California, Earthquake." The Governor supports the motion and the applicant and staff oppose it. Those favoring admission of the new report urge that it supports the contention that 0.75g is not a conservative estimate of peak acceleration for the Hosgri earthquake; those objecting contend that it does no such thing.[407]

We have examined the USGS report with care. We note that, while its analysis is new, the seismic motion records underlying it are not. For the most part these either were or might have been addressed at the reopened hearing on IV-79.[408] Indeed, the authors of USGS Open-File Report 81-365 expressly acknowledge (at p. 19) that "[u]npublished strong motion data were generously supplied to us by [*inter alia*] J.N. Brune on behalf of the University of California at San Diego." Dr. Brune was a principal witness

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[406]Intervenors raised another matter with respect to the OBE. They questioned whether the seismic specification of non-safety related plant equipment to an event of 0.20g might result in equipment failures that could jeopardize the plant's safety-related systems (Joint Intervenors' Brief at p. 78). Challenges of this type — "systems interactions" — are in fact the subject of questions directed to the applicant by the staff as a follow-up of concerns generated by the Three Mile Island Unit 2 (TMI-2) accident. (See NUREG-0660).

In response to a query by this Board, counsel for the staff outlined the status of the review of this matter and also pointed out that it was before the Licensing Board as a part of its consideration of the panoply of post-TMI issues. NRC Staff Response to Board's Request for Information on Systems Interactions, dated December 12, 1980. In these circumstances, our consideration of the matter would be premature.

[407]The joint intervenors called this USGS Report to our attention by letter of April 13, 1981. In response to our order of April 15 calling for comments on the report, the intervenors formally moved on April 27 to reopen the record to receive it. In papers filed that same day, the Governor supported and the applicant and staff objected to that proposal.

[408]In addition to seismic records from IV-79 (for which we previously reopened the record and have already considered, see ALAB-598, *supra*, 11 NRC 876), the USGS Report also reviews strong motion records from earthquakes at Livermore Valley (January 27, 1980), Horse Canyon (February 25, 1980) and Coyote Lake (August 6, 1979). USGS Report at pp. 31, 35-37. The reopened hearing was held in October 1980 and we indicated specifically that we would consider all relevant evidence bearing on the reopened issues. See fn. 14, *supra*.

for the Governor and the intervenors in this case. Our point is not that the USGS Report is irrelevant. Rather, it is that the subject matter it addresses was thoroughly litigated before us, albeit on the basis of analyses supplied by other qualified experts.[409]

More important is the caveat in the USGS Report itself (at p. 15) that for "distances less than 40 km from earthquakes with M greater than 6.6 the prediction equations [in 81-365] are not constrained by data and the results should be treated with caution." Those distances and magnitudes, however, are precisely the ones that are important in this case.

Finally, we have thoroughly examined the evidence now before us bearing on the points covered by the new Open-File Report. Even were the caveat we mentioned not present, we are satisfied that the report itself is insufficient to overcome the result required by the record as we have discussed and evaluated it in this decision. In all the circumstances, and particularly as the new report would not affect the outcome of the case, the standards for reopening are not met. ALAB-598, *supra,* 11 NRC at 879; *Kansas Gas & Electric Co.* (Wolf Creek Station, Unit 1), ALAB-462, 7 NRC 320, 338 (1978). *See also, Bowman Transportation* v. *Arkansas-Best Freight System,* 419 U.S. 281, 294-96 (1974); *United States* v. *ICC, supra; ICC* v. *Jersey City, supra.* The motion to reopen accordingly must be denied.

XI

ENVIRONMENTAL MATTERS

The environmental consequences of Diablo Canyon were reviewed following the issuance of construction permits for the facility. LBP-74-60, 8 AEC 277 (1974), *affirmed,* ALAB-254, 8 AEC 1184 (1975). Additional contentions under the National Environmental Policy Act (NEPA)[410] were considered and decided by the Licensing Board in the operating license proceeding. LBP-78-19, 7 NRC 989 (1978). The Board there held that the mandates of NEPA as well as related environmental statutes and

[409]As the Supreme Court cogently explained in *ICC* v. *Jersey City,* 322 U.S. 503, 514 (1944): "Administrative consideration of evidence — particularly where the evidence is taken by an examiner, his report submitted to the parties, and a hearing held on their exceptions to it — always creates a gap between the time the record is closed and the time the administrative decision is promulgated. This is especially true if the issues are difficult, the evidence intricate, and the consideration of the case deliberate and careful. If upon the coming down of the order litigants might demand rehearings as a matter of law because some new circumstance has arisen, some new trend has been observed, or some new fact discovered, there would be little hope that the administrative process could ever be consummated in an order that would not be subject to reopening." *Accord, United States* v. *ICC,* 396 U.S. 491, 521 (1970).

[410]42 U.S.C. §§ 4321 *et seq.*

Commission regulations had been satisfied, and that, "after weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, the Board concludes that the final environmental balance weighs in favor of the licensing of Diablo Canyon, Units 1 and 2," subject to specified conditions for the protection of the environment. *Id.* at 1035. No party took exception to LBP-78-19 or the license conditions imposed by it, and the time to file exceptions has long since expired. This Board's customary review *sua sponte* has disclosed no error warranting corrective action; there is thus no occasion for us to disturb that decision.[411]

XII

CONCLUSION

For the reasons we have explained in this opinion, the exceptions to the Licensing Board's partial initial decision of September 27, 1979 (LBP-79-26) are *denied*. As is our practice, we have also examined on our own initiative the portions of that decision not excepted to[412] or which, though excepted to, were not briefed, and have similarly reviewed the Board's partial initial decision of June 12, 1978 (LBP-78-19) to which no exceptions at all were taken;[413] we found no error warranting corrective action. Accordingly, the Licensing Board's partial initial decisions rendered in this case on June 12, 1978 and September 27, 1979 (except for security plan issues)[414] are *affirmed*.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the Appeal Board

[411]We have similarly reviewed and reached the same conclusion regarding the portions of the Board's September 27, 1979 decision dealing with aircraft and missile accidents (10 NRC at 459-63); see fn. 2, *supra*.
[412]See 10 NRC at 459-63, and fn. 2, *supra*.
[413]See Part XI, *supra*.
[414]See fn. 2, *supra*.

ADDENDUM I - LIST OF WITNESSES

APPLICANT'S WITNESSES

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Blume, John A.

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	Present Occupation:	Professional Civil Engineer Senior Civil Engineer PG&E
Traisman, Steven E.		
	Education:	M.S. Engineering Mechanics, 1969 University of Wisconsin M.S. Environmental Engineering, 1974 Johns Hopkins University
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Education:	Ph.D. Civil Engineering (Earthquake Engineering) University of California at Berkeley
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JOINT INTERVENORS' WITNESSES

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GOVERNOR BROWN'S WITNESSES

Brune, James N.

Education:	Ph.D. Geophysics, 1961 Columbia University
Present Occupation:	Professor of Geophysics University of California San Diego, California

(Dr. Brune is listed under Joint Intervenors' Witnesses — Dr. Brune appeared for Governor Brown at the reopened hearing (October 20-25, 1981)

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Present Occupation:	Engineering Consultant Rolling Hills Estates, California

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Chen, Pei-Ying

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Present Occupation:	Leader of the Seismic Qualification Review Team Conducting Audit of Applicant's Seismic Qualifications for Seismic Category I Mechanical and Electrical Equipment U.S. NRC

Devine, James F.

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Rosa, Faust

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Present Occupation:	Chief, Power Systems Branch Division of Systems Safety U.S. NRC

Hall, William J.

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McKeown, Francis A.

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Present Occupation:	Geologist USGS Lakewood, Colorado

McMullen, Richard B.

Education:	B.S. Geology, 1959 University of Florida
Present Occupation:	Senior Geologist Division of Site Safety and Environmental Analysis U.S. NRC

Newmark, Nathan M.

Education:	M.S. Civil Engineering, 1932 Ph.D. Engineering, 1934 University of Illinois

It is with sincere sorrow that we note the death of Dr. Newmark a few weeks after he appeared as a consultant to, and witness for the NRC staff at our reopened hearing in October 1980.

Stepp, J. Carl

Education:	Ph.D. Geophysics (seismology), 1971 Pennsylvania State University
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Sullivan, Edmund J.

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Present Occupation:	Technical Assistant to Assistant Director for Engineering Division of Safety Systems U.S. NRC

ADDENDUM II - LIST OF EXHIBITS

Applicant

Exhibit No.		In Evidence
R-1	Smith Drawing - Fault Rupture	68
R-2	Map of the Fault Model for the 1966 Parkfield Earthquake	68
R-3	Imperial Valley 1979 Epicentral Distances	68
R-4	Imperial Valley 1979 Significant Distances	68
R-5	"Peak Acceleration From Strong-Motion Records: A Postscript" by Boore and Porcella	225
R-6	"Peak Horizontal Ground Motions From the 1979 Imperial Valley Earthquake: Comparison With Data From Previous Earthquakes" by Boore and Porcella	225
R-7	"Tectonic Stress and the Spectra of Seismic Shear Waves from Earthquakes" by James N. Brune	800
R-8	Blume Figure I-1 (modified by Young)	1349
R-9	"Evaluation of Seismic Criteria and Design Concepts for Point Conception LNG Import Terminal Environmental Impact Report" by Agbabian Associates, Dec. 1977	1073

Joint Intervenors

R-1	"Estimation of Ground Motion Parameters," USGS Circular 795	225
R-2	"Geophysical Assessment of Peak Accelerations" by Thomas C. Hanks and Dennis A. Johnson (J.I. Exh. 47 below)	
R-3	"Site-Dependent Spectra For Earthquake-Resistant Design" by H. Bolton Seed, *et al.*	
R-4	"Relationships of Maximum Acceleration, Maximum Velocity, Distance From Source, and Local Site Conditions for Moderately Strong Earthquakes" by H. Bolton Seed, *et al.*	245
R-5	Regression Analysis of the Peak Accelerations Recorded During the October 15, 1979 Imperial Valley Earthquake	1349
R-6	"Simulation of Earthquake Ground Motions for San Onofre Nuclear Generating Station Unit 1, Final Report," by Delmar Technical Associates, May 1978 (TERA Report)	1133[1]
R-7	Supplement I to J.I. Ex. R-6, July 1979	1133[1]
R-8	Supplement II to J.I. Ex. R-6, August 1980	1133[1]
R-9	Supplement III to J.I. Ex. R-6, August 1980	1133[1]

[1]Certain portions excluded. See Appeal Board Order dated November 5, 1980.

R-10	Map - Coastal and Offshore Geology Between Point Sal and Point Estero	
R-11	Parts 1 and 2 - Preliminary Map Showing Recency of Faulting in Coastal South-Central California by Jane M. Buchanan-Banks, *et al.* 1978	361
R-12	Trifunac Figure I.1 (modified)	882
R-13	Seed Figure I-2 (modified)	
R-14	Blume SAM IV and SAM V curves (modified)	
R-15	Trifunac curve (modified)	
R-16	Transcript p. 9333	
R-17	Dr. Luco's Review of J.I. Exh. R-9	1164[2]
R-18	Analysis of Soil-Structure Interaction Effects by H.R. Seed *et al.* (J.I. Exh. 58 below)	
R-19	Dr. Luco's Review of J.I. Exh. R-7	1164[3]
R-20	Figures from "Seismic Evaluation For Postulated 7.5M Hosgri Earthquake" (Hosgri Report, part of App. Exh. A below)	

NRC Staff

R-1	NUREG/CR 1665, "Equipment Response at the El Centro Steam Plant During the October 15, 1979 Imperial Valley Earthquake"	1342

[2]Certain portions excluded. See Appeal Board Order dated November 5, 1980.
[3]Except for portions referring to San Onofre Nuclear Generating Station.

Governor Brown

R-9	Response to Proposed Task 4, Earthquake Ground Motion Simulations For San Onofre, Unit 1 by Del Mar Technical Assoc., revised September 5, 1979	911[4]
R-10	Fault Dip Sensitivity Study - Frazier (3 Pages)	912
R-11	Imperial Valley Earthquake Computation	912
R-12	Blume Figure 1-1 (modified) by Young	1112
R-13	Figure A - Comparison of TERA 1971 San Fernando Earthquake Accelerations Plotted at Seed Fault Distances With Seed Attenuation Curves; Figure B - Comparison of TERA 1971 San Fernando Earthquake Accelerations Plotted at TERA Fault Distances with Seed Attenuation Curves	1112
R-14	Comparison of IV-79 and Newmark-Diablo Canyon Horizontal Spectra Both Scaled to 0.75 G Mean Peak Ground Acceleration	1112
R-15	Comparison of IV-79 Unscaled Vertical Spectrum With Newmark Diablo Canyon Spectra	1112

[4]Cover and p. 34 only.

Cite as 13 NRC 1024 (1981) **ALAB-645**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Thomas S. Moore

In the Matter of **Docket No. 50-219**

JERSEY CENTRAL POWER AND LIGHT COMPANY
(Oyster Creek Nuclear Generating Station) **June 22, 1981**

The Appeal Board affirms the Licensing Board's order terminating this proceeding to convert the facility's provisional license to a full-term operating license.

MEMORANDUM AND ORDER

This proceeding involves the conversion to a full-term operating license of the provisional license which was issued for the Oyster Creek nuclear facility in 1969. Last September, on its review *sua sponte* of a Licensing Board termination order, this Board remanded the proceeding to the Board below for further action. ALAB-612, 12 NRC 314. That action has now been taken and, in an unpublished June 1, 1981 order, the Licensing Board once again terminated the proceeding. In the absence of exceptions to the June 1 order, we have examined it on our own initiative. Finding no error requiring corrective action, we affirm.

1. The remand in ALAB-612, *supra,* was for the purpose of calling upon the NRC staff to supply "certain additional information respecting those unresolved generic safety issues as might be applicable to Oyster Creek operation". 12 NRC at 315. Following the receipt of that information, the Board below was to appraise "the nature and extent of the relationship between each significant unresolved generic safety question" and such operation. *Id.* at 315-16.

The reasons why this course was mandated were detailed in an order entered two days earlier in *Northern States Power Co.* (Monticello Nuclear Generating Plant, Unit 1), ALAB-611, 12 NRC 301 (1980). *Monticello* also was before this Board for a review *sua sponte* of the termination of a proceeding on an application for conversion of an operating license from provisional to full-term. We there decided to have the staff submit the required supplemental material pertaining to unresolved generic safety issues directly to us, rather than to the Licensing Board. We noted, however, that "in any other parallel proceedings" the Licensing Board would have the responsibility of receiving and evaluating the information. *Id.* at 311-12.

2. The *Monticello* proceeding came to an end in ALAB-620, 12 NRC 574 (1980). On the basis of our examination of the staff's submittal to us, we found that no reason existed for disturbing or probing further any of the determinations reflected in that submittal — determinations which had led the staff to the ultimate conclusion that continued operation of the Monticello facility would not present an undue risk to the public health and safety. In that connection, we stressed the restrictive scope of our review:

> [N]o endeavor has been made to satisfy ourselves that the staff's approach to each identified [unresolved generic safety issue] corresponds exactly with what we would have done if in the shoes of the Director of Nuclear Reactor Regulation. Rather, we have limited our consideration to the plausibility of the approach and sufficiency on their face of the explanations given for the conclusions reached by the staff respecting the continued safe operation of the Monticello facility.

As we saw it, the staff had both "satisfactorily * * * come to grips with the various unresolved generic problems it [had] indicated might affect Monticello operation" and "provided an at least reasonable foundation for its several conclusions". *Id.* at 577.

3. The Licensing Board order now at hand reaches the same result respecting the staff's October 30, 1980 submission to it on continued Oyster Creek operation.[1] Our independent examination of the record has given us no cause to view the matter differently.

In this regard, the Board below might have noted that the October 30 submission reveals that, unlike Monticello, Oyster Creek is one of the eleven operating reactors which are included within the Systematic

[1]Indeed, in large measure the text of that order bears a striking resemblance to ALAB-620. That fact has not influenced our review here. The Licensing Board order must stand or fall on the record underlying it — a record not identical to the *Monticello* record before us in ALAB-620.

Evaluation Program. That Program was instituted, following Commission approval, in November 1977. As stated in the 1978 NRC Annual Report (at p. 59), the Program staff is charged with the responsibility for reviewing those eleven "older licensed power reactors, applying current licensing criteria, and for documenting the results — including the need for any necessary plant changes".[2] It appears from a recently issued status summary report that unresolved generic safety issues are within the scope of the Program. See NUREG-0485, Vol. 3, No. 5 (April 1, 1981). Accordingly, the October 30 submission contains several references[3] to work being done under the Program on generic safety questions and the specific relationship of that work to the continued safe operation of Oyster Creek.

The June 1, 1981 order of the Licensing Board is *affirmed*.
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secreatry to the Appeal Board

[2]By virtue of Section 110 of the 1980 NRC Authorization Act (Pub. L. No. 96-295), this Commission must now develop and implement a similar plan for *all* currently operating plants. The staff has proposed integrating the existing Systematic Evaluation Program into the new plan (1980 NRC Annual Report, p. 5).
[3]See pp. 19, 22, 25, 32, and 34.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD[1]

Michael C. Farrar, Chairman
Richard S. Salzman

In the Matter of **Docket Nos. 50-348A 50-364A**

ALABAMA POWER COMPANY (Joseph M. Farley Nuclear Plant, Units 1 and 2) **June 30, 1981**

The Appeal Board (1) affirms with modifications the Licensing Board's initial decision, LBP-77-24, 5 NRC 804 (1977), which concluded pursuant to Section 105(c) of the Atomic Energy Act, that the unconditioned licensing of the Joseph M. Farley Nuclear Plant, Units 1 and 2, would create or maintain a situation inconsistent with the antitrust laws and their underlying policies, and (2) modifies the license conditions imposed by the Licensing Board in LBP-77-41, 5 NRC 1482 (1977), by ordering more extensive relief generally in the form of ownership access to the plant and greater access to the licensee's transmission facilities.

ATOMIC ENERGY ACT: ANTITRUST PROVISION

Under Section 105c of the Atomic Energy Act, the NRC must review applications for permits to construct commercial nuclear power facilities to determine if the activities sought to be licensed would create or maintain situations inconsistent with the antitrust laws or their underlying policies. Where such a result would follow, the Commission may refuse a license, rescind one previously issued, or attempt to rectify the anticompetitive consequences by attaching appropriate conditions to the license. *Consumers Power Company* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 897 (1977).

[1]The third member of this Board, Jerome E. Sharfman, resigned from the Appeal Panel after oral argument was held and did not participate in this decision.

NRC ANTITRUST REVIEW: APPLICATION OF ANTITRUST LAWS

Electric utilities are not immunized from the application of the antitrust laws because of being subject to state and federal regulation; even conduct formally approved by a regulatory agency may be the basis of an antitrust violation where agency approval conveys no exemption from the antitrust laws. *Consumers Power Company* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 1008, fn. 447 (1977).

NRC ANTITRUST REVIEW: SCOPE

In making the finding (in conjunction with its review of a license application for a nuclear power plant) required by Section 105c of the Atomic Energy Act "as to whether the activities under the license would create or maintain a situation inconsistent with the antitrust laws," the Commission must examine the applicant's activities prior to the grant of the license as well as assess the licensee's projected activities under the license. *Kansas Gas and Electric Co. et al.* (Wolf Creek Station, Unit No. 1), ALAB-279, 1 NRC 559, 567 (1975).

NRC ANTITRUST REVIEW: REMEDIAL AUTHORITY

The NRC's remedial authority under Section 105c of the Atomic Energy Act is not restricted to actual violations of the antitrust laws; actions which run counter to the policies underlying those laws, even where no actual violation of statute is made out, may also warrant remedial license conditions. *Consumers Power Co.* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 908-09 (1977).

NRC ANTITRUST REVIEW: RELEVANT MARKETS

In determining relevant markets, "markets which conform to areas of effective competition and to the realities of competitive practice" must be delineated. *L.G. Balfour Co.* v. *F.T.C.* 442 F.2d 1, 11 (7th Cir. 1971).

NRC ANTITRUST REVIEW: MONOPOLY POWER

The conduct of a dominant business enterprise wielding monopoly power over the entire range of activities in which it engages is judged under a harsher light than that of a less dominant business concern; actions undertaken by those with dominance in a market may not be acceptable even though they would be legitimate if undertaken by those less powerful. *United States* v. *Aluminum Co. of America*, 148 F.2d 416 (2nd Cir. 1945); *American Tobacco Co.* v. *United States*, 328 U.S. 781, 812-14 (1946); *Consumers Power Co.* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 913 (1977).

APPEAL BOARDS: STANDARD OF REVIEW

While appeal boards generally accord deference to trial board findings, they are not held to the "clearly erroneous" standard of review employed by the federal courts of appeal. Where an appeal board, upon review of the evidentiary record, is convinced that a different result is warranted, it is free to substitute its judgment for that of the trial board. *Consumers Power Co.* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 1022-23 (1977); *Duke Power Company* (Catawba Station, Units 1 and 2), ALAB-355, 4 NRC 397, 402-05 (1976).

NRC ANTITRUST REVIEW: NOERR-PENNINGTON DOCTRINE

Under the Noerr-Pennington doctrine, actions seeking to influence legislatures, courts, and other governmental bodies are immune from antitrust liability even when undertaken for anticompetitive purposes. *United Mine Workers of America* v. *Pennington*, 381 U.S. 657 (1965); *Eastern Railroad Presidents Conference* v. *Noerr Motor Freight, Inc.*, 365 U.S. 127 (1961). However, sham attempts to influence official action are not immune. *California Motor Transport Co.* v. *Trucking Unlimited*, 404 U.S. 508 (1972).

NRC ANTITRUST REVIEW: NOERR-PENNINGTON DOCTRINE

Evidence of activity protected under *Noerr-Pennington* may nevertheless be utilized to show purpose or character of other evidence under scrutiny. *United Mine Workers of America* v. *Pennington*, 381 U.S. 657, 670 n. 3 (1965).

NRC ANTITRUST REVIEW: MONOPOLY POWER

A business enterprise possessed of monopoly power is precluded by Section 2 of the Sherman Act from willfully using such power to preserve or extend its monopoly, to foreclose actual or potential competition, to gain competitive advantage or to destroy competitors.

NRC ANTITRUST REVIEW: REMEDIAL AUTHORITY

The Commission's remedial authority under Section 105c is not limited to the activities under the license; the Commission is authorized to place appropriate conditions on licenses where necessary to rectify anticompetitive situations. *Toledo Edison Co.* (Davis-Besse Units 1, 2 and 3), ALAB-560, 10 NRC 265, 291-92 (1979); *Consumers Power Co.* (Midland Units 1 and 2), ALAB-452, 6 NRC 892, 1094-1100.

NRC ANTITRUST REVIEW: REMEDIAL AUTHORITY

In determining the appropriate relief under Section 105c, the Commission's action should harmonize both antitrust and public interest considerations. Except in an extraordinary situation, Commission-imposed conditions should be able to eliminate the concerns entailed in any affirmative finding under that Section.

NRC ANTITRUST REVIEW: REMEDIAL AUTHORITY

The fact that a transgressor has ceased its anticompetitive activity, especially when such cessation occurs after the onset of legal action, in and of itself provides no justification for dispensing with otherwise appropriate remedial requirements. *United States* v. *Oregon State Medical Society,* 343 U.S. 326, 333 (1952).

APPEARANCES

Messrs S. Eason Balch, Sr., Birmingham, Alabama, and **Terence H. Benbow**, New York, New York, argued the cause for the applicant, Alabama Power Company; with them on the briefs were **Messrs. Robert A. Buettner** and **Joseph W. Blackburn**, Birmingham, Alabama, and **Theodore M. Weitz** and **David J. Long**, New York, New York.

Mr. D. Biard MacGuineas, Washington, D.C., argued the cause for intervenor, Alabama Electric Cooperative; with him on the briefs were **Messrs. Bennett Boskey, James C. Hair, Jr.**, and **Edwin E. Huddleson, III**, Washington, D.C.

Mr. David C. Hjelmfelt, Washington, D.C., argued the cause for intervenor, Municipal Electric Utility Association of Alabama; with him on the briefs were **Messrs. Reuben Goldberg, Glenn W. Letham** and **Michael D. Oldak**, Washington, D.C., and **Maurice F. Bishop**, Birmingham, Alabama.

Mr. John D. Whitler argued the cause for the Attorney General of the United States; with him on the briefs were **Assistant Attorney General John H. Shenefield, Deputy Assistant Attorney General Joe Sims**, and **Messrs. Donald L. Flexner, Joseph J. Saunders, C. Kent Hatfield** and **David A. Leckie.**

Ms. Jane A. Axelrad argued the cause for the Nuclear Regulatory Commission staff; with her on the briefs were **Messrs. Joseph Rutberg** and **Michael B. Blume.**

DECISION

Opinion of the Board by Mr. Farrar:

This is the third antitrust case arising under Section 105c of the Atomic Energy Act[2] to reach us on the merits. The first, *Midland*, involved a nuclear plant being constructed by Consumers Power Company, which serves most of Michigan's lower peninsula. ALAB-452, 6 NRC 892 (1977).[3] The second, *Davis-Besse*, dealt with a number of reactors proposed for construction in Ohio and western Pennsylvania by several utility companies serving the City of Cleveland and the rest of the "CAPCO" territory. ALAB-560, 10 NRC 265 (1979).[4] Unfortunately, our rulings in both *Midland* and *Davis-Besse* did not come down until after the Licensing Board's two-step decision in the matter now before us.[5] Necessarily, then, that Board's opinions, in general carefully and thoughtfully crafted, were written before it had the benefit of any appellate guidance.[6]

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[2]42 U.S.C. §2135(c).

[3]*Reversing and remanding Consumers Power Co.* (Midland Units 1 and 2), LBP-75-39, 2 NRC 29 (1975).

[4]*Affirming as modified Toledo Edison Co.* (Davis-Besse Units 1, 2 and 3), LBP-77-1, 5 NRC 133 (1977).

[5]The first of the Board's decisions (Phase I) dealt with what might be called the question of "liability" (LBP-77-24, 5 NRC 804 (April 8, 1977)); Phase II addressed the matter of remedies (LBP-77-41, 5 NRC 1482 (June 24, 1977)).

[6]As already indicated, at that point our *Midland* and *Davis-Besse* decisions had not been written. And, to this day, neither the Commission itself nor the courts have spoken about the merits of an NRC antitrust case: (1) Any need for further review of *Midland* was eliminated when the parties reached a settlement while the case was on remand below. That settlement was approved by the Licensing Board last August (LBP-80-21, 12 NRC 177); because the parties were in agreement, we declined to review the matter (ALAB-610, 12 NRC 174 (August 26, 1980)). (2) In *Davis-Besse*, on the other hand, the Commission declined applicants' request that it review our decision. The case was then appealed to the United States Court of Appeals for the Third Circuit under the name *Duquesne Light Co.* v. *NRC*; the applicants later withdrew their appeal and the case was dismissed on October 8, 1980.

In those opinions, the Board below ruled that Alabama Power Company's construction and operation of the two-unit Farley nuclear power plant would create and maintain "a situation inconsistent with the antitrust laws" within the meaning of the statute unless certain remedial conditions — including access for one of the intervenors by way of purchases of "unit power"[7] — were included in the nuclear licenses. No stay having been sought, the conditions imposed have been in force while the parties' cross-appeals have been pending before us.[8]

Alabama Power tells us in its appeal that none of its past conduct warranted the finding of antitrust "liability"[9] and that, in any event, the remedy selected was too drastic. Its opponents — the Alabama Electric Cooperative (AEC), the Municipal Electric Utility Association of Alabama (MEUA), the United States Department of Justice, and the NRC staff — take the opposite tack. Their appeals argue that the applicant's past conduct was more egregious than the Board found and that a more sweeping remedy is in order.[10]

As we explain in this opinion, we find the Licensing Board's rulings not fully in accord with the principles laid out in decisions issued by us since then. In terms of the positions taken by the parties here, the upshot is that Alabama Power's opponents are entitled to a somewhat more favorable result than they obtained below. Specifically, we find that AEC should be afforded ownership access to the Farley units and that, while applicant need not extend such access to MEUA, the municipals are entitled to access to applicant's transmission system.

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[7]The Board below defined unit power as "power purchased on a contractual basis in the form of a percentage share of the output from a particular power plant. The cost of unit power includes the owner's cost of capital, costs of construction, cost of fuel and operation, and a rate of return on investment." 5 NRC at 1502.

[8]Unit 1 began commercial operation on December 1, 1977; Unit 2 recently received its operating license.

[9]That is, the finding that its activities under an unconditioned license to operate the Farley plant would maintain a situation inconsistent with the antitrust laws specified in Section 105 of the Atomic Energy Act.

[10]This capsule description of the parties' appellate positions is intended only to set the stage; it does not, of course, even begin to hint at the precise nature of the questions presented in the 1,000 pages of briefs filed with us. In that connection, the record below consisted, *inter alia*, of nearly 30,000 pages of transcribed testimony.

I.

BACKGROUND AND SUMMARY

By amending the Atomic Energy Act in 1970, Congress gave this Commission added duties to fulfill in connection with its licensing of nuclear power plants. Since that time, it has had to consider, in addition to safety and environmental matters, the antitrust ramifications of its licensing actions.[11] Specifically, as we said in *Midland* (6 NRC at 897, footnotes omitted):

> Under Section 105c of the Atomic Energy Act, it must review applications for permits to construct commercial nuclear power facilities to determine if the activities sought to be licensed would create or maintain situations inconsistent with the antitrust laws or their underlying policies. Where such a result would follow, the Commission may refuse a license (or rescind one previously issued) or attempt to rectify the anticompetitive consequences by attaching appropriate conditions to the license. As the Commission has reiterated, the Atomic Energy Act's antitrust provisions reflect "a basic Congressional concern over access to power produced by nuclear facilities" and represent legislative recognition "that the nuclear industry originated as a Government monopoly and is in great measure the product of public funds [which] should not be permitted to develop into a private monopoly via the [NRC] licensing process . . . ."

The governing statute provides the procedures by which this review is to be accomplished; we have described its workings elsewhere.[12] Here, the

[11]The Commission's responsibilities in the antitrust sphere prior to 1970 were less definitive. See *Cities of Statesville* v. *AEC*, 441 F.2d 962 (D.C. Cir., *in banc*, 1969) and the history recited in *Toledo Edison Co.* (Davis-Besse Unit 1), ALAB-323, 3 NRC 331, 337-40 (1976).

[12]*Kansas Gas and Electric Co.* (Wolf Creek Unit 1), ALAB-279, 1 NRC 559 (1975).

review was duly initiated when the Commission referred Alabama Power's construction permit application to the Attorney General of the United States for his advice concerning its potential antitrust consequences. The Department of Justice's analysis led it to respond that the plant should not receive an unconditional license and that an antitrust hearing should be held. In that connection, petitions to intervene filed by AEC and MEUA were granted by the Licensing Board (over the applicant's opposition). The entry of these two organizations alongside the statutory parties — the Commission staff and the Attorney General — completed the lineup of paticipants opposed to the award of an unconditional license to Alabama Power.

For introductory purposes, the business operations of the utility parties to the proceeding can be simply described.[13] The applicant, Alabama Power, is a wholly-owned subsidiary of the Southern Company, a public utility holding company which also owns Georgia Power Conpany, Gulf Power Company,[14] and Mississippi Power Company, all of which function under an interchange contract as the Southern Company Pool. Alabama Power generates, transmits and distributes electricity in central and southern Alabama (the eleven most northern counties in the State are served primarily by the Tennessee Valley Authority).[15] At retail, it has residential, commercial and industrial customers; it wholesales electricity to sixteen municipalities with their own distribution systems (twelve of which comprise the membership of the intervenor MEUA), to eleven rural distribution cooperatives,[16] and to the other intervenor, the Alabama Electric Cooperative. The AEC, in turn, is a generation and transmission cooperative whose membership is made up of four municipalities,[17] two industrial mills, and fourteen rural cooperatives.[18]

In terms of generating facilities, the applicant had in operation at the

[13]The Licensing Board's first decision contains a more complete description of the parties' operations as well as of those of other entities in the surrounding area. See 5 NRC at 820-33.

[14]Gulf Power operates in the Florida panhandle.

[15]Southern's operating companies thus embrace a contiguous area covering not only the Florida panhandle and much of Alabama but also southeastern Mississippi and most of Georgia. See D.J. Ex. 1008.

[16]Ten of these are members of the Alabama Electric Cooperative. See fn. 18, *infra.*

[17]There are a total of 22 municipally-owned systems in the geographic area of interest — the twelve in MEUA, the four in AEC, four others supplied at wholesale by Alabama Power but not affiliated with either intervening organization, and two that purchase their power requirements from TVA. The Licensing Board lists the town of Robertsdale, one of the unaffiliated municipal systems, as purchasing wholesale power from Riviera Utilities (see 5 NRC at 828); the town now gets its power from applicant. MEUA Brief, 25; APCO Reply Brief, 46-47.

time of trial thirteen hydroelectric plants and eight fossil-fueled plants, totalling over 6,000 megawatts in capacity.[19] By comparison, the AEC had two hydro and six fossil plants totalling 137 megawatts. The MEUA's members had no generating capacity.

We need not pause here to describe how the electric utility industry generally functions, in Alabama and elsewhere, to produce a reliable electric power supply. We went into that subject in detail in *Midland*,[20] and the Board below — after finding that "the principles of electric power supply production and coordination are generally applicable throughout the electric utility industry" and "do not vary significantly among electric utilities regardless of differences in locations * * * ' " — covered the subject quite thoroughly itself here. 5 NRC at 833-37.

The Licensing Board had to deal with numerous claims made by the applicant's opponents concerning alleged anticompetitive practices it was said to have engaged in through the years. In order to evaluate those claims in context, the Board first undertook to determine what product and geographic markets were relevant. It concluded that the applicant's service area constituted the relevant geographic market; the only product market it held relevant was that for wholesale power. In this regard, the Board rejected the notion that there was a market in either of the other suggested products — *i.e.*, retail power or coordination services.[21] 5 NRC at 879-894.

Using its findings delineating the relevant market as a touchstone, the Board found that the applicant possessed monopoly power in that market (5 NRC at 896-901); it then reviewed the evidence bearing on the applicant's alleged anticompetitive practices (5 NRC at 901-957). In all instances but five, the Board exonerated the applicant. With respect to those five transactions, however, it found the applicant's conduct to have been anticompetitive in nature and to have resulted in a situation inconsistent with the antitrust laws. The upshot was the conclusion that the

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[Throughout this decision, "__________ Brief" refers to the appellate briefs filed by the parties on November 14, 1977; "__________ Reply Brief" refers to the responses filed on April 14, 1978. The parties will be referred to in such citations as APCO, AEC, MEUA, Justice, and Staff.]

[18]AEC supplies all the power requirements of its municipal and industrial members and three of the rural co-ops, as well as some of the needs of five other co-ops (who are also customers of Alabama Power); these constitute AEC's "on-system" members. It has no direct physical access to five co-ops in Alabama (who receive all their power from the applicant) and to one in Florida (served by Gulf Power). These six are called its "off-system" members.

[19]Of the eight fossil-fueled plants, applicant owns six of them outright, and shares in the ownership and output of the two others. The capacity figure shown includes only applicant's portions of the two shared facilities See 5 NRC at 821-22.

[20]See particularly 6 NRC at 950-57.

[21]Based largely on its rejection of the retail power market, the Board concluded that MEUA was not entitled to any access to the Farley units. See 5 NRC at 961.

activities under the nuclear licenses would maintain that situation (5 NRC at 957-961).

In other words, the Board held that the nuclear licenses had to be conditioned to ameliorate the effects of the anticompetitive situation then existing. The hearing then moved into its second phase, having to do with the appropriate remedy. The Board heard additional evidence on that score (but did not allow MEUA to participate[22]) and then rendered its second and final decision. It imposed a number of conditions upon the license, but rejected others which the applicant's opponents believed were necessary. In terms of access to the nuclear facility itself, the Board held that allowing AEC to purchase unit power was sufficient and that no ownership participation was called for.

As already indicated, all parties appealed. Among them, they manage to challenge — from both sides — nearly every significant holding made by the Board below.[23]

In deciding the matter, we take up first — and reject — certain broad arguments the applicant makes that, if accepted, would largely insulate its actions from antitrust scrutiny (Part II). In Part III, we then consider the questions raised as to the nature of the relevant markets. Although we are in total agreement with the Board below on its determination of the market for firm wholesale power, the principles we set out in *Midland* and *Davis-Besse* — both handed down after the decision below — lead us to disagree with the Licensing Board's rejection of the proposed markets for coordination services and retail power.

We proceed in Part IV to hold that the applicant has monopoly power in these other markets as well as in the wholesale market. We turn then to that aspect of the appeals which gives us the most difficulty: to what extent the applicant has used its monopoly power in violation of the antitrust laws or their underlying policies. The Licensing Board found it had done so only in certain respects; we believe that in reaching that conclusion it cast the applicant's activities in too favorable a light. With respect to MEUA, we also had to reassess the findings below in light of our holding expanding the relevant markets in the case. The additional violations we perceive and our findings relating to MEUA are discussed in Part V. Finally, we turn in Part VI to the question of what remedies are appropriate in light of our additional findings on "liability" together with those violations already perceived by the Board below.

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[22]See 5 NRC at 1484 n. 5.

[23]As previously intimated (see fn. 17, *supra*), all parties filed concurrent briefs as appellants on November 14, 1977. Before their responsive briefs were due, we handed down *Midland*. The time for filing the second set of briefs was then extended to allow the parties to adjust their thinking to take *Midland* into account. Oral argument was held on March 8, 1979.

II.

APPLICANT'S ARGUMENTS AGAINST ANTITRUST SCRUTINY

The applicant raised three broad arguments against antitrust scrutiny. First, it argues that there is no room here for any finding of "liability" because it is so "pervasively regulated" that it cannot be held to possess monopoly power in the relevant market. It next contends that Section 105c of the Atomic Energy Act forbids a broad inquiry into its past activities for findings of liability — that any remedial action taken against it must be based solely on its predicted or potential future activities. Finally, it argues that the Licensing Board was wrong in basing its findings of liability on "anticompetitive conduct." According to the applicant, Section 105c requires that actual violations of the antitrust laws or the clear policy underlying them be found. We deal with these arguments in order.

A. Pervasive Regulation

As noted by the Licensing Board,[24] this proceeding arises under Section 105c of the Atomic Energy Act, which requires the Commission to determine in connection with its licensing of the Farley plant "whether the activities under the license would create or maintain a situation inconsistent with the antitrust laws as specified in subsection 105c." The specified antitrust laws are the Sherman Act,[25] Wilson Tariff Act,[26] Clayton Act,[27] and the Federal Trade Commission Act,[28] For the purpose of making the required finding, the Licensing Board conducted an inquiry into the applicant's activities. Measuring these activities principally against three of the specified antitrust laws — the Sherman, Clayton and the Federal Trade Commission Acts — and the policies underlying them, the Board found that in five instances the activities engaged in by the applicant came within the proscription of those laws and their policies. In reaching these conclusions, the Board first conducted a market analysis (applying recognized antitrust principles) and found that a market for wholesale power existed in the applicant's area of operations. Proceeding further, it then found that the applicant enjoyed monopoly power in that market.

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[24] 5 NRC at 812.
[25] 15 U.S.C. §§ 1-7.
[26] 15 U.S.C. §§ 8-11.
[27] 15 U.S.C. §§ 12-27, 44; 18 U.S.C. § 402; 29 U.S.C. §§ 52-53.
[28] 15 U.S.C. §§ 41-49.

The applicant vigorously objects to the finding that it possesses monopoly power in the relevant market. In the portion of its brief devoted to this issue,[29] applicant argues that to have monopoly power it must first be shown that it has the power to control prices or to exclude competitors from the relevant market. Detailing the extent to which it purportedly is regulated, it insists that this "pervasive regulation" by the state and federal governments precludes it from having either of the necessary powers.[30]

Applicant's contention is not new. We find that it merely attempts to put in different clothing a time-worn and discredited argument that seeks to justify immunity from the antitrust laws. It is too late in the day for the argument that state and federal regulation — even with respect to electric utilities — bring with them a form of dispensation from the antitrust laws. If any earlier doubt existed on this score, it was put to rest by the Supreme Court several years ago. As observed by the Court of Appeals for the Seventh Circuit in *City of Mishawaka, Ind.* v. *Indiana & Michigan Electric Co. (Mishawaka I,)*[31] citing *Cantor* v. *Detroit Edison Co.,*[32] it is a "now settled axiom that after *Otter Tail Power Co.* v. *United States,* 410 U.S. 366, 93 S. Ct. 1022, 35 L. Ed. 2d 359, 'there can be no doubt about the proposition that the federal antitrust laws are applicable to electric utilities.' "

In recognition of this proposition, the applicant urges that it is not arguing for immunity from the antitrust laws.[33] Rather, as we understand it,

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[29]APCO Brief, 5-13.

[30]In applicant's words: "Applicant will demonstrate that state and federal regulation to a substantial degree control all aspects of Applicant's growth and development, its marketing practices, its operations, and its wholesale and retail rates. The existence of this regulation negates the inference of the Board that Applicant possesses either the power to control prices or exclude competitors." *Id.* at 2. According to the applicant, the activities which are regulated include: rates and charges, finance, entry into service area, withdrawal from service and abandonment of facilities, acquisition, merger and consolidation, system extensions, transmission and interconnections, coordination reliability and quantity of service, arrangements with service organization and suppliers, accounting, and competition. *Id.* at 5-13.

[31]560 F.2d, 1314, 1321 (1977), *cert. denied,* 436 U.S. 922 (1978).

[32]428 U.S. 579, 596 n. 35 (1976).

[33]At oral argument before us, applicant's counsel was asked whether the applicant's assertion that the Alabama Public Service Commission considered anticompetitive matters in dealing with matters before it insulated the applicant from antitrust liability. Mr. Balch, applicant's counsel, answered as follows:

> "I don't believe we are contending that Applicant is immune from anti-trust liability. If the board has the impression that we are considering that, I would like to state here and now we are not contending that."

App. Tr. 21-22. ["App. Tr." refers to the transcript of the oral argument held before us on March 8, 1979; "Tr." refers to the transcript below.]

the applicant is relying upon a facially different argument: that it cannot be found to possess monopoly power. In the words of its counsel:

> I am suggesting that if there is a federal agency or a state agency which has the ultimate control over prices, that Alabama Power Company cannot, as a matter of definition, have the power to control its prices.[34]

This formulation of applicant's argument does not aid its case. In *Midland*, we were confronted with essentially the same argument and found ourselves compelled to reject it. The applicant for a nuclear power license there, like the applicant here, was seeking to avoid antitrust scrutiny of its activities. One of the bases on which it attempted to do so was the regulation to which some of its activities were subjected under the Federal Power Act. Rather than claiming immunity from the antitrust laws because of this regulation, it had argued that because the Federal Power Commission[35] might order it to interconnect with other utilities, the company *ipso facto* lacked monopoly power. To that we responded:

> We fail to perceive how a regulatory scheme that admittedly grants no immunity from the antitrust laws, by its mere existence, alters the character of what is otherwise monopoly power. Consumers' argument is an attempt to slip in via the back door a proposition the courts have barred at the front, namely, that regulation for other purposes can attenuate the antitrust laws. That argument has been rejected. *Mt. Hood Stages, Inc.* v. *Greyhound Corp.* 555 F.2d 687, 691-92 (9th Cir. 1977); *International T. & T Corp.* v. *General T. & E. Corp.*, 518 F.2d 913, 935-36 (9th Cir. 1975), and cases cited. The best that can be said for it is that "the impact of regulation must be assessed simply as another fact of market life." *Id.* at 936.

6 NRC at 1008.[36]

We know of no reason why that same response is not dispositive of the applicant's "pervasive regulation" argument here.[37] To be sure, the

[34]App. Tr. 34.

[35]Now the Federal Energy Regulatory Commission (FERC).

[36]Moreover, as noted in the margin of our *Midland* decision, "it is settled that even conduct formally approved by a regulatory agency may be the basis of an antitrust violation where agency approval conveys no exemption from the antitrust laws. *United States* v. *Radio Corp. of America*, *supra*, 358 U.S. at 350-51; *Cantor* v. *Detroit Edison Co.*, *supra*, 428 U.S. at 596-98; *California* v. *FPC*, 369 U.S. 482, 489 (1967); *United States* v. *Philadelphia Bank*, *supra*, 374 U.S. at 350-52; *Litton Systems, Inc.* v. *Southwestern Bell Tel. Co.*, 539 F.2d 418, 422-24 (5th Cir. 1976); *City of Mishawaka* v. *Indiana and Michigan Electric Co.*, *supra*; *Almeda Mall, Inc.* v. *Houston Power and Light Co.*, *supra*, Trade Reg. Rep. par. 61,485 (S.D. Tex. 1977)." 6 NRC at 1008 fn. 447.

[37]In conjunction with its "pervasive regulation" argument, the applicant stresses that "the

argument in *Midland* was made in terms of the Federal Power Commission, while the asserted justification here is the increased restriction on the activities of applicant as a result of both state and federal regulation. But we see no significant difference in the two situations. What the argument boils down to in either case is that government regulation somehow serves to relieve the activities from close scrutiny under the antitrust laws. The law on this point is well-settled against the applicant's position. As *Midland* makes clear, the applicant's claim of the impact "pervasive regulation" has on its activities is simply another factor which must be assessed in examining applicant's activities for conformance to the antitrust laws.[38]

B. Scope of Inquiry

We turn now to the applicant's second broad argument against granting any antitrust relief. Specifically, it would have us set aside the Licensing Board's findings of liability — which formed the basis for that Board's remedial action — as founded upon a number of critical errors. Applicant's point seems to be that the Board roamed so far afield and delved so deeply in conducting its inquiry into applicant's activities that it went beyond the permissible reaches of Section 105c of the Act. According to this argument, the Act allows inquiry only into activities likely to occur in the period after the license is issued and not (as was done here) into the applicant's past activities.

The applicant argues that a rule barring consideration of past activities is compelled by the narrow scope of Section 105c inquiry intended by the Joint Committee on Atomic Energy. Alluding to the Joint Committee's statement that the licensing process should be used to "nip in the bud any

electric utility industry, in its historical development, has been recognized as a natural monopoly." APCO Brief, 19. Without ruling on the validity of the applicant's statement, we fail to see how a natural monopoly status aids the applicant's central argument that it cannot be found to possess monopoly power because the power to set prices or exclude competitors lies elsewhere, in the state and federal regulatory agencies. By definition, a natural monopolist has the power to exercise requisite control over prices or potential competitors. If anything, the applicant's argument on this score is self-defeating.

[38] *Accord, Davis-Besse, supra,* ALAB-560, 10 NRC at 282-86.

Brief mention should be made here of the Public Utility Regulatory Policies Act of 1978 (PURPA) (Pub. L. No. 95-617, 92 Stat. 3117). Counsel for applicant sought to inject PURPA into the proceeding at the oral argument before us (App. Tr. 242-45, 256); we declined to consider the Act at that time but invited applicant to submit a written memorandum on its importance to the case. Applicant sent us a memorandum on March 16, 1979; all the other parties submitted responses. According to the applicant, the existence of PURPA should have a "substantial impact on this Board's deliberations," including our decision on the existence of monopoly power. APCO Memorandum, 4. We think otherwise. We have carefully reviewed all the submitted materials; we are in complete agreement with the basic position of the applicant's opponents on this point. Nothing in PURPA causes us to change our findings on monopoly power, applicant's past conduct, or the appropriate remedies in this case.

incipient antitrust situation," the applicant contends that this "clearly focuses on future, not past, activities."[39] In this same vein, the applicant intimates that this is what the Joint Committee intended when it "made it clear that the standard it was expecting a board to apply was that 'it is *reasonably probable* that the activities *under the license* would, *when the license is issued or thereafter,* be inconsistent with any of the antitrust laws or the policy clearly underlying these laws.' " (Emphasis supplied by the applicant.)[40]

In our judgment, the applicant has misapprehended the thrust of the Joint Committee's statements. It derives from them an intent which does not give consideration to the statements in their entirety; nor does it give recognition to the words of the statute to which the statements relate. Properly considered, the statute could not reasonably support the position the applicant advocates.

As already seen, Section 105c requires the Commission, in conjunction with its review of a license application for a nuclear power plant, to "make a finding as to whether the activities under the license would create or maintain a situation inconsistent with the antitrust laws." It is significant that Section 105c is concerned with both a situation which would be *created* when the license issued and a situation which would be *maintained* by the license issuance. Although this latter finding does require an assessment of the future, it equally clearly requires a review of the situation which preceded the license. In other words, as we held in *Wolf Creek,*[41] a determination of the antitrust effects of granting a license can be made only after the situation leading up to the grant has been ascertained.

Read with these words and meaning of Section 105c in mind, the statements of the Joint Committee take on a far different hue than that painted by the applicant. The Joint Committee's statement that the licensing process should be used to "nip in the bud any incipient antitrust situation" can thus be seen as a endeavor to explain Section 105c's injunction against the use of a nuclear license to "create" a situation inconsistent with the antitrust laws, and not, as the applicant insists, as a limitation on the scope and level of antitrust inquiry.[42] Similarly, the Joint Committee's statement that a "reasonably probable" standard shall apply in making the antitrust determination called for by Section 105c, deals with

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[39]APCO Brief, 44.

[40]*Ibid.*

[41]*Kansas Gas and Electric Co. et al.* (Wolf Creek Station Unit No. 1), ALAB-279, 1 NRC 559, 567 (1975).

[42]*Id.,* 1 NRC at 572-73.

the degree of probability which governs that determination.[43] Neither th Joint Committee's words nor any reasonable inferences from their contex fairly support the applicant's suggestion that there exists a ban agains looking other than forward at the applicant's projected activities under th license. Indeed, both the statute and the Joint Committee's statement strongly suggest otherwise. As we recognized in *Wolf Creek*, thei requirement of Commission assessment of the antitrust implications o future activities of the applicant cannot be made *in vacuo*.[44] Here, a elsewhere, the past is prologue. Past conduct, good or bad, often indicate what future conduct might be. This was recognized by no less than th Supreme Court when it warned that "size carries with it an opportunity fo abuse that is not to be ignored when the opportunity is proved to have bee utilized in the past."[45] This indicates that a meaningful assessment of th issue before us — *i.e.*, whether issuance of a license for construction an operation of a nuclear power plant would create or *maintain* a situatio inconsistent with the antitrust laws — cannot be made without firs considering the current and past activities of the license applicant. We hav little hesitance in construing Section 105c as permitting inquiry into the pas activities of the applicant; indeed, the statute and Commission decision require it. *Wolf Creek*, *supra*, 1 NRC at 573 and authorities there cited

C. Standard for Finding of Liability

Applicant's third broad argument concerns the standard utilized by th Licensing Board in arriving at its finding on monopolization. As w understand its position, the applicant seems to advance three grounds fo faulting the way in which the Board reached its findings. First, it says tha "the Board concluded that it need not find a violation of the antitrust law but could be satisfied with a showing of 'anticompetitive' conduct whic need not have been bottomed on a specific violation.' "[46] It next states tha the Board considered not only "anticompetitive" conduct but conduc which "tended" to be anticompetitive.[47] It then argues that in proceedin on these premises the Board failed to base its conclusions on the antitrus laws.[48] In short, the applicant seems to be arguing that (assuming it i wrong in its position that consideration of past activities is barred) unde Section 105c all that is cognizable are actual violations of the antitrust law

[43] *Midland*, *supra*, 6 NRC at 927 (quoting the Joint Committee Report); *Wolf Creek*, *supra*, NRC at 569-70.
[44] *Wolf Creek*, *supra*, 1 NRC at 572-73.
[45] *United States* v. *Swift & Co.*, 286 U.S. 106, 116 (1932) (Cardozo, J.).
[46] APCO Brief, 44.
[47] *Ibid.*
[48] *Id.* at 47.

As we understand applicant's argument, it believes this standard was contemplated when "the Joint Committee made it clear that the standard it was expecting a board to apply was that 'it is reasonably probable that the activities under the license would, when the license is issued or thereafter, be inconsistent with any of the antitrust laws or the policy clearly underlying these laws.' "[49]

We find this argument without merit. In *Midland*, we addressed the question, *inter alia*, of whether finding a "situation inconsistent with the antitrust laws" necessarily depended upon a finding of actual violations of those laws.[50] We there ruled that Section 105c was not restricted to actual violations:

> The Licensing Board was correct in holding that proof of an actual violation of the antitrust laws is not required to show the existence of a situation "inconsistent with" them for Section 105c purposes. The Congressional framers of the section (the members of the Joint Congressional Committee on Atomic Energy) were originally divided between those who favored proof of an antitrust violation before allowing Section 105c remedies to be imposed and those who thought a showing of circumstances merely "tending" to such a violation should suffice to allow that relief. An accommodation between the two views was eventually reached. The members of the Joint Committee agreed that proof of conditions which ran counter to the *policies* (underlying those laws, even where no actual violation of statute was made out, would warrant remedial license conditions under Section 105c. We need not linger over the matter; this compromise is expressly manifested in the report of the Joint Committee and is reflected in the Commission's decisions.[51]

These observations apply to applicant's argument here as well. In this respect, we find no evidence to support applicant's charge that the Licensing Board considered conduct which "tended to be anticompetitive" in making its five findings of monopolization. Our analysis of the Licensing Board's decision reveals that each of its findings of monopolization was made on the basis that the acts in question were "anticompetitive."

Finally, we turn again to *Midland* for the answer to the applicant's argument that the Licensing Board erroneously based its findings on mere anticompetitive conduct. The Licensing Board there had reasoned that a "situation inconsistent with the antitrust laws" within the meaning of

[49]*Id.* at 44 (emphasis deleted).
[50]See 6 NRC at 907-14
[51]*Id.*, 6 NRC at 908-09 (footnotes omitted). *Accord*, *Wolf Creek*, *supra*, 1 NRC at 570.

Section 105c amounts to "anticompetitive conduct." The Department of Justice criticized that analysis, claiming that a focus solely upon conduct without consideration of market structure would ignore essential elements in such a situation. We rejected the Department's argument:

> We do not agree that the Licensing Board's determination to concentrate on the applicant's conduct necessarily caused it to go astray in the manner suggested by the Department. What an inquiry is labelled is of lesser moment than how it is carried out. In our judgment, evaluation of business "conduct" in a case like this one, exploring charges essentially bottomed on Section 2 of the Sherman Act and its underlying policies, requires the application of the same monopolization and policy concepts as an investigation of an anticompetitive "situation." This is so because, as with other statutes, actions permissible under the antitrust laws in one situation may be proscribed in another. An antitrust analysis of an applicant's conduct must therefore be undertaken in the context of the "situation" in which that conduct occurred — in other words, against the background structure of the relevant market. Of course that analysis of a utility's conduct must (among other things) be sensitive to judicial and FTC antitrust rulings that the actions of a dominant business enterprise have to be tested against a more stringent standard than applies to actions of smaller concerns in highly competitive markets, and must also take account of the general rule that electric utilities are not exempt from the Federal antitrust laws, particularly where they voluntarily enter into commercial relationships governed in the first instance by business judgment and not regulatory coercion.[52]

This analysis is dispositive of applicant's argument here. We hold that, in applying Section 105c to the instant case, the Licensing Board did not err in the manner suggested by the applicant; our own antitrust scrutiny must go forward.

III.

RELEVANT MARKETS

At the outset, we endorse — over the applicant's objection — that portion of the Licensing Board's analysis which led it to conclude that the market for wholesale power in the applicant's service area was a relevant

[52] *Id.* 6 NRC at 912-13 (footnotes omitted).

market for the purposes of this proceeding. For the reasons which follow, however, we disagree with that Board's holding that there are no other relevant markets. As we explain, there are relevant markets both for coordination services and retail power; the geographic bounds of both markets also correspond to the applicant's service area.

A. Coordination Services Market

1. The Product Market. In the electric utility business, there is a common practice among the companies of interchanging power and energy and sharing responsibility for building new generating facilities to achieve economic benefits unattainable by an individual utility acting alone. Generally known as "coordination," the practice includes various arrangements among utilities for reserve sharing, emergency exchange of power and energy, economy exchange of power and energy, maintenance scheduling, seasonal capacity exchange, and staggered construction. The simple purpose of these arrangements is to allow producers of firm power[53] to lower their costs of production.

In the proceeding below, Justice, AEC and MEUA claimed that the sale or exchange of such power and energy and associated services comprised a relevant market for antitrust purposes — namely, a "coordination services" market separate from the wholesale and retail power markets.[54] Although taking a somewhat different position, the staff also claimed that there was a market for such services.[55] Not surprisingly, the applicant denied the existence of such a market.[56]

The Licensing Board rejected the proffered coordination services market on the ground that it "clearly would include a variety of factors that in no way could be close substitutes for one another." 5 NRC at 886. Although

[53]We defined firm power in *Midland* as "essentially a utility commitment to supply electric energy to a customer on demand for as long as needed. One contracting for firm power (whether at retail or wholesale) is buying not merely energy, but assurance that (barring some extraordinary unforeseen circumstance) the utility will make that power available without interruption when called for." 6 NRC at 950.

[54]Justice and MEUA referred to it as a "regional power exchange" market. Justice Prehearing Brief Below, 55-58; MEUA Prehearing Brief Below, 28-31. AEC denominated it as the "bulk power supply services market." AEC Prehearing Brief Below, 24. We first adopted use of the term "coordination services" market in our *Midland* decision. We use that term here as we think it best describes the practice which makes up that market. For a detailed discussion of the factors which make up the coordination services market, see *Midland*, 6 NRC at 902-03, 949-77.

[55]The staff's original position was that the elements of the coordination services market combined with the market for firm wholesale power to form a single bulk power services market. Staff Prehearing Brief Below, 52-54. However, it no longer adheres to this position. In view of our *Midland* decision, the staff now concedes that a separate market for coordination services exists. Staff Reply Brief, 43-44.

[56]See APCO Proposed Findings, 447-57.

the Licensing Board expressly recognized that in some cases a number of diverse services could be clustered and treated as a single market (citing *United States* v. *Philadelphia National Bank*,[57] it apparently thought that *United States* v. *Grinnell Corporation*[58] precluded that treatment here. Interpreting *Grinnell* as requiring the factors making up the proffered market to be "reasonably interchangeable" with each other, the Board found that they were "not usually close substitutes for one another" and, hence, "not in the same market." *Id.* at 887.

On appeal, the parties essentially adhere to their original positions. The applicant supports the Licensing Board's decision, its principal post-*Midland* argument being that the existence of a coordination services market in the area involved here lacks evidentiary support.[59] The other parties oppose the conclusion reached by the Licensing Board.[60] Their argument basically is that not only is there evidence indicating the existence of such a market, but that a finding to that effect is required by *Midland* and applicable judicial decisions. We agree with this position.

a. Because the Licensing Board decision turned on what it considered to be the teaching of *Grinnell*, we begin our analysis with a detailed review of that case. *Grinnell* involved the question of whether the defendant company had monopolized the market for accredited central station service[61] in violation of Section 2 of the Sherman Act. The District Court had treated the entire accredited central station service business as a single market.[62] The company argued, however, that the individual central

[57]374 U.S. 321 (1963).
[58]384 U.S. 563 (1966).
[59]APCO Reply Brief, 23-38.
[60]Justice Brief, 135-149; Justice Reply Brief, 14-20; Staff Brief, 10-20; Staff Reply Brief, 42-44 AEC Brief, 83; AEC Reply Brief, 11-13; MEUA Brief, 41-46.
[61]Central station service, simply put, protects premises by installing thereon fire or burglary (or both) detection devices which automatically transmit an electric signal to a central station which is manned 24 hours a day. Upon receipt of a signal, the central station, when appropriate, dispatches guards to the protected premises and notifies the police or fire department directly. An accredited central station service is one which has been approved by insurance underwriters. 384 U.S. at 566-67.
[62]Among the various central station services offered were the following:

(1) automatic burglar alarms;
(2) automatic fire alarms;
(3) sprinkler supervisory service (any malfunctions in the fire sprinkler system — *e.g.* changes in water pressure, dangerously low water temperatures, etc. — are reported to the central station); and
(4) watch signal service (night watchmen, by operating a key-triggered device on the protected premises, indicate to the central station that they are making their rounds and that all is well; the failure of a watchman to make his electrical report alerts the central station that something may be amiss).

station services are so diverse that, under *du Pont*,[63] they cannot be lumped together to make up the relevant market.

In upholding the lower court's decision, the Supreme Court declared:

> But there is here a single use, *i.e.*, the protection of property, through a central station that receives signals. It is that service, accredited, that is unique and that competes with all the other forms of property protection. We see no barrier to combining in a single market a number of different products or services where that combination reflects commercial realities. To repeat, there is here a single basic service — the protection of property through use of a central service station — that must be compared with all other forms of property protection.

384 U.S. at 572.

The Court went on to say:

> Burglar alarm service is in a sense different from fire alarm service; from waterflow alarms; and so on. But it would be unrealistic on this record to break down the market into the various kinds of central station protective services that are available. Central station companies recognize that to compete effectively, they must offer all or nearly all types of service. * * * We held in *United States* v. *Philadelphia Nat. Bank*, 374 U.S. 321, 356, that "the cluster of services denoted by the term 'commercial banking' is a distinct line of commerce." There is, in our view a comparable cluster of services here.

Then, specifically addressing *du Pont*, the Court explained:

> There are, to be sure, substitutes for the accredited central station service. But none of them appears to operate on the same level as the central station service so as to meet the interchangeability test of the *du Pont* case. Non-automatic and automatic local alarm systems appear on this record to have marked differences, not the low degree of differentiation required of substitute services as well as substitute articles.

Id. at 572-73.

The Supreme Court in *Grinnell* did not, as the Licensing Board apparently thought, lay down a rule that a market could never be

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Id. at 566 n.4.

[63] *United States* v. *E.I. du Pont de Nemours & Co.*, 351 U.S. 377 (1956) (the cellophane case).

comprised of products and services which were not interchangeable with each other. For, in holding that the combination of services comprising the central station service constituted a relevant market, the Court expressly indicated that it was following the course it had adopted in *Philadelphia National Bank.* In that case, the Court found that the cluster of clearly diverse products (various kinds of credit) and services (such as checking accounts and trust administration) denoted by the term "commercial banking"[64] comprised a product market "sufficiently inclusive to be meaningful in terms of trade realities." 374 U.S. at 356-57.

To be sure, the Court in *Grinnell* did take note of its ruling in *du Pont* that products and services which consumers may reasonably interchange for the same purposes make up a relevant market. But in *Grinnell,* the "interchangeability" with which the Court was concerned related to whether there were in the market place available alternatives to overall central station service itself; the Licensing Board's application of the "interchangeability" test here would indicate a contrary belief that the individual products and services making up the central station service had to be interchangeable with each other. In other words, the fact that central station service was made up of various products and services which were not interchangeable did not prevent the Court from holding the central service itself to be a relevant market. In this respect, the Court's action was not novel. It did no more than follow an avenue it had opened up in *Philadelphia National Bank* some three years earlier.[65]

b. Owing to the erroneous view it took of *Grinnell,* the Board below rejected the proffered coordination services market on grounds we cannot uphold. We must then take the next step and ascertain for ourselves whether such a market exists in terms of "commercial or trade realities" and, if so, what that market's dimensions are. Fortunately, that work has been made easier by our prior decision in *Midland.* Notwithstanding the

[64]More specific examples of banking "products" identified by the Court were: unsecured personal and business loans, mortgage loans, loans secured by securities or accounts receivable, automobile installment and consumer goods installment loans, tuition financing, bank credit cards, revolving credit funds. Examples of banking services included: acceptance of demand deposits from individuals, corporations, governmental agencies, and other banks; acceptance of time and savings deposits; estate and trust planning and trusteeship services; lock boxes and safety deposit boxes; account reconciliation services; foreign department services (acceptances and letters of credit); correspondent services; and investment advice. 374 U.S. at 326 n.5.

[65]For other cases holding that a bundle of products and services can constitute a relevant market, see *United States* v. *Connecticut National Bank,* 418 U.S. 656 (1974); *United States* v. *Marine Bancorporation, Inc.,* 418 U.S. 602 (1974); *United States* v. *Phillipsburg National Bank,* 399 U.S. 350 (1970); *United States* v. *United Shoe Machinery Corp.,* 110 F. Supp. 295 (D. Mass. 1953), *aff'd per curiam,* 347 U.S. 521 (1954); *Credit Bureau Reports, Inc.* v. *Retail Credit Co.,* 358 F. Supp. 780 (S.D. Texas 1971), *aff'd* 476 F.2d 989 (5th Cir. 1973).

fact that *Midland* involved other utilities in a different part of the country, we find its teachings useful here for the reason expressed by the Licensing Board based on its analysis of the evidence in this case:

> The principles of electric power supply production and coordination are generally applicable throughout the electric utility industry (Mayben, Direct, pp. 3-9). These principles do not vary significantly among electric utilities regardless of differences in locations, although they may change to a certain extent depending on corporate policy and financial requirements (Mayben, Direct pp. 8-9; Tr. 5,576-5,586; *FPC National Power Survey,* Part I, Chapter 17 "Coordination for Reliability and Economy," December 1971).

5 NRC at 834.

In *Midland,* we traced in painstaking detail the operations of the electric utility industry.[66] We discussed the manner in which utilities interact with each other in planning for and constructing the necessary transmission and distribution facilities and in operating them. We explained how, because of the peculiar characteristics of electricity, utilities buy, sell and exchange surplus bulk power and associated services to improve the efficiency and reliability of their operations. For reasons there discussed, we concluded that there existed a separate coordination services market consisting of these types of transactions. We stated:

> [C]oordination arrangements usually comprise several differing types of surplus power transactions and associated services . . . . [T]hese various power transactions are not reasonably interchangeable with wholesale power. But neither are they necessarily interchangeable with one another. All, however, serve an essentially similar function. That function is facilitating production of firm bulk power at lower cost and with greater reliability by making profitable use of otherwise surplus generating capacity. These arrangements constitute a "bundle of services" which merits recognition as a distinct market similar to the way various services offered by commercial banks fall in one and the same product market. *United States* v. *Philadelphia National Bank, supra,* 374 U.S. at 356.

6 NRC at 975.

We know of no compelling reason for reaching a different conclusion here. As will be seen, the evidence in this proceeding reveals that the same

[66]See 6 NRC at 949-74.

kinds of transactions found to occur in Michigan take place in Alabama as well.[67]

The Southern Company Power Pool Intercompany Interchange Contract (D.J. 3009),[68] to which applicant is a party, provides the contractual framework within which the members of the Pool engage in coordination services transactions. Although not every type of service available under the agreement is specifically identified, the terms of the agreement, viewed in light of the manner in which the utility industry generally operates, leave little room to doubt that the various coordination services activities are actively pursued by the utilities involved.

For proof of the validity of this observation, we need but cite applicant's own admission contained in the power pool agreement:

* * *

> WHEREAS, each of the POWER COMPANIES and their respective customers achieve substantial economies *through the common planning, development, and coordination of their operations which they have successfully practiced for many years, and*
>
> WHEREAS, such common planning, development, and coordination provides certain advantages to POWER COMPANIES and their respective customers including:
>
> (a) The staggering of the construction of new generating facilities so that each of the respective POWER COMPANIES can construct and install for their respective territorial loads the optimum size generating facilities which produce maximum economies of scale;
>
> (b) An opportunity for each of the respective POWER COMPANIES to dispose of surplus energy and capacity that may be available from time to time due to the staggered construction of generating units, seasonal variations in demands for electric power, and variations in patterns of the diversity of loads imposed from time to time on the respective POWER COMPANIES;
>
> (c) An opportunity to utilize the seasonal and diversity patterns of other utilities not contiguous to each of the respective POWER COMPANIES for the outlet of surplus capacity and energy which may

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[67]We found in our *Davis-Besse* decision a similar market to exist in the territories served by the utilities there involved. 10 NRC at 287, 301-02.

[68]In referring to the exhibits and testimony submitted below, we have followed the system of notation used by the Licensing Board. See 5 NRC at 820 n.4.

be available from time to time, together with the opportunity, because of such variation in seasons and diversity of loads, to acquire from other utilities energy at a low cost and thus avoid or defer the construction of generating capacity to meet seasonal loads;

(d) The opportunity to pool reserves thus reducing the magnitude of reserve capacity required by the respective POWER COMPANIES in order to assure reliable service to their respective customers and

(e) Improvements in the reliability of electric service through the use of transmission interconnections which provide the respective POWER COMPANIES with the opportunity to call upon one another as well as other utilities with which they, or any of them, are interconnected to provide backup service in case of emergencies or breakdowns in excess of the reserves carried by the respective POWER COMPANY;

* * *

D.J. 3009, pp. 2-3 (emphasis supplied).

Other evidence confirms that the applicant engages in various "coordination services" transactions. It participates in joint ownership arrangements as, for example, with the Georgia Power Co. over the Gaston coal-fired generating plant (D.J. 1002); it shares reserves with the other companies in the Southern Pool (D.J. 603, 604, 605, and 3009); it engages in short-term capacity exchanges with neighboring utilities (Mississippi Power and Light, D.J. 3002; Duke Power Co., D.J. 3003; South Carolina Electric & Gas Co., D.J. 3004; Tennessee Valley Authority, D.J. 3007; and Florida Power Corporation, D.J. 3008); it participates in seasonal capacity exchanges with TVA and with the Florida Power Corporation (D.J. 3007, 3008, 3009, 603, 604, and 605); and it exchanges emergency, maintenance and economy energy with other utilities (D.J. 3002, 3003, 3004, 3007, 3008, 3009, 603, 604, and 605).

Even without our *Midland* decision as precedent, we would reach the same conclusion here. As we have emphasized, court decisions teach that, for antitrust analysis purposes, a relevant market must reflect commercial or trade realities.[69] Guided by that rule, our review of the record in this proceeding persuades us that there exists a coordination services market

[69]See, *e.g.*, *Phillipsburg National Bank*, *supra*, 399 U.S. at 360; *Grinnell*, *supra*, 384 U.S. at 571-76; *Philadelphia National Bank*, *supra*, 374 U.S. at 356-57.

comprised of the types of transactions for the sale and exchange of power and energy and associated services discussed above.[70]

c. The applicant does not disagree with the applicability of the "trade realities" rule to the matter at hand. Indeed, it specifically endorses that rule's controlling effect here.[71] It does, however, dispute the conclusion advocated by its opponents. Its position essentially is that, whatever may be said of the electric utility industry generally, the evidence in this record simply is insufficient to show a coordination services market exists in the area of interest here.[72]

To support this position, the applicant challenges the testimony of Mr. Mayben and Dr. Wein, Justice's two principal witnesses. At the core of its attack is the proposition that these witnesses possess no factual knowledge of the operations of the utilities in Alabama (beyond the terms of certain contracts and rate schedules furnished them) and that, consequently, their testimony lacks foundation and is entitled to no weight.[73]

We cannot accept applicant's position. To begin with, we disagree with its thesis regarding the state of the witnesses' factual knowledge of the operations of the utilities involved. Both Mr. Mayben and Dr. Wein have expertise in the utility field.[74] Beyond that, Mr. Mayben had studied not

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[70]In *Midland*, we excluded from the coordination services market there involved "developmental coordination" — *i.e.*, the construction of power plants on a staggered basis or as joint ventures by two or more utilities with the intention of sharing the power generated by them — but included within that market the purchase and sale of "unit power" from such plants. 6 NRC at 976. Similarly, we do not include "developmental coordination" within the coordination services market held to exist here.

[71]In applicant's own words:

> "The touchstone of market analysis is identifying patterns of trade and commercial realities in a designated area."

APCO Reply Brief, 37.

[72]Applicant also advances another argument. Avowedly to show the "lack of commercial reality" of the coordination services market, the applicant explains in detail how it is part of an "integrated public utility system" with three other utilities which form the Southern Company, a holding company approved by the SEC; and how AEC gained by obtaining its deficit power and energy requirements from applicant rather than from the four-company power pool. APCO Reply Brief, 32-37; see also App. Tr. 79-92. Far from showing a lack of commercial reality, the fact that AEC and the applicant engage in such arrangements and that AEC finds it economical to do so indicates the very opposite — that there is a market for bulk power to meet deficit requirements.

[73]APCO Reply Brief, 23-38.

[74]Mr. Mayben is a professional engineer registered in some thirteen states. Since 1965, he has been a partner and supervising executive engineer with R. W. Beck and Associates involved in providing consultant engineering services to various utilities. His work experience has included the design of power generating stations, high-voltage transmission lines and substations; and power supply planning with particular concern with power pooling and coordinated supply. He has served as the principal Systems Engineer to the Missouri Basin Systems Group (MBSG), a power planning and power pooling organization, whose electric utility members

only the terms of the power pool and other agreements entered into by the utilities in Alabama and in the neighboring areas, but the rate schedules on file with the Federal Power Commission (now Federal Energy Regulatory Commission); in addition, and perhaps most important, he had analyzed the pool operating minutes — which detail the actual transactions that take place.[75] Dr. Wein, in turn, based his testimony on the existence of a

have generation and transmission facilities covering a multi-state area in the Upper Missouri River Basin. Since 1967, he has also worked extensively in the development and implementation of an ongoing bulk power supply program for the Nebraska Public Power District, a utility which has the bulk power responsibility for a major portion of the State of Nebraska. Mayben, Direct, 1-5.

Dr. Wein's background is equally impressive. He is a professor at the Graduate School of Business Administration at Michigan State University, a position he has held since 1959. From 1961 through 1963 he was on leave while serving as Chief Economist and Head of the Office of Economics of the Federal Power Commission (now Federal Energy Regulatory Commission). Thereafter, he, along with others, established the Institute of Public Utilities at Michigan State University in 1965. Before becoming a professor at Michigan, he was Associate Professor of Economics and Industrial Administration at the Carnegie Institute of Technology, a consulting economist for industry, principal economist of the Antitrust Division of the Justice Department (where he also served as special advisor to the Attorney General on antitrust problems in the steel industry), principal economist in the Office of Price Administration, a senior statistician with the Army Air Forces, a principal economist in the War Production Board and a junior economist in the U.S. Commerce Department. He holds a masters degree in economics from Columbia University and a Ph.D. in economics from the University of Pittsburgh. Wein, Direct, 1-16.

[75]On cross-examination, Mr. Mayben explained the basis for his knowledge of the operations of APCO in the following manner:

> Q. Mr. Mayben, am I correct in my understanding that the knowledge which you have of such portion of the so-called regional power exchange market denominated by you is based upon transactions reflected in certain rate schedules on file with the Federal Power Commission which were furnished to you by the Department of Justice?
>
> A. Yes, that information was used in my preparation of this proposed Exhibit 101.
>
> Q. Does your knowledge of such portion of the regional exchange market come from any other source of information which you can specify?
>
> A. Yes. It comes from my experience in working with clients who are engaged in regional exchange activities and my ability to interpret contracts as to the types of transactions which customarily occur under interconnection agreements which have interchange type service schedules to them.
>
> Q. Other than this general knowledge, Mr. Mayben, is there any other source for the particular regional power exchange market which you assert here?
>
> A. Well, of course, I did examine the pool Operating Committee Minutes, and information there led me to believe that in fact there were transactions taking place pursuant to the contracts that the Department of Justice provided to me.

Tr. 1721-22.

coordination services market in large part on what he learned from Mr. Mayben concerning the manner in which utilities operated.[76] Considering

[76]Dr. Wein explained the basis for his testimony as follows:

MR. MILLER: Just a minute. Your were asked about Mr. Mayben.

THE WITNESS: That's right. I asked him then whether the structure of the industry — of course I know some of that myself, but I wanted to get his view, as to whether wholesale power was a reasonable type of transaction, one which occurs in Alabama, and of course I asked about the [*Midland*] case, because we were both associated there, too.

Yes. He thought that there are wholesale transactions and he described the kinds of conditions under which wholesale transactions take place.

Of course, there was a question of retail, where does wholesale leave off and retail begin. That sort of thing. That's the sort of thing I asked Mr. Mayben to do.

In the [*Midland*] case, I asked him to do another.

MR. MILLER: I don't think you were asked about that.

THE WITNESS: I'm sorry. I sort of mix these things up.

BY MR. BALCH:

Q. Did you ask Mr. Mayben to undertake this analysis or investigation without any further delineation or instructions?

A. Which analysis and investigation?

Q. You said you asked him to find out what kind of transactions take place.

A. He didn't have to make any analysis or investigations. He knew. He just told me and explained to me what they meant. Then I read up about it.

CHAIRMAN GLASER: Well, did he tell you what the source of his knowledge was?

THE WITNESS: Well, he said the source of his knowledge was, he was an engineer, had negotiated many contracts and he knows the business. I didn't know beyond that.

BY MR. BALCH:

Q. Did you assume that the same kind of transactions would take place in the southeast as have taken place perhaps in the northeast or the Missouri Basin?

A. All I asked him were the kinds of things that would take place in a power pool. Then I asked him, did it make much difference whether it would be in Alabama or any other place and he said, the importance might change. Some might have more sorts of transactions. Some might have less sorts of transactions. But in effect, the transactions, all could be classified under very common classification.

Q. Did he choose the transactions from which his analysis would be made, or did you choose the transactions from which the analysis would be made?

the universality of these utility practices, confirmed by the Board below and by us in *Midland*,[77] we find no merit to the applicant's position that the testimony of Mr. Mayben and Dr. Wein lacks factual foundation.

An even more compelling reason requires rejection of applicant's argument. Although expressed in terms of a failure of the other side's proof, the unstated premise underlying the argument is that applicant in fact does not engage in the kind of coordination activities to which Mr. Mayben and Dr. Wein testified. The critical failing of this premise is that it runs directly counter to the very words subscribed to by the applicant and the other parties to the Southern Company power pool agreement — an agreement which has continued in effect for some 30 years.[78] In that agreement, the signatories not only specifically admitted to having "successfully practiced for many years . . . common planning, development, and coordination of their operations," but also to a desire to "continu[e] . . . coordinated operation."[79] Applicant would now have us disregard those words as no more than wasted ink. This we cannot do.

To sum up, we are satisfied from our review of the record that, for purposes of this proceeding, a coordination services market exists in the general area of applicant's operation. We need only to determine its geographic dimensions to complete our analysis of that market. We turn now to that task.

2. **The Geographic Market.** In the proceeding below, Justice took the position that a coordination services market "by its very nature does not lend itself to precise geographic market definition. Electric utilities with access to this market range far and wide in search of useful power exchange transactions; they are not restricted to specific geographic limits or certain identified utilities with whom they may deal."[80] For these reasons, Justice maintained that precise definition of the geographic boundaries of this entire market is not necessary to a consideration of monopolization charges; it suffices to focus attention on a separate economic entity or submarket within the broader market.[81]

In *Midland*, Justice took a similar position. On that occasion, we said:

A. I think we sort of jointly agreed on what the transactions were.

Tr. 13,358-60.

[77]See 5 NRC at 833-37 (*Farley* below); 6 NRC at 1066-67 (*Midland*); see also pp. 1050-1051, *supra*.

[78]The power pool agreement bears an original date of October 16, 1950. This gives an indication of the extended period during which applicant has been involved in coordination activities. See D.J. 3009.

[79]*Id.* at pp. 2-4.

[80]Justice Prehearing Brief Below, 57.

[81]*Id.* at 58.

> We agree with Justice's legal position. Where a discrete submarket exists within an overall geographic market, monopolization of the submarket is itself an antitrust violation. *Brown Shoe Co.* v. *United States, supra,* 370 U.S. at 336-37; *Case-Swayne Co.* v. *Sunkist Growers, Inc., supra,* 360 F.2d at 455-59; *In re Luria Brothers and Co., supra,* 62 FTC at 612-14. A submarket must correspond to commercial realities and be economically significant, *Brown Shoe, supra,* and its existence is a question of fact that must be "charted by a careful selection of the market area in which the seller operates and to which the purchaser can practicably turn for suppliers." *United States* v. *Philadelphia National Bank, supra,* 374 U.S. at 359.

6 NRC at 977.

Those same observations guide us here. The record in this proceeding discloses that the applicant engages in exchanges of power directly or through other Southern Pool members with surrounding electric utilities, including Mississippi Power & Light Co., Florida Power Corp., Duke Power Co., South Carolina Electric & Gas Co., and TVA (Mayben, Direct, 54-55; D.J. 101, 3002, 3003, 3004, 3007, 3008; Wein, Direct, 62-64). Thus, at first impression there might seem to be support for a finding of a broad geographic market encompassing the areas in which these utilities operate.

But we need not pause to look for a precise definition of the geographic boundaries of such an overall market. For that is not the market relevant to our inquiry. For purposes of this proceeding, we must focus on that market area, within the overall market, to which the smaller utilities in Alabama can practically turn for suppliers.

The record in this proceeding discloses that the area within which AEC and the other utilities comprising MEUA[82] may seek coordination services is limited to applicant's service territory and nearby environs — central and south Alabama. Applicant owns all transmission lines in the area over 115 kv and controls all transmission facilities to utilities outside that area. 5 NRC at 900-01; D.J. 1000; D.J. 1006; D.J. 1008; AEC X CRL-1A; St. John, Direct, 7, 39; Harris Tr. 25,455-59. As a result, it has the power to grant or deny access by AEC and the other utilities to the kind of coordination services engaged in by APCO. For these reasons, we conclude

[82]While MEUA might arguably be considered a participant (or potential participant) in the coordination services market, we think it worth repeating a point we made in *Midland*: for a utility without any generating capacity of its own, "[c]oordination power services are not useful to it and for its purposes are not functionally interchangeable with wholesale power. In short, given the nature of coordination power, [non-generators] literally cannot substitute coordination power for wholesale power as a long-term source of firm electric power." 6 NRC at 963. As the Board below noted, none of the members of MEUA owns or operates any generating facilities. 5 NRC at 827.

there exists, for purposes of our antitrust analysis, a *relevant* coordination services market in central and south Alabama, the area within which AEC and the other smaller utilities are confined for access to that market in terms of "commercial or trade realities."

B. Retail Market

In the proceeding below, Justice and both intervenors submitted that the retail market for firm power constituted a relevant market within which to examine applicant's conduct.[83] The product market was defined as the supply of firm power to the ultimate consumer;[84] the geographic market was seen as corresponding to central and southern Alabama, "the area where applicant sells or could reasonably compete to sell at retail."[85]

The Licensing Board agreed that "[r]etail firm power is clearly a distinct product market." 5 NRC at 887. Citing *Otter Tail Power Co.* v. *United States*,[86] the Board further found that the economic viability of retail distribution systems is worthy of antitrust protection. *Id.* at 889. It nevertheless rejected the proposed market. While conceding that some competition exists "in the interstices of the service areas of retail distribution systems," the Board found that the local distribution of retail power is a natural monopoly and that the rivalry among retail sellers is insufficient to bind all of central and south Alabama into one geographic market. *Id.* at 888. And, while it determined that the hundreds of individual local markets would have been proper subjects for examination, the Board saw no purpose in examining such "natural monopoly" situations for antitrust violations. The Board concluded: "Competition *between* retail distribution systems, if it is of only infra-marginal proportions, is presumably outside of the scope of antitrust remedy." *Id.* at 889 (emphasis in original).

The Board sought to bolster its conclusion by referring to *Otter Tail.* In that case, the Board wrote, "the focus [was] upon the retail distribution entity as a buyer (or potential buyer) in the wholesale power market." Every anticompetitive practice in the case was said to have taken place at the wholesale level. The relief decree "in every facet, affected retail distribution systems in their access to and role as buyers in the market for bulk wholesale power." This led the Board to write that there is a "market which

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[83]The NRC staff argued below the relevance of only one market — that for "bulk power supply and bulk power supply services." Staff Proposed Findings, 27 (¶3.02). On appeal, the staff changed its position in light of our decision in *Midland;* it now maintains that separate markets exist for coordinated services and for wholesale power. Staff Reply Brief, 42-45; see also, fn. 55 *supra.* The staff made no mention of the retail market either below or on appeal.

[84]See, *e.g.*, Justice Proposed Findings, 62 (¶4.01).

[85]See, *e.g.*, Justice Proposed Findings, 65 (¶4.07).

[86]410 U.S. 366 (1973), *affirming in part and remanding in part,* 331 F. Supp. 54 (D. Minn. 1971).

is singularly relevant for the licensing of nuclear facilities to generate electricity: the market for wholesale power." *Id.* at 889-890.

Justice, AEC, and MEUA all excepted to the Board's rejection of the proffered retail market.[87] On appeal, they argue that the Board was factually incorrect when it failed to find sufficient competition at retail to justify grouping central and south Alabama into one geographic market. Moreover, they cite both *Otter Tail* and our decision in *Midland* as requiring reversal of the rejection below of the retail market.

1. **The Market in Otter Tail.** We begin our analysis by taking issue with the Licensing Board's interpretation of *Otter Tail.* As the Board stated, the violations in that case took place at the bulk power level; the remedies were applied at that level as well. But the *market* involved in the case was the retail market. It was this market that the defendant was attempting to monopolize; the remedies were designed to effectuate competition at the retail level, not the wholesale level. The district court's decision in *Otter Tail* puts any doubt about this to rest. See 331 F.Supp. 54, 58, 61 (D. Minn. 1971).

In the case now before us, applicant is allegedly attempting to monopolize (or has succeeded in monopolizing) three separate markets. It is further claimed that an unconditional license to operate the Farley facility assertedly will have anticompetitive effects on all three markets. In such a situation, we do not read *Otter Tail* as mandating that we restrict ourselves to an analysis of the wholesale market. To the contrary, we see that case as standing for the proposition that the markets relevant for analysis are all those in which anticompetitive effects may be felt.

2. **The Product Market.** Beyond its espousal of the view that the bulk-power market is the "singularly relevant" market in NRC antitrust actions,[88] the Licensing Board appeared to have one fundamental problem with the proposed retail market: it simply did not believe there was sufficient actual (or potential) competition at retail to justify antitrust analysis. The advocates of the market contend that the Board was factually incorrect in its assessment of the amount of competition at retail; they see the retail situation in Alabama as nearly identical with the situation we found in *Midland* to exist in Michigan.[89] Applicant, on the other hand, argues that the potential for retail competition in Michigan was far greater

[87]Justice Exceptions, pp. 2-3 (Exceptions 6 and 7); AEC Exceptions, pp. 2-3 (Exceptions 5 and 6); MEUA Exceptions, pp. 1-2 (Exceptions 4, 5 and 6).
[88]A view not shared by us in *Midland* (6 NRC at 949-97) and *Davis-Besse* (10 NRC at 270, 301-02); in both cases all three markets offered here were found relevant.
[89]Justice Brief, 148-49; Justice Reply Brief, 24-28; MEUA Brief, 6-17.

than in Alabama; it sees no inconsistency between the Licensing Board's decision and *Midland*.[90]

In assessing the extent of retail competition, it is important to consider the nature of the industry involved. Most retail consumers of electricity are locked into a particular supplier; the residents of Birmingham, for example, must currently look to applicant for their electric needs. As the Supreme Court said in *Otter Tail* (410 U.S. at 369): "[e]ach town . . . generally can accommodate only one distribution system, . . . making each town a natural monopoly market for the distribution and sale of electric power at retail." Clearly we are not dealing with a product that is susceptible to intense competition for every sale.

This is not to say that retail competition is either impossible or unprotected by the antitrust laws; *Otter Tail, Midland,* and *City of Mishawaka* v. *American Electric Power Co. (Mishawaka II)*[91] are cases that all hold otherwise. Although competition for individual users already taking electric service from a supplier may be unlikely to occur,[92] competition can take place for certain new loads or for the right to be sole distributor in a municipal area.[93] There can also be "yardstick competition";[94] the existence of a potential competitor may have an effect on the actions of another distributor.

In Alabama, franchise, individual load, and yardstick competition are all present to some degree. In terms of franchise competition, Alabama law prohibits utilities from serving within municipal corporate limits without the permission of the municipal government.[95] An examination of a list of applicant's franchises (prepared in 1973) reveals that applicant had 313

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[90]APCO Reply Brief, 38-44.

[91]465 F. Supp. 1320 (N.D. Ind. 1979), *aff'd in part and remanded on other grounds,* 616 F.2d 976 (7th Cir. 1980), *cert. denied,* 449 U.S. 1096, 66 L.Ed. 2d 824 (1981).

[92]Although such competition is rare, we found in our *Davis-Besse* decision that street-to-street, head-to-head competition took place in a good part of the City of Cleveland. 10 NRC at 274. While there is less of it in Alabama, the Board below found such competition in the Town of Samson. 5 NRC at 888.

[93]The fact that local distribution may be a natural monopoly does not mean the identity of the monopolist cannot change. In *Otter Tail,* for example, the sole competition found by the Court was for the control of local distribution monopolies.

[94]"Yardstick competition" is a form of competition in which two sellers (in this case, distributors of retail power), not directly competing against each other for sales, have their pricing policies (and any other practices deemed relevant by purchasers) compared. As it relates to the retail distribution of electricity, a local distributor's performance is measured against that of other nearby utilities. If yardstick competition exists in the area, the local distributor will have to compare favorably with the other utilities or it will be replaced. If this form of competition is not present, the local distributor need not be concerned about meeting the price and services of other utilities.

[95]Farley Direct, 46; 562-64; Alabama Constitution of 1901, § 220.

different franchises in 273 municipalities. Of those, only 26 franchises in 24 locations are terminable; the balance are perpetual.[96]

In terms of its retail sales, in 1973 applicant made 51% of such sales in municipalities where it holds perpetual franchises, 9% in muncipalities where it has terminable franchises, and 40% outside of municipalities (where no franchises are required).[97] Perpetual franchises in Alabama are not exculsive;[98] municipalities may offer competing franchises to other utilities. Under the terms of the Booth Act,[99] however, municipalities may not establish a municipally-owned system without first offering to purchase the facilities of the existing franchisee. Should the franchisee decline the offer, the municipality may establish its own competing system, but the original franchise (unlike in Michigan and in the states served by Otter Tail) would still be in effect.[100] Thus, in the vast majority of its service area, applicant can be subjected to head-to-head competition, but it cannot necessarily be replaced. Due in no small part to the economic difficulties inherent in establishing a competing system, no municipality in applicant's service area has ever set up a distribution system to compete against one of applicant's franchises.[101]

Alabama Power has acquired some other distribution systems since 1950, but it takes pains to point out that none of these acquisitions has been at the expense of municipally-owned systems.[102] The primary acquisition was that of the Birmingham Electric Company (by merger) in 1952.[103] Other acquisitions included Liddell Power Company (a privately-owned utility largely operating in Camden, Alabama) in 1955,[104] the electric facilities of West Point Manufacturing Company (a textile company that previously provided electric service to its former "mill villages") in 1960,[105] and the

[96]APP.X JMF-82. Of the terminable franchises, three (Bay Minette, Brewton, and the transmission franchise in Dothan) are listed as "terminable;" the other franchises expire in a certain number of years (usually thirty years after issuance). While our arithmetic does not square with applicant's testimony that it holds franchises in only 261 municipalities (Crawford Direct, 30), the discrepancy may be based on the limited nature of some of the franchises listed in JMF-82.

[97]Crawford Direct, 119. In comparison, 45% of Consumers Power's retail sales were made under perpetual franchises. *Midland*, 6 NRC at 933.

[98]See *Bessemer* v. *Birmingham Electric Co.*, 248 Ala. 345, 27 So. 2d. 565 (1946).

[99]Title 48, Alabama Code §§ 342-347.

[100]There is some question as to whether a municipality possesses the authority to condemn an established distributor's property. See App. Tr. 151.

[101]The town of Ozark initiated a proceeding under the Booth Act in 1956 in an attempt to establish its own distribution system. Applicant elected not to sell its facilities and the town never constructed a competing system. See *Alabama Power Co.* v. *Alabama Public Service Commission*, 267 Ala. 474, 103 So. 2d 14 (1958).

[102]APCO Reply Brief, 39.

[103]Farley Direct, 227-32.

[104]*Id.* at 246-47.

[105]*Id.* at 270-71.

electric facilities of Mount Vernon Mills (another textile company) in 1968.[106] During this same time period, the company sold small amounts of its distribution system in areas into which the cities of Bessemer, Sylacauga, and Opelika extended their corporate limits.[107] In addition to these transactions, applicant has been approached at times by towns requesting that it supply retail service in lieu of the service then being provided by cooperatives.[108] In other instances, unincorporated rural communities presently served by cooperatives have considered incorporating and extending a franchise to applicant.[109]

As mentioned earlier (see pp. 1062, *supra*), there is no head-to-head competition for most electric loads. Nonetheless, all the parties agree that there is some competition for individual loads.[110] This competition occurs in: (1) the town of Samson (served by both applicant and Covington Electric Cooperative, which compete on a house-by-house basis); (2) outlying areas annexed by a municipality where another supplier currently serves at retail;[111] (3) rural areas either where competition for individual loads is permitted (in certain circumstances) by non-duplication agreements or where rural systems are located near each other and have not signed any such agreements; and (4) outlying areas where a municipally-owned system wishes to expand.[112] Applicant argues that the opportunities for such head-to-head competition are "minimal."[113] While we can agree that there is not head-to-head competition for the great percentage of retail sales in the area, we do not believe such competition can be ignored.[114]

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[106] *Id.* at 322-23.

[107] *Id.* at 247-51.

[108] See, *e.g.*, DJX 4012-24 (Town of Samson); DJX 4205-16 (Fulton); DJX 4319 (Clio); DJX 4320 (Red Level); DJX 4321 (Goshen).

[109] See, *e.g.*, DJX 4185 (Pennington); DJX 4317-4318 D (Riverview).

[110] See, *e.g.*, APCO Reply Brief Below, 228; Justice Proposed Findings, 41-45 (¶¶2.35-2.45). ["________ Reply Brief Below" refers to the parties' responses below to the proposed findings of fact.]

[111] In such a situation, the system franchised by the municipality (or, if the case may be, a municipally-owned system) can compete in the annexed area with the preexisting distributor. Head-to-head competition can result or the nonfranchised system can sell its facilities to the other system.

[112] In Alabama, there does not appear to be any legal limit to the extent municipally-owned systems may expand outside municipal corporate limits, subject to the grant of a franchise should the system wish to provide service in another incorporated area. In Michigan, by contrast, the expansion of municipal systems beyond municipal corporate boundaries is limited. *Midland*, 6 NRC at 940.

[113] APCO Reply Brief Below, 228.

[114] In this context, we note the following dialogue between applicant's president, Joseph Farley, and counsel for the Department of Justice (at Tr. 20,804-05):

> Q: Don't your franchises substantially protect you against the loss of your retail business?

There is also yardstick competition taking place in Alabama. The Licensing Board wrote: "possibly the yardstick most often used in measuring the performance of any retail distribution system in central and south Alabama is that of another distribution entity in the same area."[115] The presence of yardstick competition plays a significant role in franchise and individual load competition; when one utility cannot meet another's rates or service, it can lose customers.[116]

In sum, retail competition is not completely absent from central and southern Alabama. Nor has applicant shown us any legal prohibitions barring greater competition. To be sure, the *economic* barriers to increased competition are substantial. The same was true in *Midland* where we found the retail market relevant. We repeat what we said there:

> This is not to suggest that competition to distribute electric power in lower Michigan is totally free and open, or even that major market changes are in the offing. But because this potential competition manifests itself only periodically and is more limited than that found in some unregulated markets, it is not for those reasons less deserving of antitrust protection. To accept Consumers' position on the relevant

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[Mr. Farley]: No sir, they are non-exclusive and there is an awful lot of load that is outside of municipal corporate boundaries, particularly industrial business today tends to locate outside the municipalities rather than in the middle of urban areas.

Q: So the fact that you have franchises that are to a great extent perpetual to serve in municipalities doesn't give you the feeling of being protected against losing business in those areas where you are franchised, Mr. Farley?

A: No sir, they are perhaps of some protection but as I have pointed out to you in the first place we experienced all the 11 counties of northern Alabama in which we had franchises and municipalities and we saw what happened there, that we were not protected there in any sense. We also know that a great deal of growth, industrial and commercial growth at this point in time tends to be outside of municipal corporate boundaries. Municipalities are finding it at least in our area harder and harder to extend their corporate limits and the tendency, as I said, is for a lot of the major industrial growth and some of the commercial growth to be outside of the municipal franchised areas.

Q: Are you saying there is a possibility of competition for such growth to serve such growth electrically, is that right?

A: Well, yes, sir, even when both systems are there. At retail if a load is over a certain size, 200 megawatts, under the tariffs that have been filed without, I might add, protest from the cooperatives, it's either party's business.

[115] 5 NRC at 888.

[116] See, *e.g.*, DJX-4329E (Vanity Fair Mills chooses service from Clark-Washington Cooperative because its bid was lower than applicant's); DJX-4319 (town of Clio expresses interest in service from APCO because cooperative service is more expensive); DJX-203 (City of Dothan challenges applicant's service to the town of Taylor by claiming Dothan's municipal system could provide better and cheaper service).

> retail geographic market would in effect nullify that protection. That result is simply out of line with the recent Supreme Court decisions in this area.
>
> It must also be kept in mind that Consumers was not born with a 77% or 100% portion of that retail market. Rather, it acquired its large share in no small part by the same slow competitive processes that it now suggests are too unlikely and remote for us to consider.

6 NRC at 988-89 (footnotes omitted).

We note too that, in similar circumstances involving the wholesale market in this case, the Licensing Board found the proposed market relevant for antitrust analysis. The Board recognized the obstacles to wholesale competition:

> A municipality served by Applicant under a franchise cannot shift easily to AEC; an AEC member cannot shift readily to Applicant for wholesale power. Clearly we are talking about competition at the margin here. As Applicant's witness Crawford testified in response to a question as to whether there was competition for wholesale loads: "The answer to that question is a qualified yes." (APP.X BJC-A (Crawford) p. 131).

5 NRC at 895.

The Board nonetheless concluded the market was relevant:

> Yet one of the lessons of economics is the importance and efficacy of marginal adjustments. In economic matters, tails often do wag dogs. In this market setting, it is precisely because buyers are often locked into one seller, and a seller limited to a definite geographic area for its retail customers, that the "tail wag" should be preserved. It represents one outlet for the limited competition possible in electric power supply. It is the very type of competition that, in regulated or quasinatural monopoly settings, the antitrust laws should be especially zealous to maintain, either to mitigate any undesirable effects of the market structure or the shortcomings of regulatory authorities. The preservation of this rivalry would seem to require the existence of a number of different buyers and sellers (although not at the expense of economic efficiency).

Id. at 895-96.

We think the same analysis holds true for the retail market. Competition in the market may be limited, but it is nevertheless entitled to protection under the antitrust laws.[117]

3. **The Geographic Market.** There remains the task of defining the geographic boundaries of the retail market. The Licensing Board concluded that no relevant geographic market could be found; it specifically rejected applicant's service area as the relevant market. (5 NRC at 888-89). We disagree.

In determining relevant markets, courts must "delineate markets which conform to areas of effective competition and to the realities of competitive practice." *Sargent-Welch Scientific Co.* v. *Ventron Corp.*, 567 F.2d 701, 710 (7th Cir. 1977), *cert. denied*, 439 U.S. 822 (1978), quoting *L.G. Balfour Co.* v. *F.T.C.*, 442 F.2d 1, 11 (7th Cir. 1971). The District Court in *Mishawaka II, supra*, a monopolization case involving a large Midwestern utility, found the application of this "practical approach" to be "relatively simple." The court explained its determination that defendant's service area constituted the relevant market:

> "The geographic location of the market is usually determined by an examination of the areas in which the particular firm actually competes or operates. If it concentrates its sales and service in one area, this area will normally be the relevant market." E. Kintner, *An Antitrust Primer, A Guide To Antitrust And Trade Regulation Laws For Businessmen*, pp. 102-103 (2d Ed. 1973).
>
> Here, defendant I & M has a clearly defined service area in Indiana and Michigan within which it sells electric power and energy at retail pursuant to franchises granted by the municipalities and townships. I & M has tariffs on file for those areas in the Public Service Commissions of Indiana and Michigan, pursuant to which it offers to sell electricity at retail to all interested buyers. Moreover, as the defendants have stated, no other public utility is allowed to sell electric energy at retail within this area.

465 F. Supp. at 1325.

Applicant protests the use of its service area to denote the geographic scope of the retail market. Its argument is two-pronged: if the test is "the area where applicant sells or can reasonably extend its retail sales," the whole state should be included in the market. If, on the other hand,

[117] *See, Midland, supra*, 6 NRC at 988.

"commercial reality" is used as a guidepost, the market should be broken down into small submarkets where competitive conditions are similar.[118]

We have no trouble in rejecting the contention that the whole state constitutes the appropriate geographic market. We think the Board below applied the correct principle in rejecting the same argument applied to the wholesale market:

> The entire state of Alabama would be an appropriate geographic market area only if wholesale suppliers in northern Alabama (TVA is the obvious entity involved here) could compete for retail loads in central and southern Alabama and Applicant could sell in the eleven northernmost counties of the state as well. Such is not the case.

5 NRC at 893. The Board noted that applicant does not attempt to sell power in the northern counties and that TVA is legally prohibited from selling power in most of the rest of the state. *Ibid.*[119] Given these circumstances, we see no reason to utilize the political boundaries of the state as the geographic limits for the retail market.

It is certainly true, as the applicant points out,[120] that the competitive situation differs in various parts of applicant's service area. But the same was true in *Otter Tail;* the different states involved had different franchise limitations and regulatory requirements, and certain municipalities had greater access than others to alternative transmission lines.[121] Nonetheless, the District Court in that case rejected the argument that each town in the defendant's service area be regarded as a separate geographic market.[122]

In *Midland* as well, the applicant argued that its service area could not be considered a relevant geographic market. In that case, the applicant proposed that an "open/closed" distinction be made; areas where competition was considered highly improbable were to be excluded from consideration.[123] The applicant here offered the same argument to the Board below.[124] We need not rehearse in detail the reasons why we rejected this argument in *Midland.*[125] We do think it worth repeating that, although

[118]APCO Reply Brief, 42-44. See also, APCO Reply Brief Below, 209-34.

[119]TVA is prevented by statute (16 U.S.C. § 831n-4(a)) from supplying power in areas not receiving power from TVA before July 1, 1957. Prior to that date, the only systems receiving power from TVA in south and central Alabama were the municipally-owned ones operating in the cities of Bessemer and Tarrant City. 5 NRC at 828, 829, 893.

[120]See APCO Reply Brief, 43.

[121]See 410 U.S. at 371.

[122]331 F. Supp. at 58-59. The District Court's market definition was apparently accepted by the Supreme Court. See 410 U.S. at 369-70.

[123]See 6 NRC at 978-79.

[124]See APCO Reply Brief Below, 228.

[125]See 6 NRC at 983-90.

different competitive factors might justify the division of a market into various submarkets:

> "submarkets are not a basis for the disregard of a broader line of commerce that has economic significance." This is especially true where the charge is that a firm has monopolized that broader line of commerce. [Applicant's] arguments in effect seek to focus our attention on those areas where door-to-door competition is now taking place and to have us ignore those areas where the company has already acquired dominance. To do so would be to manifest tacit acceptance of [applicant's] present market position as sacrosanct. This is simply not the case, legally or factually.[126]

We adhere to the approach taken in *Otter Tail, Midland,* and *Mishawaka II.* Those cases indicate that where a firm operates in a discrete service area and is charged with monopolizing retail sales in that same area, the service area may constitute the relevant geographic market for the purpose of antitrust analysis.

We add one last point. In many cases, the identification of a relevant geographic market is a crucial factor in the case because of its importance in determining a firm's market share (and hence, whether the firm possesses monopoly power). Although we find applicant's service area to be the relevant geographic market for the retail product market, our finding of monopoly power in the retail market is not solely dependent on market shares. See pp. 1071-1074, *infra.*

IV.

MONOPOLY POWER

Our determination that there are three relevant markets involved here must be followed by consideration of whether the applicant possesses monopoly power in these markets. This is so because business practices undertaken by those with dominance in the market may not be acceptable even though they would be legitimate if undertaken by those less powerful.[127]

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[126] 6 NRC at 990, quoting *United States* v. *Greater Buffalo Press,* 402 U.S. 549, 553 (1971) and *United States* v. *Phillipsburg National Bank,* 399 U.S. 350, 360 (1970).

[127] *Midland, supra,* 6 NRC at 913, citing *United States* v. *Aluminum Co. of America,* 148 F.2d 416 (2nd Cir. 1945); *American Tobacco Co.* v. *United States,* 328 U.S. 781, 812-14 (1946); *United States* v. *United Shoe Machinery Corp.,* 110 F. Supp. 295, 342-46 (D. Mass. 1953), *affirmed per curiam,* 347 U.S. 521 (1954); *cf. U.S. Steel Corp.* v. *Fortner Enterprises,* 429 U.S. 610, 612 fn. 1 (1977).

As we did with the Licensing Board's decision that the wholesale market is a relevant one (see pp. 1046-1047, *supra*), we adopt as our own that Board's decision that the applicant does indeed have monopoly power in the wholesale market.[128] Because, however, that Board believed no other markets to be relevant, it had no occasion to examine the extent of the applicant's control of those markets. We do so now.

A. Coordination Services Market

Once again we look to the teachings of *Midland* to help us determine whether the applicant here possesses monopoly power in the coordination services market. As we there explained (6 NRC at 998):

> The nature of the coordination services market does not . . . lend itself to an easy calculation of market shares. A utility is both buyer and seller in this market. Whether in any given time period it is a *net* buyer or a *net* seller is in part fortuitous, depending on operating conditions in its own and its neighboring power supply systems. Justice therefore undertook to show Consumers' possession of monopoly power in this market directly, by proving that its control of access to the market and its domination of power generation and transmission within it gives the company that power. This is a valid approach. (Emphasis in original).

Applicant's domination of power generation and transmission in its area of service is evident. The applicant is a vertically and horizontally integrated electric utility engaged in the generation, transmission and distribution of electricity.[129] As observed by the Board below, applicant's generating capacity in 1974 was 6,246 MW; it had additional planned capacity scheduled to be operative in 1979 of 2,380 MW.[130] It generates all of the power for its retail power needs. Disregarding the federally-owned capacity utilized in central and southern Alabama, applicant in 1974 held approximately 98% of the generating capacity in that area.[131]

In contrast, AEC had generating capacity in 1974 of only 137 MW, and a total planned capacity, scheduled for 1979, of 557 MW. It generates only a portion of the power requirements of its members.[132] As mentioned

[128]Applicant has excepted to the Licensing Board's treatment of its in-house distribution of bulk power as sales in the wholesale market. APCO Brief, 38-40. For the reasons given by the Board below (5 NRC at 890-92, 894-96) and by us in *Midland* (6 NRC at 990-97), we agree that such in-house distribution properly belongs in the market.

[129]5 NRC at 820.

[130]*Id.* at 821-22, 898.

[131]*Id.* at 898-99.

[132]*Id.* at 824-27, 898-99.

previously (see p. 1037, *supra*), none of the members of MEUA owns or operates any generating facilities.[133]

As for transmission, the applicant owns all transmission lines in the market over 115kv and controls all transmission facilities providing access to utilities outside the market area. With respect to lower voltages, applicant is also dominant. AEC owns 995 miles of generally low voltage transmission lines, only 15% of the amount owned by the applicant.[134] For their part, the members of MEUA own only 71 miles of low voltage lines.[135]

Although the above is only a rough description of the generating and transmission facilities in central and south Alabama, the dominant position of the applicant in either activity is readily apparent. Its dominance, particularly over the transmission facilities in south and central Alabama, places the applicant in a unique position to control access to the market for coordination services. By refusing to "wheel" power,[136] it is able as a practical matter to prevent the other utilities operating in the area from coordinating with the larger utilities outside it. This was aptly demonstrated at the hearing below.

During the course of the hearing, the question of how AEC might best coordinate its power generating expansion plans with the purchase of power from the applicant to meet AEC's projected power needs came up for consideration. In this connection, it was brought out that AEC was in the process of installing two 210 MW generating units on the Tombigbee River. This prompted the question of how the surplus capacity in those units, were they to be completed, could be disposed of by AEC if the applicant did not purchase it. The possibility of some third utility was suggested. But to dispose of the surplus capacity, it was conceded by applicant's witness that the transmission facilities of the applicant would have to be used.[137] If, for whatever reason, the applicant decided not to accommodate AEC, the cooperative would not be able to dispose of its surplus generating capacity.[138]

The applicant, however, claims in its brief that AEC is already connected to the system of the Georgia Power Company at the Walter F. George Lock and Dam. It argues that "there is no reason why AEC cannot, if it so desires, engage in power supply transactions with Georgia Power or through Georgia Power's system with Duke Power Company, South

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[133] *Id.* at 827.
[134] *Id.* at 900-01.
[135] *Id.* at 827.
[136] "Wheeling" is a term of art in the electric power industry, defined as the "transfer by direct transmission or displacement [of] electric power from one utility to another over the facilities of an intermediate utility." *Otter Tail Power Co.* v. *United States, supra,* 410 U.S. at 368.
[137] Harris, Tr. 25,443-44.
[138] *Id.,* 25,444-45.

Carolina Electric and Gas, Savannah Electric or Florida Power Corporation, all of which are interconnected with Georgia Power's system."[139] It also claims that AEC owns major transmission lines in close proximity to existing lines of Gulf Power Company and has other lines only a short distance from the South Mississippi Electric Power Association's system. The applicant suggests AEC can interconnect with these utilities and through them with others.[140] On the other side, Justice points out that "AEC has no interconnection to any utility other than Applicant."[141] This means that without the use of applicant's facilities, additional costly transmission lines would have to be built before AEC is able to coordinate power supply activities with Georgia Power.[142] From the standpoint of the nation's resources and the economy of the ratepayers that would be affected, constructing new lines when adequate facilities exist results in waste and places an additional, unnecessary burden upon ratepayers. In any event, there is no assurance that the other utilities mentioned would engage in the arrangements for the different type of coordination services which would be made possible were interconnection physically available.[143] We reject the applicant's position. It simply has failed to rebut the showing that its predominant control of transmission and generation gives it monopoly power over the sale of coordinated services in the relevant market area.

B. Retail Market

We wrote in *Midland* that the retail market lends itself to traditional market share analysis, with market shares being determined by calculating the amount of electric energy in megawatt hours (MWh) each utility sold to its retail customers. 6 NRC at 1009-1010. Applying these methods of determining market shares to the case at bar, the retail market in southern and central Alabama was divided (in 1972) as follows:[144]

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[139]APCO Brief, 29.

[140]*Ibid.*

[141]Justice Reply Brief, 30. We accept the validity of this statement inasmuch as applicant's own witness has testified that in any disposition of surplus power by AEC from its planned Tombigbee units, the transmission facilities of the applicant will have to be used. Harris, Tr. 25,444.

[142]An eight-mile extension of a 115 kv line with switching and other equipment to permit interconnection would cost from about $500,000 to $750,000. Brownlee, Tr. 25,663.

[143]According to AEC's counsel, AEC has "no idea whether Georgia [Power] would be willing to engage in it." App. Tr. 106.

[144]Wein, Direct, 67; Foltz, Tr. 12,841-43.

	MWh sold (x 1000)	% of market
Alabama Power Company	21,657	88
Municipal Systems	1,610	7
Distribution Cooperatives	1,335	5
Alabama Electric Cooperative	62	0

Applicant's share of 88% is clearly sufficient in normal circumstances to warrant the inference of monopoly power.[145] Applicant argues, however, that reliance on market shares is misplaced in this case. It claims that the economic characteristics of the industry (and its attendant regulation) result in higher market shares than would be found in a more conventional industry. Moreover, we are told, state and federal regulation of applicant's activities prevent it from possessing monopoly power.[146]

These arguments are nearly identical to those made by Consumers Power, and rejected by us, in *Midland*.[147] We have carefully reviewed that earlier ruling and its application to the facts of this case. We conclude that applicant's argument must fail; we find it possesses monopoly power in the retail market.

In the first place, the economic setting of the industry supports the finding that applicant possesses monopoly power. We have noted earlier that, while competition is legally permitted in Alabama, the economic barriers to the entry of new competitors in the industry are high indeed.[148] As we pointed out in *Midland*, high entry barriers reinforce the inference of monopoly power suggested by high market shares.[149]

More importantly, applicant's dominance of transmission and generation facilities further bolsters the finding of monopoly power. As the Board below noted, this dominance enables applicant to influence its present and

[145]See *Midland*, 6 NRC at 1010-11 and cases there cited.

[146]APCO Brief, 35-37; APCO Reply Brief, 52-53. Applicant advanced these arguments in the context of monopoly power in the wholesale market (no retail market having been found below). Although we deal with them here in the context of the retail market, our discussion and the arguments themselves apply with equal force to both markets.

[147]6 NRC at 1011-19.

[148]See p. 1062, *supra*.

[149]6 NRC at 1012-13, citing *Weber* v. *Wynne*, 431 F. Supp. 1048, 1054-56 (D.N.J. 1977); *United States* v. *United Shoe Machinery Corp.*, *supra*, 110 F. Supp. at 343-44; *Golden Grain Macaroni Co.*, 78 FTC 63, 163 n. 9, 180 (1971).

potential competitors' access to the basic inputs necessary for the production and sales of reliable and economical firm bulk power.[150] The dominance of what in essence constitute certain factors of production in the industry, viewed in conjunction with applicant's high market shares and the high economic barriers facing new competitors, would ordinarily compel a finding of monopoly power in the retail market.

It is at this point that the second thrust of applicant's argument presents itself. Monopoly power has long been defined as the power to control prices or exclude competitors.[151] Applicant would have us believe that the federal and state regulation of its activities precludes it from either controlling prices or excluding competitors and thus from possessing monopoly power.

We have already supplied a general answer to this argument (see pp. 1039-1042, *supra*). We need only particularize that answer by adding here that a vertically-integrated utility's ability to monopolize a retail market is not dependent on its ability to set its own retail rates. In *Otter Tail, supra,* the defendant cut off its retail competitor's supply of wholesale power. In *Mishawaka II,* the defendant threatened to curtail its competitors' supply and additionally charged them excessive rates for the wholesale power it did supply. In both cases, it was the dependence of the retail systems on a vertically-integrated competitor for their source of supply that enabled the integrated utility to monopolize the retail market.

There is no question in this case that applicant's competitors are wholly or partially dependent upon applicant for their supply of electric power. In such a situation, the courts in *Otter Tail* and *Mishawaka II* found defendants to be possessed of monopoly power despite the existence of the same federal regulatory scheme under which applicant operates.[152]

Nor do we believe the existence of the Alabama Public Service Commission (APSC) changes matters in this regard. For example, the Licensing Board found that the applicant unlawfully refused (or threatened to refuse) to sell wholesale power to AEC for resale to the military facility at Fort Rucker. 5 NRC at 942-45. In its appellate papers, applicant conceded that state law prohibits such a refusal.[153] Given the circumstances, it would

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[150]5 NRC at 899-901. Although the Licensing Board found monopoly power only in the wholesale market, we think it self-evident that the control of the basic components necessary to produce firm bulk power would yield the same result in the retail market.

[151]See, *e.g., United States* v. *Grinnell Corp., supra,* 384 U.S. at 571; *United States* v. *E.I. du Pont de Nemours and Co., supra,* 351 U.S. at 391; *American Tobacco Co.* v. *United States,* 328 U.S. 781, 811 (1946).

[152]In *Mishawaka II,* for example, the District Court described a mechanism by which the defendants were able to circumvent meaningful federal regulation of their wholesale rates in an effort to drive retail competitors out of business. 465 F. Supp. at 1327-29. We do not imply that the applicant here pursued a similar course of conduct, merely that if it had chosen to do so, federal regulation would not have saved applicant's competitors.

[153]APCO Brief, 80-81.

appear that AEC could have sought an order from the APSC which eventually might have resulted in AEC's being provided the power. But the state regulatory body was powerless to prevent applicant's initial refusal to deal. As the court in *Mishawaka II* pointed out, belated aid from regulatory bodies, often forthcoming only after extensive and costly litigation, is not an adequate antitrust remedy.[154] We think it self-evident that such an inadequate remedial mechanism is insufficient to deprive a regulated utility of monopoly power.

Having found that the applicant possesses monopoly power in each of the relevant markets, we now turn our attention to the charges that it has improperly wielded that power.

V.

MONOPOLIZATION

A. Situation Inconsistent with the Antitrust Laws

Our review of the Licensing Board's determinations on the various charges of monopolization leads us to observe that the Board did an unusually thorough job of marshalling, discussing and analyzing the sometimes complicated facts surrounding the various transactions. It examined closely each of the allegations of applicant's misuse of its monopoly power. It took into consideration the evidence bearing on each claim and the demeanor and credibility of the witnesses who gave pertinent testimony. On that basis, the Licensing Board viewed the evidence as sustaining only five of the specific monopolization charges.[155]

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[154]465 F. Supp. at 1329. See also, *Mishawaka I, supra,* 560 F.2d at 1325:

> Delay, combined with the multiple rate increases, could mean that the customer has been put out of business by his supplier-competitor. You cannot give refunds to a corpse.

[155]The instances of conduct which the Board found inconsistent with the antitrust laws relate to the following:

(1) Applicant's refusal to offer AEC fair coordination between 1968 and 1972. 5 NRC at 916-25.

(2) Applicant's insertion of contractual provisions in its various agreements with AEC and the municipal electric distribution systems precluding alternate sources of supply. *Id.* at 931-32.

(3) Applicant's inclusion in its contracts with preference customers of the Southeastern Power Administration (SEPA) requiring them to purchase all their additional power needs from the applicant. *Id.* at 933-37.

(4) Applicant's conduct with respect to AEC's efforts to provide power to Ft. Rucker. *Id.*

As mentioned at the outset of our opinion, all parties dispute the Licensing Board's conclusions. The applicant contends that the Board was correct in rejecting the bulk of the charges but that it erred in its five findings of anticompetitive conduct. The other parties argue the opposite. Each of them maintains the Board below did not go far enough. While agreeing with the Licensing Board's findings of anticompetitive conduct, these parties claim in various particulars that the Board erroneously decided that other activities were not anticompetitive.[156] It has thus become incumbent on us to examine the record on all these charges ourselves.[157]

at 942-45.

(5) Applicant's exclusion of smaller utilities from regional coordination. *Id.* at 946-957.

[156]A summary of the aspects of applicant's conduct which were found not to be inconsistent with the antitrust laws is found in the Board's Phase II decision dealing with remedy. 5 NRC at 1488-90. In capsule form, they cover the following:

1. The various types of coordination for economy and reliability which applicant obtained as a member of the Southern Company pool.
2. Applicant's opposition through use of judicial and administrative forums to AEC's obtaining REA loans for the construction of new generation and transmission lines.
3. Applicant's wholesale rate reductions to AEC occurring at times when AEC was considering installation of generating facilities.
4. The 1972 Interconnection Agreement between applicant and AEC (with elimination of the "protective capacity" provision).
5. Applicant's conduct relating to ownership participation by AEC and MEUA in the Farley plant.
6. Applicant's conduct relating to the generating plant proposed to be constructed by the City of Dothan, Alabama.
7. Applicant's conduct in opposing construction by SEPA of high voltage transmission lines.
8. MEUA's allegations of "price squeeze" practiced by applicant.
9. Applicant's use of the courts and administrative agencies.
10. Other allegations of anticompetitive conduct by the applicant such as offers to purchase various distribution systems, attempted acquisition of certain transmission lines, and efforts to serve a new shopping center near Enterprise, Alabama.

[157]We should note there that, while we generally accord deference to trial board findings, it is settled law that we are not held to the "clearly erroneous" standard of review employed by the federal courts of appeal. Where our review of the evidentiary record convinces us that a different result is warranted, we are free to substitute our judgment for that of the trial board. See, *e.g.*, *Midland, supra*, 6 NRC at 1022-23; *Duke Power Company* (Catawba Station, Units 1 and 2), ALAB-355, 4 NRC 397, 402-05 (1976); K. Davis, *Administrative Law Treatise* (2d Ed. 1980), § 17.16.

We have done so, but from a somewhat different perspective than that of the Licensing Board. This stems principally from two factors. The first is that unlike the Licensing Board — which found the applicant to possess monopoly power only in the market for wholesale power — we have found that applicant has monopoly power in the coordination services and retail power markets as well. This means that we must look upon the applicant's conduct as that of a dominant business enterprise wielding monopoly power over the entire range of activities in which it engages, and judge it under a harsher light than that of a less dominant business concern. As we stated on another occasion, judicial and FTC rulings teach that "the actions of a dominant business enterprise have to be tested against a more stringent standard than applies to actions of smaller concerns in highly competitive markets."[158]

The other principle affecting our view of the record is that the evidence must be viewed in its entirety and not with the eye focused only on isolated segments as though they were independent of each other. For the courts have stressed

> the importance of viewing the evidence as a whole to give the antitrust plaintiff the full benefit of his proof, rather than tightly compartmentalizing the case and wiping the slate clean after considering each piece of evidence.[159]

In this connection, the applicant's opponents accuse the Licensing Board, in denying all but five of their claims of misuse by the applicant of its monopoly power, of giving inadequate attention to the pattern of anticompetitive conduct indicated by the record. We agree with their position on this point.

Our own examination of the record with these two principles at the fore suggests strongly that it would be permissible for us to find any number of additional alleged instances of misconduct to have been part of an anticompetitive pattern and thus subject to obloquy. But weighing the record is in no small part a matter of judgment. We must recognize and accept that the Licensing Board heard the witnesses and evaluated their demeanor at first hand; we have only the printed word on the cold page before us. In these circumstances, we are unpersuaded that there is sound cause to substitute our own judgment on most of the conclusions reached below. The licensing boards are, as we have said before, this agency's

[158] *Midland, supra,* 6 NRC at 913.
[159] *Id.* at 914, citing *United States* v. *Empire Gas Corp.*, 537 F.2d 296, 299 (8th Cir. 1976), *Cert. denied,* 429 U.S. 1122 (1977).

principal fact finders.[160] We thus accept the Licensing Board's findings except in two areas where the record compels findings of a situation inconsistent with the antitrust laws: the first deals with the applicant's selective use of low wholesale rates to discourage AEC from constructing its own generating stations; the second concerns the applicant's refusal to extend an ownership interest in the Farley plant to AEC. We now deal with these matters in order.

1. **Low Wholesale Rates.** The Licensing Board examined four instances in which APCO was alleged to have lowered its wholesale rates for the purpose of preventing AEC from installing generating units. The Board rejected the allegations, finding no anticompetitive conduct in each instance. Specifically, the Board concluded:

(1) A 1941 rate reduction to a number of utilities, which came at a time when certain distribution cooperatives were forming AEC and were seeking an REA loan to construct new generation and transmission facilities, was legitimately motivated by applicant's desire to reduce its number of different wholesale rates and not to forestall self-generation by AEC. 5 NRC at 908-09.

(2) A 1946 rate reduction offer to AEC, made after AEC applied for an REA loan to construct a new steam plant and associated transmission lines, was to allow applicant to continue selling wholesale power to AEC and "to dissuade AEC from proceeding with its plans to construct [a generating plant and transmission] which applicant considered uneconomical and wasteful duplication of its existing facilities;" was made in good faith with the encouragement of REA; and was not anticompetitive in intent or motive. *Id.* at 910.

(3) A 1950 offer to AEC of a rate reduction, after AEC had again taken action to obtain REA funds for the construction of another version of its earlier planned steam plant, "had the distinct purpose of improving the reliability of AEC's electric system," and did not represent "anticompetitive conduct with the clear purpose of maintaining a monopoly in self-generation." *Id.* at 911.

(4) A 1958 rate reduction to cooperatives and municipals (the so-called "Coosa" reduction) was essentially forced upon applicant as a condition of applicant's receiving licenses to develop hydroelec-

[160]See *Catawba, supra*, 4 NRC at 404.

tric projects on the Coosa River, and was not anticompetitive. *Id.* at 912-13.

With respect to the Coosa rate reduction, we are satisfied with the findings made below. We do, however, take a different view of the three earlier reductions. We believe they were instituted for the purpose of preventing AEC from developing its own generation, and as such were inconsistent with the antitrust laws.

As a preliminary matter, we address the Licensing Board's treatment of the *Noerr-Pennington* doctrine. That doctrine, established by the Supreme Court in *Eastern Railroad Presidents Conference* v. *Noerr Motor Freight, Inc.*, 365 U.S. 127 (1961); and *United Mine Workers of America* v. *Pennington*, 381 U.S. 657 (1965), essentially renders immune from antitrust liability actions which seek to influence legislatures, courts, and other governmental bodies even though they are undertaken for anticompetitive purposes. A third case, *California Motor Transport Co.* v. *Trucking Unlimited*, 404 U.S. 508 (1972), limited the doctrine somewhat by providing that sham attempts to influence official action are not immune.[161] As the Board below recognized in an order issued during the Phase I hearing,[162] evidence of conduct designed to influence governmental action can be used for two purposes. First, a party is always free to show that the conduct falls within the sham exception to *Noerr-Pennington*. Second, according to the principles set out in *Pennington* footnote 3, a party can use exempt activities as evidence of general anticompetitive intent in order to shed light on nonexempt activities.[163]

In this case, there is no question that applicant actively used legal and administrative proceedings in attempts to prevent AEC from installing its own generation.[164] Applicant's opponents argued below that this use of the legal process fell within the sham exception (and thus was itself inconsistent with the antitrust laws), and that, even if such activity is exempt from

[161]For example, good-faith litigation may be exempt from antitrust liability, but the repetitive filing of frivolous legal claims for the sole purpose of harming a competitor is not. See, *e.g.*, 404 U.S. at 513; *Otter Tail, supra*, 410 U.S. at 380.

[162]LBP-75-69, 2 NRC 822 (1975).

[163]381 U.S. at 670 n. 3. The footnote reads as follows:

> "It would of course still be within the province of the trial judge to admit this evidence, if he deemed it probative and not unduly prejudicial, under the 'established judicial rule of evidence that testimony of prior or subsequent transactions, which for some reason are barred from forming the basis for a suit, may nevertheless be introduced if it tends reasonably to show the purpose and character of the particular transaction under scrutiny.'"

[164]See 5 NRC at 902-08.

antitrust liability, the Board should derive from it evidence of applicant's anticompetitive intent. The Board found the activity protected.[165] It further ruled that "there is no room for application of *Pennington* footnote 3 regarding the admissibility of immunized transactions to shed light on the 'purpose and character' of nonimmunized transactions, because the challenged litigation was both immunized and itself not anticompetitive under the antitrust laws."[166]

We can readily agree with the Board's determination that the use the applicant made of administrative and judicial process is protected under *Noerr-Pennington.* The Board's handling of *Pennington* footnote 3 is quite another matter. We read that footnote as plainly allowing the admission of evidence concerning "immunized" transactions where such evidence sheds light on nonimmunized transactions.[167] As applicant itself admitted, protected *Noerr-Pennington* material may be used "to show purpose or character of other evidence under scrutiny."[168]

We now turn to the matter of applicant's low wholesale rates. The Licensing Board was unable to find that the rate reductions "represented anitcompetitive conduct with the clear purpose of maintaining a monopoly in self-generation."[169] We think applicant's otherwise protected use of judicial and administrative proceedings sheds a good deal of light on those rate reductions. It seems clear to us that applicant was strongly opposed to AEC's installation of generation. Nor do we doubt that the institution of low rates could have served to undermine AEC's efforts in this regard. All this added to the timing of the reductions in question (each occurred at a time when AEC was seriously pursuing new self-generation options) leads us to the compelled inference that the reductions were motivated with the intent of discouraging AEC's self-generation.

Interestingly enough, the Board below agreed that a purpose of the 1946 reduction was to prevent AEC from pursuing a proposal to build a 23 MW plant at Gantt. Although the Board found that the 1941 and 1950 reductions were motivated by applicant's desire to lower the number of rates in its rate structure and to improve the reliability of AEC's system (see p. 1072, *supra*), we find the timing of the reductions more than a

[165]*Id.* at 902-08, 940-41.
[166]*Id.* at 941 (reference omitted).
[167]See *Schenley Industries, Inc.* v. *New Jersey Wine and Spirit Wholesalers Ass'n.,* 272 F. Supp. 872, 886 (D.N.J. 1967), wherein the District Court wrote:

> In a footnote to the *Pennington* opinion, the Supreme Court did leave open the use of evidence on protected lobbying activity in the manner Schenley proposes, namely, to demonstrate anticompetitive intent.

[168]APCO Reply Brief Below, 286.
[169]5 NRC at 911.

coincidence. We can agree with the Licensing Board that the applicant's use of the governmental processes available to it was conduct protected under *Noerr-Pennington*. But the full circumstances surrounding applicant's rate reductions, including its history of legal opposition to AEC generation, compel the conclusion that the reductions were part of a long campaign to forestall AEC from installing its own generating capacity.

Our only difficulty in reaching this conclusion stemmed from unease at adopting the notion that AEC could suffer a legally cognizable injury from having a low rate offered, not to one of its competitors, but to itself. Unlike the usual situation, where the offended party is helpless in the face of price concessions offered either to its competitors or to its potential customers, AEC here had the power to defuse the applicant's tactic. It simply could have declined to let the opportunity to purchase power at a reduced rate deter it from building its own generating capacity.

The short answer to our concern is that, owing to the applicant's monopoly position, AEC had no *practical* alternative to accepting the reduced rate and dropping its plans for expansion. Not only its own short term fiscal health — a critical matter to a business lacking a monopolist's power — was at stake; but a refusal of the applicant's offer would have brought down upon it the objections of the REA and others who might point out that the insistence on going ahead appeared to involve an unnecessary duplication of effort.

What we are left with, then, is the conclusion that these lowered rates were the opening salvo in the pattern adhered to through the years in which the applicant sought to forestall AEC from installing its own generating capacity, and to keep AEC as a captive customer — even at the cost of short-term profit — rather than allow it to develop as a competitor, thus assuring applicant's long-term health. As already indicated, it might be possible to build on this to find that a great many more instances of anticompetitive conduct fit into this same pattern. We decline, however, to do so, giving due deference to the analysis of the Board below.

One final matter remains. The Licensing Board found, in regard to the 1946 reduction, that applicant was properly motivated by a desire to prevent "uneconomic and wasteful duplication." (5 NRC at 910.) In the first place, we do not understand why AEC's construction proposal necessarily involved a duplication of applicant's facilities. Applicant has built numerous generating facilities; if its chief concern was duplication, it could have staggered AEC's proposed construction in with its own plans. More important, we do not believe an ostensible desire on the part of a monopolist to avoid "wasteful duplication" constitutes a legitimate defense under the antitrust laws to charges that the monopolist has prevented prospective competitors from entering a market. The argument that it does is merely another version of the regulated industry defense we addressed earlier (see pp. 1039-1042, *supra*). An electric utility may prefer to avoid competition, but it cannot accomplish this goal through anticompetitive means.[170]

2. **Denial of Ownership Access to Farley**

a. The other count on which the record compels us to disagree with the Licensing Board involves the applicant's alleged denial of ownership access to the Farley units. The Board below declined to find that the applicant had denied such access to AEC. According to that Board, there was no "hard evidence substantiating" such a charge; that on the contrary Mr. Farley, applicant's President, "made it quite clear in his testimony before the Board that Applicant does not take the position that it would not sell ownership." 5 NRC at 929.

With all due deference to the Licensing Board, we construe the record differently. Our assessment of all the surrounding evidence persuades us that although the applicant never explicitly stated it was absolutely rejecting the possibility of selling an ownership share in Farley to AEC, it fully intended not to make such a sale unless forced to do so.

From at least 1969, it was applicant's policy to maintain sole ownership in the Farley plant. This was made clear in an internal confidential memorandum of the company circulated among the officers and attorneys representing it in negotiations with AEC.[171] That memorandum stated in unequivocal language: "The company is unalterably opposed to potential demand from one of more distribution cooperatives, or from AEC, for part ownership in the SEALA nuclear plant."[172] This policy remained essentially unchanged over the years.[173] Thus, it is not surprising to find that even

[170]See also *Davis-Besse, supra*, 10 NRC at 323-27.

[171]D.J. 6040; Vogtle, Cross, Tr. 23,135.

[172]D.J. 6040, p. 4. "SEALA" was the earlier name for the Farley plant.

[173]On April 6, 1971, shortly after AEC expressed an interest for joint ownership of the plant, the applicant filed Amendment No. 13 to the license application for construction of the Farley units. The amendment stated: "The plant is planned to be wholly owned by Alabama Power Company and is not planned for construction or operation as a joint venture with any other entity." See Justice Brief, 79. In this regard, James H. Miller, Jr., a senior vice-president of Alabama Power who participated in various negotiations and discussions with AEC concerning interconnections and joint ownership participation in Farley, testified:

CHAIRMAN GLASER: Mr. Miller the company has never been in favor of a joint ownership arrangement with AEC to your knowledge; has it?

THE WITNESS: Not to my knowledge, no. sir.

Miller, Tr. 21,476.

though AEC expressed interest in acquiring a share in the Farley plant as early as 1971,[174] some two years later applicant was still arguing for the sale of unit power.[175] To be sure, applicant's representatives met with AEC on repeated occasions to discuss the subject of access to Farley power,[176] but the meetings did not progress much beyond the exploratory stage. During this period, the applicant's main efforts were directed not so much towards seeking an acceptable agreement on the joint ownership of the plant but in getting AEC to agree to the purchase of wholesale or unit power. The result was that when these hearings began in late 1974, the parties were far from reaching agreement on joint ownership of Farley, even in principle.[177] The effect of applicant's actions was to deny AEC reasonable access to Farley.

In holding that the applicant acted to deny AEC an ownership in the plant, we have fully considered the testimony of Mr. Farley. But unlike the Board below, we find in it no support for the proposition that the applicant did not have a position against selling an ownership share in the plant. Rather, we find it to point forcefully the other way.

For its conclusion that the applicant had no position against selling an ownership interest in Farley to AEC, the Board below relied on two statements made by Mr. Farley at the hearing. On one occasion, Mr. Farley was asked whether his company was willing to provide the municipalities

[174]Letter from AEC to Mr. Farley dated April 27, 1973. App. Exh. BMG-21.

[175]As late as November 26, 1973, AEC's overtures toward acquiring an ownership interest in the Farley plant were being met by a recitation of claimed barriers against any kind of joint ownership arrangement. AEC Exh. 32. It is significant that the existence of problems claimed to be serious obstacles to joint ownership of the Farley plant were not raised until some two years after AEC's expression of interest in the plant. In 1974, the applicant was still resisting the sale of a share in Farley to AEC. On October 29 of that year, applicant's counsel Mr. Balch wrote to AEC's counsel Mr. Boskey outlining the applicant's understanding of the positions of the parties expressed at a meeting which had been held earlier among representatives of both organizations. In that letter, applicant's counsel continued to urge that "the most fruitful approach to this matter from Alabama Power's point of view is to consider a unit power approach which avoids the complex problems which would arise from any attempt at this time to restructure the ownership of the Farley units." App. Exh. 173 at pp. 11-12. Earlier, on August 16, 1973, Mr. Farley had written to AEC urging that it purchase "power from a mix of the company's generation under applicable rate schedules and, thereby, in effect, have access to the Farley plant." The letter went on to indicate that, inasmuch as AEC indicated a desire to participate specifically in Farley, the applicant invited discussions to explore the possibility of unit power purchase by AEC. AEC Exh. 30.

[176]5 NRC at 929.

[177]By late 1974, the parties had not yet reached the stage of negotiating over firm proposals. On June 20, 1974, AEC wrote to Mr. Farley to raise several matters including the desire for a meeting to resume discussion on a joint ownership arrngement for the Farley plant. AEC Exh. 35. Mr. Vogtle responded for the applicant. On the subject of joint ownership, the response was no more than a bland invitation to discuss the matter at the next meeting with the request that AEC "furnish any definitive proposal to the Company for review" before the next meeting. AEC Exh. 36. By October of that year the applicant was continuing in its pursuit of a unit power arrangement with AEC. See fn. 175, *supra*.

and AEC access to Farley units by means of ownership participation. Mr. Farley's response was:

> The matter as to ownership has been discussed with representatives of the cooperatives and to a certain extent, the municipals, and the company is in this position, that we have not taken the position that we would not sell ownership.[178]

Later in the hearing, Mr. Farley was again asked about the request of AEC for an ownership share of the Farley plant. In response to this question by a Licensing Board member, the following transpired:

> [MR.FARLEY]: We have been in negotiations with the Cooperative in ways that have certainly been explored here in this hearing heretofore. I don't consider the sale of the company's property or ownership in the plant or something of that nature quite in the same light that I do the offering of the utility service or utility coordination. We have not, obviously, reached agreement with the cooperative on the sale of a portion of the plant but it is not inconceivable that we might.
>
> MR. MILLER: What does that mean, Mr. Farley?
>
> THE WITNESS: It means, sir, that as of this point in time, as I have answered questions heretofore, Mr. Miller, that we don't have a policy that we would not sell a portion of a plant because we may. We think it's got all kinds of problems with it.[179]

True enough, one could read these statements to convey the thought that the applicant has no position *against* the sale of an ownership interest in the plant.[180] But to succumb to this would be to be misled by the applicant's judicious phrasing of its answers in the double negative. That tactic cannot obsure the fact that the company has steadfastly avoided indicating directly that it *would* share ownership. When other testimony of Mr. Farley is considered, it clearly appears that the applicant did not intend to sell. This becomes even more patent when Mr. Farley's statements are viewed alongside the company's dealings with AEC after the time in 1971 when AEC expressed interest in acquiring an ownership interest in the plant.

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[178]Farley, Cross, 19,185.
[179]Farley, Cross, 20,599.
[180]At another instance during the hearing, Mr. Farley was asked about the company's policy toward joint ownership of the plant with others. To this, Mr. Farley's reply was that "there just simply isn't a policy on it." Farley, Cross, 19,198-99. We find this answer inconsistent with the 1969 policy statement and the action subsequently taken by the applicant.

The crucial testimony came after the exchanges relied on by the Licensing Board. Mr. Farley was asked by counsel for the Department of Justice whether the applicant was willing to offer ownership participation in the Farley plant to AEC. Mr. Farley responded:

> I find it difficult to answer the question yes or no....

When asked by the Licensing Board Chairman for an explanation, Mr. Farley replied:

> If this Board were to impose a license condition which were to be upheld that the Company should sell an interest in the nuclear plant, then we'll sell an interest in the nuclear plant.[181]

Thus, when pressed on the point of the applicant's willingness to enter into a joint ownership agreement with AEC, Mr. Farley's testimony was that the company would do so — but only under compulsion by this agency. Stated in more direct terms, Mr. Farley was saying in effect that the applicant had no intention of voluntarily entering into an arrangement with AEC for joint ownership of the plant.

Mr. Farley's last statement is even more revealing when considered in the context of the 1969 statement in which the policy of the company is expressed as being "unalterably opposed to sharing in the ownership of the plant with AEC or with any one or more of the cooperatives."[182] Viewed in that light, it becomes clear that the company had a position: to resist to the last selling an ownership share of the plant to AEC.[183]

b. Our inquiry does not end here. The next step we must take is to determine whether applicant's conduct respecting its refusal to sell an ownership interest in the Farley plant constituted anticompetitive action. For the reasons which follow, we hold that it does.

In Part IV of our decision, we found that the applicant possessed monopoly power in the wholesale and retail markets for electricity in

[181]Farley, Cross, 27,949-50.

[182]See p. 1081, *supra*.

[183]The question of whether applicant denied MEUA ownership access is a much closer one. Nothing in the record indicates that applicant would have viewed an ownership request from MEUA more favorably than that from AEC. On the other hand, after reviewing the testimony of Mr. St. John carefully, it seems clear to us that MEUA did not pursue ownership access as actively as did AEC. See Tr. 4547-98. We are particularly concerned with the timing of MEUA's request, which appears to have come well after this proceeding got under way. Tr. 4551-4580.

We believe resolution of this matter is unnecessary to our disposition of the case. We can assume that if a timely request was made, it would have been rejected. The key issue remains whether MEUA is entitled to ownership access. We discuss that point later (see pp. 1124-1125, *infra*).

central and south Alabama and in the coordination services market in that area. Being possessed of monopoly power, the applicant is precluded by Section 2 of the Sherman Act from willfully using it to preserve or extend its monopoly, to foreclose actual or potential competition, to gain competitive advantage or to destroy competitors. Moreover, it is not only full-fledged violations of the antitrust laws that are of concern in these licensing proceedings. Section 105c of the Atomic Energy Act, which governs the proceeding here, condemns as well conduct which runs counter to the policies underlying those laws.[184]

Viewed against these limitations on permissible conduct by one who is a monopolist, we have no hesitancy in concluding that the applicant's actions in denying AEC a joint ownership share in Farley constituted anticompetitive behavior. The evidence leaves no doubt in our minds that the actions of the applicant in this regard were deliberately directed toward avoiding sharing in the ownership of the plant for fear that granting AEC an ownership interest in the plant would lead to erosion of the applicant's wholesale and retail business. As candidly put by Mr. J. H. Miller, Jr., applicant's senior vice-president:

> Should intervenors be allowed to acquire a portion of the Farley Nuclear Plant, extending the utilization of subsidized financing, it could bring about an inherently unfair competitive position between them on the one hand and Alabama Power on the other. It could, in fact, in the long-term place Alabama Power's competitive position in jeopardy to such a point that Alabama Power would no longer be viable.

Miller, Direct, 150.[185]

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[184] *Midland, supra,* 6 NRC at 1019; see pp. 1044-1046, *supra.*

[185] The testimony of Mr. Farley was to the same effect:

> Q. [Mr. Leckie, Justice Counsel]: You were concerned, though, in the time period 1969 to 1971 with the possibility that your wholesale business might be eroded if you were to sell a share of the Farley Unit to Alabama Electric and/or to the municipal systems?
>
> A. [By Mr. Farley]. We were concerned that the differentials through these facts and financing costs might cause a problem, yes, sir.
>
> Q. Were you concerned with a possible erosion of retail business at that time?
>
> A. Yes, sir, because all along has been the concept in Alabama Electric Cooperative's request that we wheel for them where ever they want. And that would include retail. That thread has been through many of our discussions and negotiations and that remained then and it remains now.

Although the possible future loss of business is undoubtedly of legitimate concern to any business enterprise, it cannot be used by a monopolist to justify conduct designed to preserve or enhance its dominant position in the competitive market. At the very least, if not a violation of the antitrust laws, such conduct runs counter to the policies underlying those laws.

That observation unquestionably applies to the situation here. Applicant's 1969 policy statement and the testimony of its two senior officers leave no doubt as to the company's short and long-range objectives in refusing to share in the ownership of Farley: the preservation of its dominant power in the wholesale and retail markets for electricity in central and south Alabama. That objective, as we have seen, is one that is condemned by Section 105c and the antitrust laws referred to therein. This being so, it follows that action undertaken by the applicant toward that end is no less unacceptable under the law.

B. MEUA's Appeal

MEUA was denied a remedy below because the Board found that there was no "significant actual or prospective competition between [MEUA and applicant] at the retail distribution level." 5 NRC at 961.[186] Implicit in this denial was the Board's view that MEUA was also not a competitor in the wholesale market.[187] MEUA's appeal is thus essentially double-barreled; it contends both that the rejection of the retail market was incorrect and that it was wrongfully excluded from the wholesale market.

As we explained earlier (see pp. 1059-1068, *supra*), we disagree with the Licensing Board's rejection of the retail market. Before we analyze the effect of this finding on MEUA's case, we turn to the claim that the Licensing Board erroneously excluded MEUA from the wholesale market.

CHAIRMAN GLASER: In fact, hasn't it been the case that the company's been concerned about Alabama Electric Cooperative taking away Alabama Power Company's customers since the inception of the cooperative?

THE WITNESS: Well, sir, I wouldn't say, Mr, Chairman, since the inception of it because this didn't really get to be, well, several years — in the early days of its — in the late '40's, perhaps, would be a better time. I think the cooperative was organized about '41 or '42, or something like that and it was some years after that before the west —

CHAIRMAN GLASER: In any event, for the last 20 years the company has been concerned about it?

THE WITNESS: Yes, sir.

Farley, Cross, 20,802-04.

[186] In the ensuing discussion, the term MEUA refers to both the organization collectively and its members singularly.

[187] See 5 NRC at 1484 n. 5.

1. **Wholesale Market.** Although the Licensing Board determined that there was a relevant wholesale market in central and southern Alabama, it excluded MEUA from the remedial hearing on the grounds that MEUA was not an actual or potential competitor in the market.[188] MEUA, Justice and staff dispute this ruling, arguing on appeal that the municipals are potential competitors. They argue that this is true because the municipals are on the edge of the market, that applicant's activities in the past have discouraged their entrance, and that such entrance is feasible if the municipals are granted a share of the Farley facility.[189] MEUA relies on a second string to its bow. In the alternative, it argues that its members are currently in competition in the wholesale market. We deal with this latter argument first.

a. MEUA advances two bases on which it would have us find that it is presently in actual competition in the wholesale market. It first notes that although it now does not engage in selling power at wholesale, one of its members, Riviera Utilities,[190] at one time provided wholesale service in Baldwin County. It then claims that Riviera was forced out by applicant's anticompetitive conduct. To prevent the applicant from benefiting from its wrongdoing, MEUA's argument is that we should look upon the market in terms of the situation existing at the time Riviera engaged in wholesale service and not the present. Secondly, MEUA argues that its decision to purchase wholesale power instead of supplying its own needs through self-generation is a form of present wholesale competition.

We need not devote much attention to the argument that the exercise of a decision to "make-or-buy" is an indication that actual competition for the sale of wholesale power exists. All MEUA's decision to buy tells us on the record of this case is that it is a wholesale customer of the applicant. Without any generating capacity of its own, we simply do not believe that MEUA as a buyer of electricity at wholesale is in actual competition with a selling entity.

The question of MEUA's past role in the market is a more complicated matter. Although Riviera Utilities lost its last wholesale customers during the course of the proceeding below,[191] there is no dispute that Riviera at one time provided wholesale service to other retailing entities. Indeed, in its description of wholesale competition, the Licensing Board included

[188] *Ibid.*
[189] MEUA Brief, 22-41; Justice Brief, 54-61; Staff Brief, 23-26, 40-42.
[190] Riviera Utilities is the name of the municipally-owned utility in the town of Foley.
[191] MEUA Brief, 25; 5 NRC at 828.

references to competition between Riviera and applicant.[192] Nonetheless, the Board excluded MEUA from the market without explanation.

Although the Licensing Board did not deal directly with Riviera's role in the wholesale market, it did limit sellers in the market to "those entities generating and providing bulk electric power to distribution entities." 5 NRC at 890. Riviera, it should be pointed out, was not a generating entity. MEUA challenges any suggestion that generation is a precondition to being in the market; it claims the market should include all entities selling bulk power to distribution systems.[193] The fact that Riviera no longer sells power at wholesale, we are told, is not relevant, if Riviera is excluded from the market, "any monopolist would be immune from antitrust liability upon accomplishing destruction of its rival."[194]

We can agree with MEUA up to a point. Theoretically, ownership of generation need not be a prerequisite to entrance in the wholesale market. And certainly any destruction of a competitor is a fact we could hardly ignore. But our assessment of the record simply does not comport with that of MEUA.

The town of Foley acquired Riviera Utilities in 1941.[195] Riviera at the time had three wholesale customers in south Baldwin County: the towns of Robertsdale and Fairhope, and the Baldwin County Electric Membership Cooperative. It supplied its wholesale and retail power requirements, in 1941 and at all times afterwards, through wholesale purchases from applicant. Eventually, all of Riviera's wholesale customers decided to take service from applicant instead.

Although MEUA would have us believe that applicant was responsible for Riviera's loss of its wholesale customers, the record indicates otherwise. We find that Foley's role was purely that of a middleman; it purchased power from one party and sold it at a markup to another. Its wholesale customers were prevented by contractual barrier from dealing with applicant directly; when the barriers were removed, the customers elected to receive their power from applicant. In this regard, it should be noted that applicant charges uniform wholesale rates throughout the state; it did not lower its rates to attract the new business. Applicant further claims[196] — and the record does not indicate otherwise — that it received no additional revenue from its new customers; it simply sold the same amount of power at the same price without going through a middleman. When questioned

[192] 5 NRC at 895, citing, *inter alia*, St. John, Direct, 10-14; DJX 4298, 4301, 4308-4311; Tr. 23,477-23,487.
[193] MEUA Brief, 24.
[194] *Id.* at 26.
[195] APCO Reply Brief, 45.
[196] APCO Reply Brief, 46 n. 312.

about the loss of Riviera's wholesale customers, Mr. St. John was unable to point to any conduct on applicant's part in taking over service to Riviera's customers that could be considered wrongful.[197] Nor did he indicate that Riviera sought cheaper sources of bulk power elsewhere (if any were in fact available). In these circumstances, we are simply unwilling to say that applicant contributed to the destruction of its wholesale rival. Common sense would seem to indicate that a wholesale supplier that does nothing more than buy power from one supplier and sell it at a higher price to distributors will be unable to remain in existence if their customers can deal directly with the supplier.[198] Riviera having lost its customers through operation of market forces, we find no basis for faulting the applicant in this regard. This being so, whatever the competitive situation may have been when Riviera was a seller of wholesale power, the fact is that MEUA is not now an actual competitor in the wholesale market.

b. As mentioned earlier, Justice, NRC Staff, and MEUA all argue that MEUA is a potential competitor in the wholesale market. Our attention is directed to any number of court decisions dealing with potential competition as it affects mergers under Section 7 of the Clayton Act.[199] Applicant questions the propriety of relying on merger cases to determine whether MEUA's members are potential competitors at the wholesale level.[200] We need not decide this issue, for we do not believe MEUA qualifies as a potential entrant even under the principles enunciated in the cases it cites.

The reasoning for our rejection of the notion that MEUA is a potential entrant to the wholesale market is founded upon our assessment of its ability to enter the market. We accept, for the purposes of argument, MEUA's contentions that it is eager to enter the market, that it is in a similar line of commerce, that actual penetration of the market is unnecessary, and that MEUA is the most likely new entrant.[201] Nonetheless, we read the cases as requiring a showing that MEUA is either (1) capable of entering the market on its own, or (2) currently influencing competitive conditions in the market. MEUA has not made either showing.

[197]See Tr. 3683-94.
[198]In this connection, see *New England Power Co.* v. *Federal Power Commission*, 349 F.2d 258, 260 (1st Cir. 1965), wherein the F.P.C. noted that the prevailing industry practice was for the middleman to be eliminated and that the Commission could see no reason why the middleman in the case should not be eliminated.
[199]*E.g., United States* v. *Marine Bancorporation*, 418 U.S. 602 (1974); *United States* v. *Falstaff Brewing Corp.*, 410 U.S. 526 (1973); *Federal Trade Commission* v. *Proctor & Gamble Co.*, 386 U.S. 568 (1967); *United States* v. *Penn-Olin Chemical Co.* 378 U.S. 158 (1964); and *United States* v. *El Paso Natural Gas Co.*, 376 U.S. 651 (1964).
[200]APCO Reply Brief, 48 n. 322.
[201]See MEUA Brief, 31-41.

A look at the cases helps illuminate the nature of these requirements. In *Marine Bancorporation*,[202] the acquisition of a Spokane, Washington bank by a Seattle bank seeking to penetrate the Spokane market was allowed; the Supreme Court found that the purchase did not eliminate the Seattle bank as a potential competitor in the Spokane market because the bank lacked other feasible means of entering the market. The Court thus allowed the acquisition to take place. 438 U.S. at 632-639.

In *Falstaff*,[203] the Supreme Court reversed and remanded a decision approving a national brewery's purchase of a New England brewery. The District Court found conclusive the testimony of witnesses for the acquiring firm indicating that it would not have entered the New England market by any other means. The Supreme Court thought otherwise:

> The specific question with respect to this phase of the case is not what Falstaff's internal company decisions were but whether, given its financial capabilities and conditions in the New England market, it would be reasonable to consider it a potential entrant into that market . . . . [I]f it would appear to rational beer merchants in New England that Falstaff might well build a new brewery to supply the northeastern market then its entry by merger becomes suspect under § 7. The District Court should therefore have appraised the economic facts about Falstaff and the New England market in order to determine whether in any realistic sense Falstaff could be said to be a potential competitor on the fringe of the market with likely influence on existing competition.

410 U.S. at 533-534.

In *Procter & Gamble*,[204] the acquisition of a bleach manufacturer by a company specializing in household products was disallowed. The Supreme Court found, *inter alia*, that the acquisition would eliminate the acquiring company as a potential competitor in the market for bleach. There was no evidence indicating that the acquiring company intended to enter the bleach market *de novo;* however, the Court found it to be a potential competitor on the ground that *de novo* entry was feasible and that the threat of *de novo* entry exerted "considerable influence on the market.' " 386 U.S. at 580-581.

In the two other cases relied upon by MEUA, *Penn-Olin* and *El Paso*,[205] the potential competitors were substantial forces. In *Penn-Olin*, the court

[202]See fn. 199, *supra*.
[203]*Ibid*.
[204]*Ibid*.
[205]*Ibid*.

found both merging companies capable of entering the market independently and noted that even if only one company entered the market, the other could have exerted a procompetitive influence by virtue of its position on the edge of the market. 378 U.S. at 173-176. In *El Paso,* the acquired company (Pacific Northwest) was found to have the capability to enter the California market and to have been "a substantial factor in the California market" through its attempts to enter the market. 376 U.S. at 658-661.

All these cases have a common thread: in each case the test for determining whether a company would be considered by the Court to be a potential competitor in a relevant market involved whether it had a present capability of entering that market or was reasonably viewed by others in the market as having the capability of entering it at any time it desired.

In the case at bar, MEUA seeks to establish its capability of entering the market through rather curious, indeed circular, reasoning. MEUA in the past has forsaken generation because of the costs involved.[206] In this regard, the Board below found that the municipality of Dothan had not seriously considered installing generation (5 NRC at 930-31); we agree with this finding. No solid evidence was shown to indicate that MEUA is considering building its own generation in the near future; the best that could be said for MEUA's members is that they might possibly be interested in installing peak-sharing units.[207] Nor did MEUA identify any other potential bulk power supplier it has considered dealing with in order to reduce its dependence on applicant's generation. MEUA's potential entrance in the market seems instead to hinge on access to Farley. If it is allowed to purchase a share of the plant, we are told, MEUA will be able to compete at wholesale with applicant.[208] In fact, MEUA's counsel admitted at the Phase II hearing that access to Farley is "a *sine qua non* of it being likely or feasible for [MEUA] going into the wholesale market." Tr. 27,022.

Like the Licensing Board, we are left unmoved by this reasoning. The ultimate issue in this case is whether this agency should mandate that applicant accord intervenors access to the Farley facility. In terms of potential competition, we believe MEUA's capability to enter the market must be assessed without regard to the Farley facility.[209] And the record indicates that, without access to Farley, MEUA does not have the

[206]Tr. 3635; 27,029-30.
[207]See Tr. 3878-3888, 3907-3909. At the time of the hearing below, it appeared that MEUA had made no real studies addressing the installation of peak-sharing generation. Tr. 3907.
[208]MEUA Brief, 30-31.
[209]In this regard, it is useful to explore what MEUA's role in the market would have been if the Farley facility were never built. MEUA's counsel was questioned about this at the Phase II hearing; while his response was necessarily speculative, it is certainly clear that MEUA's entrance into the market would have been far more difficult than that of the potential competitors in the cases it cites. See Tr. 27,030-27,033.

capability to enter the wholesale market. We simply can not accept MEUA's argument that if it is granted access to Farley, it could compete in the wholesale market — and that therefore it is a potential competitor in the market and is entitled to such access.

Nor can MEUA claim recognition as a potential competitor in the market for wholesale power on the basis of the second test — that it is currently influencing competitive conditions in the market. MEUA contends that applicant was aware of the municipal systems' desires to install generation and reacted to this desire by pursuing a course of anticompetitive conduct.[210] According to MEUA's argument, the applicant inserted anticompetitive conditions into its wholesale contracts in order to prevent AEC and MEUA from installing generating units. But the Licensing Board found no evidence to support this charge. (5 NRC at 932). Applicant may have been aware of MEUA's desire to enter the market and that MEUA would encounter difficulties in installing generation,[211] but it does not necessarily follow that applicant's conduct was dictated thereby. If a company does not possess the capability to enter a market, it must be assumed, absent evidence to the contrary, that its activities or even its presence do not affect competitive conditions in the market.[212] Given these circumstances, we cannot conclude that MEUA exerted an appreciable influence on the wholesale market.[213]

2. **Retail Market.** Because the Licensing Board found the retail market not to be a relevant one, it did not address the competitive situation at retail between MEUA and applicant. Before the question of remedy for MEUA can be addressed, we must first examine this retail situation and how it has been affected (if at all) by applicant's past conduct.

a. MEUA is composed of the municipal systems of the following 12 cities: Alexander City, Dothan, Fairhope, Foley, LaFayette, Lanett, Luverne, Opelika, Piedmont, Sylacauga, Troy, and Tuskegee. All twelve purchase the bulk of their power supply from applicant; eleven receive additional power from SEPA.[214] 5 NRC at 827-828.

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[210]MEUA Brief, 32.

[211]MEUA Brief, 29.

[212]*Marine Bancorporation, supra,* 418 U.S. at 639-640.

[213]We note here that our finding that MEUA is not likely to install its own generating capacity in the future, coupled with the fact that its members have produced no power in the recent past, lead us to the conclusion that MEUA should not be considered a participant in the market for coordination services in central and southern Alabama. Nothing we have seen in the record below changes our view that non-generating utilities have no appreciable role to play in that market. See fn. 82, *supra.*

[214]The City of Troy purchases no SEPA power; it acquires all its power from applicant. 5 NRC at 828.

Mr. H. Sewell St. John, Sr., the Secretary-Treasurer of MEUA, testified below at great length on the nature of retail competition in central and southern Alabama. His testimony indicated that there is some head-to-head competition, usually for large new loads, between applicant and at least five members of MEUA.[215] This competition has been limited in part by the existence of territorial agreements between applicant and all five of the municipal systems, but it nonetheless must be reckoned with.

While Mr. St. John was able to show that competition between applicant and MEUA exists, neither he nor any other witness was able to identify any harm that a municipal system had suffered because of applicant's assertedly anticompetitive conduct. This is not to say that applicant has never acted in a manner inconsistent with the antitrust laws in its dealings with the municipals; the Board below found (and we agree) that applicant's wholesale contracts with the municipal distributors on their face would discourage the latter from installing their own generation and transmission and from dealing with alternative bulk power suppliers. But we are simply unconvinced that these contractual provisions had any effect on the municipal's retail business.

In the first place, no evidence was presented to indicate that the municipals were either seriously interested in or capable of building their own generating plants or seeking out other bulk power supplies.[216] It can by no means be taken as a given that, at a time when applicant's wholesale rates were concededly low and economies of scale were allowing the construction of larger and more efficient units, isolated municipalities would have chosen to enter the generating field.[217] Nor can we assume, without supporting evidence, that the municipalities would have looked elsewhere for power. As Mr. St. John pointed out (St. John, Direct, 17), even with access to applicant's transmission lines the number of wholesale suppliers the municipals could have feasibly dealt with was limited. We are

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[215]See Tr. 2894 *et. seq.* (Opelika); Tr. 2928 *et. seq.* (Alexander City); Tr. 2996 *et seq.* (Sylacauga); Tr. 3059 *et seq.* (Piedmont); and Tr. 3512 *et. seq.* (Dothan).

[216]We find instructive the examples referred to by MEUA in its brief as illustrative of applicant's success in discouraging the municipals and AEC from developing alternate sources of bulk power. With the exception of applicant's alleged refusal to coordinate with Dothan (see fn. 217, *infra*), all relate to situations involving applicant's dealings with the cooperatives instead of with the municipalities. See MEUA Brief, 56-77.

[217]As far as the record shows, only one municipal, Dothan, considered installing its own generation. The Licensing Board found that there was little evidence presented on this issue (5 NRC at 930-31); we agree. As best as we can tell, Dothan commissioned a study to investigate alternative methods of acquiring bulk power and the study recommended that Dothan continue to purchase power from applicant. See App. X 9. The consultants who performed the study did not explain their decision in their report and they were not called to testify. As for the other municipals, Mr. St. John stated that they never reached the stage of spending money on engineering studies because they did not believe they could generate power as cheaply as they could purchase it. Tr. 3,635.

never told who these potential suppliers were, what their wholesale rates were, or whether they actually had power available. Nor was our attention pointed to an instance where a municipal system investigated the possibility of using applicant's transmission to buy elsewhere.

The second basis for our belief that the municipals were not harmed by any of applicant's anticompetitive practices stems from the municipals' past success in the retail market. The last municipally-owned system taken over by the applicant was that of the town of Headland more than forty years ago. Tr. 2797. And in those towns where Mr. St. John described retail competition between applicant and municipals, the municipals seem to be holding their own.[218] Mr. St. John admitted that the municipals have been profitable, and that applicant has not prevented them from subsisting as viable business entities.[219] Our own review of Mr. St. John's testimony leaves us unconvinced that applicant has even attempted to suppress the municipals, much less succeeded in doing so.

b. MEUA makes one other argument in connection with the retail market. It contends at great length that, since the early 1970's, its members have been subject to a price squeeze rendering them incapable of competing for new industrial loads.[220] It asserts that since that time the applicant has charged MEUA excessively high rates for the wholesale power it purchases.

The Licensing Board rejected the price squeeze argument. It noted that a price squeeze was not apparent on the evidence presented by MEUA. The Licensing Board also saw "no evidence that MEUA members are anything but financially viable." 5 NRC at 939. In addition, it found other evidence in the record which weakened, if not vitiated, the validity of the charge. Moreover, it found that even if a squeeze had existed as charged, it was not of sufficient significance for purposes of Section 105c of the Act.

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[218]See, *e.g.*, Tr. 2906-07 (Opelika successfully competed for a shopping center); Tr. 3043-44 (although applicant has a franchise to compete for loads of more than 100 kva in Sylacauga, the municipal system serves all such loads); Tr. 3512 (Dothan served industrial customer outside its contractually-assigned areas).

[219]Tr. 4079-4081. Our point here is not that the applicant lacked the economic power to drive the municipals under, but that the record before us does not show that it attempted to do so. Compare *Midland, supra*, 6 NRC at 1018-19.

[220]See MEUA Brief, 89-108; MEUA Proposed Findings Below, 52-89. As defined by the Licensing Board (5 NRC at 937):

> A price squeeze involves the economic behavior of a vertically integrated firm *vis a vis* [a] rival who is not similarly integrated. If a manufacturer both marketed its product through its own distribution channel and sold to independent distributors as well, the manufacturer would be engaging in a single price squeeze if it unduly raised the wholesale price to the independent distributors who competed with the manufacturer at retail. A double price squeeze occurs if, in addition to the tactic just mentioned, the vertically integrated manufacturer unduly lowered the retail price of the product in its own outlets as well.

We agree with the Licensing Board's handling of the price squeeze issue. In the first place, we cannot accept the definition urged upon us by MEUA that a price squeeze occurs whenever "a retailer cannot purchase at a wholesale rate sufficiently low to enable it to compete . . . [at retail with its wholesale supplier] and produce a positive margin sufficiently high to cover the costs."[221] This definition purportedly reflects the reasoning applied in the landmark *Alcoa* case.[222] But we do not believe that case established such a protectionist standard.[223] The correct focus of a price squeeze, as the Board below found, is on the pricing policies of the integrated firm. In this case, the crucial factor is whether applicant's wholesale and retail prices adequately reflect production costs.[224] The Board below found no evidence that applicant's retail rates have been kept unjustifiably low,[225] and nothing alluded to in MEUA's brief convinces us that applicant's wholesale rates are set unfairly high.

Beyond the question of whether a price squeeze has in fact occurred, we think it important to reiterate the Licensing Board's view of the consideration that can be given to evidence of a price squeeze in an NRC antitrust proceeding. We are not empowered to establish wholesale rates; that function resides in the FERC. We are interested in evidence of a price squeeze only insofar as it sheds light on the "intent and purpose of Applicant in its competitive relationship with other parties."[226] For the reasons set forth by the Licensing Board, we do not believe MEUA has met its burden in advancing this contention; the evidence does not establish that applicant has set its retail and wholesale rates at levels designed to prevent MEUA from competing for industrial customers.

With our assessment of the factual record made below now complete, we turn to the question of remedy.

221MEUA Brief, 92.
222*United States* v. *Aluminum Company of America*, 148 F.2d 416, 436-438 (2d Cir. 1945).
223While Judge Learned Hand never explicitly delineated the elements of a price squeeze in *Alcoa*, he did find that the defendant's price for the raw material was higher than a "fair price." *Id.* at 437.
2245 NRC at 937 n. 265.
225*Id.* at 939.
2265 NRC at 940.

VI.

REMEDY

In the proceeding below, the Licensing Board — finding five instances of anticompetitive action by the applicant and invoking several "public interest" considerations — ordered the imposition of a number of conditions on the licenses which may be issued to the applicant for the two units of the Farley Nuclear Plant. The principal conditions required the applicant (1) to provide AEC with access to the Farley plant in the form of unit power; (2) to provide transmission services to enable AEC to make effective use of that power; and (3) to provide AEC with backup bulk power to cover those situations when Farley is down for maintenance or other causes. 5 NRC at 1501-09. The Board below considered the conditions warranted upon a "weighing and evaluating [of] the various antitrust and other public interest concerns." *Id.* at 1501-02.

These conditions extended benefits only to AEC. The Licensing Board ruled that MEUA was not entitled to relief because "there was no significant actual or prospective competition between Applicant and [MEUA] at the retail distribution level, nor other conduct of Applicant toward MEUA or its members which was inconsistent with the antitrust laws within the meaning of Section 105c of the Atomic Energy Act." 5 NRC at 1484. A grant to MEUA of access to Farley under those conditions, according to the Board, "might be considered an unwarranted attempt to restructure the electric power industry at the retail level, rather than fulfilling the statutory mandate of antitrust review under Section 105c." *Ibid.*

All the parties object. The applicant's basic position is that no remedy in the form of license conditions is warranted by the Licensing Board's findings. If license conditions are nonetheless found necessary, we are told, the sale of wholesale power rather than unit power would be more appropriate.[227] On the other hand, the remaining parties argue that the remedy does not go far enough. For various asserted reasons, each of these parties claims that the Licensing Board erred in not ordering more extensive relief — generally ownership access to the Farley plant and greater access to APCO's transmission facilities. Their thesis is that, on the facts of this case, a stronger remedy that that imposed by the Licensing Board is mandated by the Atomic Energy Act and applicable principles of antitrust law.

[227]APCO Brief, 82-89.

A. Remedial Standards Under Section 105c

In view of our findings that the applicant engaged in anticompetitive conduct beyond that which the Board below attributed to it, we need not decide whether the license conditions imposed by the Licensing Board constituted a remedy appropriate to the limited "liability" findings it made. Our finding that the applicant's refusal to grant AEC ownership access to Farley constituted anticompetitive action, along with the other determinations made by us in Parts III, IV, and V, *supra*, have significantly changed the dimensions of the "situation inconsistent" which must be considered in determining the remedy. The decision that is called for on our part, therefore, is not so much a determination of whether the relief ordered by the Licensing Board should be upheld, but rather what remedy we believe to be appropriate in light of the "situation inconsistent" as *we* find it.

This brings us to the question of the standard to be applied in determining the license conditions for the plant. The applicant argues that "an antitrust tribunal, given a choice of remedies addressed to anticompetitive conduct, should choose the least onerous adequate remedy available."[228] It goes on to say that "[i]n the context of Section 105c(6) of the Act, 'the adequacy' of a particular remedy depends upon two principal factors: (1) on a case-by-case basis, whether the remedy neutralizes the impact of the licensed facility upon the competitive situation in a particular market in light of the affirmative findings under Section 105c(5) and detailed evidence of the existing competitive situation in that market; and (2) whether the remedy selected has a nexus to the Applicant's activities under the license."[229] Implicitly, the applicant is telling us that the Commission's remedial antitrust authority is a narrow one, extending only to the neutralization of whatever competitive advantage the licensed facility may add to the preexisting competitive situation and limited to the activities under the license.

The other parties have a far more expansive view of the Commission's remedial authority. They suggest in varying ways that the Commission has the authority to impose any license conditions it deems necessary to cure or eliminate the situation found inconsistent.[230]

We find the applicant's view of the Commission's antitrust remedial authority unduly restrictive. It cannot be sustained by the language of Section 105c of the Act; nor is it supported by the legislative history of that provision.

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[228] *Id.* at 84.

[229] *Id.* at 85 (footnotes omitted).

[230] Staff Brief, 32-35; Justice Brief, 9-16; AEC Brief, 41; MEUA Brief, 126.

In both *Midland*[231] and *Davis-Besse*,[232] we had occasion to consider the scope of the Commission's remedial authority under Section 105c.[233] In the latter case, we were confronted with the argument, like the nexus argument of the applicant here, that the Commission may only grant relief that would govern activities under the license. We disposed of that argument with the following answer:

> To begin with, the limiting phrase "activities under the license" is not in Section 105c(6) which governs the scope of relief. To the contrary, paragraph (6) is cast in the broadest terms. In pertinent part it provides where the Commission finds a situation inconsistent with the antitrust laws that it "shall have the authority to issue or continue a license as applied for, to refuse to issue a license, to rescind a license or amend it, and to issue a license with such conditions as it deems appropriate." The provision conveys the message that Congress did not want nuclear plants authorized in circumstances that would create or maintain anticompetitive situations without license conditions designed to redress them. This construction is fully warranted on the face of paragraph (6). This is also the meaning specifically ascribed to it by its congressional authors, the Joint Committee on Atomic Energy:
>
> > "The Committee believes that, except in an extraordinary situation, Commission-imposed conditions should be able to eliminate the concerns entailed in any affirmative finding under paragraph (5) [of Section 105c] . . ."

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[231] 6 NRC at 1094-1100.

[232] 10 NRC at 282-94.

[233] The pertinent paragraphs of Section 105c are (5) and (6). They read:

> (5) Promptly upon receipt of the Attorney General's advice, the Commission shall publish the advice in the Federal Register. Where the Attorney General advises that there may be adverse antitrust aspects and recommends that there be a hearing, the Attorney General or his designee may participate as a party in the proceedings thereafter held by the Commission on such licensing matter in connection with the subject matter of his advice. The Commission shall give due consideration to the advice received from the Attorney General and to such evidence as may be provided during the proceedings in connection with such subject matter, and shall make a finding as to whether the activities under the license would create or maintain a situation inconsistent with the antitrust laws as specified in subsection 105a.
>
> (6) In the event the Commission's finding under paragraph (5) is in the affirmative, the Commission shall also consider, in determining whether the license should be issued or continued, such other factors, including the need for power in the affected area, as the Commission in its judgment deems necessary to protect the public interest. On the basis of its findings, the Commission shall have the authority to issue or continue a license as applied for, to refuse to issue a license, to rescind a license or amend it, and to issue a license with such conditions as it deems appropriate.

10 NRC at 291 (references omitted).

Then, we went on to explain:

> When construing this provision [Section 105c(6)] in *Midland*, we stressed that "no type of license condition — be it a requirement for wheeling, coordination, unit power access, or sale of an interest in the plant itself — is necessarily foreclosed as a possible form of relief. Section 105c imposes no limits in this respect; it gives the Commission 'authority . . . to issue a license with such conditions as it deems appropriate.' " In other words, as we explained when faced with similar arguments in *Wolf Creek*, "[S]ection 105c(6) simply directs the Commission to place 'appropriate' conditions on licenses where necessary to rectify anticompetitive situations. This is an invocation of the Commission's discretion, not a limitation on its powers. Had Congress wished to do the latter, it would have said so in unmistakable terms."
>
> The idea that the remedies in the antitrust arsenal are sufficient to overcome the violations is neither original nor recent. Rather, this settled tenet is one of the "principles developed by the Antitrust Division, the Federal Trade Commission, and the Federal Courts" which we apply in proceedings under section 105c. The Supreme Court has reiterated that "relief in an antitrust case must be 'effective to redress violations' and 'to restore competition.' " And "adequate relief in a monopolization case should . . . render impotent the monopoly power found to be in violation of the [Sherman] Act."

Id. at 292 (references omitted).

In sum, the Commission's remedial authority under 105c(6), while not boundless, is more extensive than the applicant believes. The Commission has wide discretion in fashioning "appropriate" license conditions "where necessary to rectify anticompetitive situations." "[N]o type of license condition — be it a requirement for wheeling, coordination, unit power access, or sale of an interest in the plant itself — is necessarily foreclosed as a possible form of relief."[234] And the license condition need not be confined in its application to activities under the license. This is not to suggest, however, that the Commission's authority to impose "appropriate" license conditions is *carte blanche.* The authority to act may not be divorced from the purposes of the legislation. It does not include the authority to employ

[234] *Midland, supra,* 6 NRC at 1099.

license conditions "as an implement to restructure the electric utility industry."[235]

The question then remains: What are the considerations which the Commission may factor into its decision of "appropriate" license conditions? In its decision, the Board below considered not only antitrust factors but other "public interest" factors as well in arriving at the appropriate license conditions for the Farley facility. These public interest factors included (1) the "need for power" (*i.e.*, the need for the generating capacity represented by the Farley plant to meet the anticipated power demands of the applicant's service area); (2) AEC's tax and other advantages stemming from its status as an electric cooperative; (3) the "grandfathered" nature of the antitrust review associated with the fact that construction permits for Farley were applied for prior to the enactment of Section 105c in 1970; and (4) the Board's finding that all anticompetitive conduct by the applicant had ceased by early 1972. According to the Board, Section 105c(6) of the Act mandated that it consider these public interest factors in addition to the relevant antitrust factors:

> It is indisputable that these antitrust laws embody a fundamental national policy regarding the preservation of competition in our economic system. But a finding of inconsistency with the antitrust laws under Section 105c(5) does not end the inquiry, but leads to a consideration of other public interest factors in accordance with Section 105c(6). The latter section requires the Commission then to consider "such other factors, including the need for power in the affected area, as the Commission in its judgment deems necessary to protect the public interest" (42 U.S.C. Section 1235(c) (6)).

5 NRC at 1496 (footnote omitted).

The propriety of the Licensing Board's use of these public interest considerations as mitigating factors in fashioning appropriate license conditions is disputed by several of the parties.[236] AEC argues that under Section 105c(6), public interest factors may be taken into consideration only to determine whether to issue or continue a license. Where as here no one is urging the refusal, rescission or revocation of a license, AEC claims that those public interest factors cannot be invoked to allow less stringent license conditions.[237] AEC sees this result as required by the portion of the first sentence of Section 105c(6) (underscored in its brief) directing the Commission to consider certain factors necessary to protect the public

[235]*Id.* at 1100.
[236]Staff Brief, 43-47; Justice Brief, 28-34, 41-52; AEC Brief, 28-31, 35-38.
[237]AEC Brief, 29-31.

interest "in determining whether the license should be issued or continued."[238] The NRC staff and Justice follow a different tack. Rather than taking issue with the propriety of considering public interest factors in fashioning appropriate license conditions, they disagree with the Licensing Board's use of those factors in this case. Specifically, they do not believe public interest considerations here lie in favor of mitigating license conditions which otherwise might be appropriate.[239]

Because we are undertaking to determine the appropriate license conditions ourselves based on a set of findings different from that on which the Licensing Board premised its conditions, it is bootless to spend effort on each detailed aspect of the Licensing Board's assessment of the public interest considerations factored into its decision.[240] In a more general vein, however, we disagree with AEC's reading of Section 105c(6) that public interest considerations are relevant only for determining whether a license should issue or have its life extended.

In resting on the quoted portion of the first sentence of Section 105c(6) for its interpretation of the statute, AEC gives the Section too crabbed a reading. Its error lies in its failure to give full effect to the remaining sentence of the Section: "On the basis of its findings, the Commission shall have the authority to issue or continue a license as applied for, to refuse to issue a license, to rescind a license or amend it, and to issue a license with such conditions as it deems appropriate." With the single qualification that the Commission decision be based on its findings, the operative words of the sentence are without restriction. This being so, we decline to read Section 105c(6) as precluding the Commission from considering the "need for power" and other public interest factors in its determination of license conditions and from imposing less onerous conditions if it decides that both the situation inconsistent found under (5) and the public interest findings under (6) make those conditions appropri-

[238]The first sentence of Section 105c(6), with the portion emphasized by AEC in italics, reads as follows:

> "In the event the Commission's finding under paragraph (5) is in the affirmative, the Commission shall also consider, *in determining whether the license should be issued or continued*, such other factors, including the need for power in the affected area, as the Commission in its judgment deems necessary to protect the public interest."

Id. at 29.

[239]Staff Brief, 43-47; Justice Brief, 41-52.

[240]The Board's assessment can be found at 5 NRC at 1496-1501.

ate.[241] This, of course, does not mean that antitrust concerns should be ignored or overriden by other public interest considerations. For as the Joint Committee's report expressly states, except in an extraordinary situation, the Commission's action under paragraphs (5) and (6) should harmonize both antitrust and public interest considerations.[242]

B. Appropriate Remedial Conditions

1. **Objective.** Our task, then, is to decide on the license conditions which serve here to "harmonize both antitrust and such other public interest considerations as may be involved." But before we embark on that journey, we turn again to the Atomic Energy Act for an analysis of the purposes and objectives to be served by our decision.

One of the basic foundations on which the Atomic Energy Act rests is the principle of free competition in private enterprise. This principle is manifested at the very outset of the Act by the policy declaration that the "development, use, and control of atomic energy shall be directed so as to ... strengthen free competition in private enterprise."[243] This policy finds manifestation again in Section 105 of the Act. In that Section, the Congress made it clear that the national antitrust laws were to continue in full force and effect with respect to atomic energy matters. It did so by explicitly providing that "[n]othing contained in the Act shall relieve any person from the operation" of the antitrust laws (subsection 105a); and by following with a provision (subsection 105c) which calls for an antitrust review of every nuclear power plant prior to its construction. Thus, through the mechanism of the antitrust laws, the Congress sought to protect free competition in private enterprise in the development and use of atomic energy. Nor did Congress stop with the protection afforded by the antitrust laws. It significantly widened the area of potential Commission action by directing that the policies underlying the antitrust laws must be given effect

[241]That findings under both (5) and (6) are to be taken into account in fashioning license conditions is made clear in the Report of the Joint Committee on the bill which enacted Section 105c into law:

> The Committee believes that, except in an extraordinary situation, Commission-imposed conditions should be able to eliminate the concerns entailed in any affirmative finding under paragraph (5) while, at the same time, accommodating the other public interest concerns found pursuant to paragraph (6). Normally, the committee expects the Commission's actions under paragraphs (5) and (6) will harmonize both antitrust and such other public interest considerations as may be involved.

Report by the Joint Committee on Atomic Energy to accompany H.R. 18679, H.R. Rep. No. 91-1470, 91st Cong., 2d Sess., p. 31 (1970).

[242]*Ibid.; accord, Midland, supra,* 6 NRC at 1098 fn. 733.

[243]Atomic Energy Act, Section 1; 42 U.S.C § 2011.

as well. As a further measure of protection, the legislation was not limited to situations involving actual violations of the antitrust laws or the then-underlying policies. Situations involving the *reasonable probability* of contravention of those laws and the policies clearly underlying them were also made subject to remedial action by the Commission.[244]

The remedial action the Congressional authors had in mind was that "except in an extraordinary situation, Commission-imposed conditions should be able to eliminate the concerns entailed in any affirmative finding under paragraph (5)."[245] And as we emphasized earlier (p. 1114, *supra*), this concept is consistent with settled tenets of antitrust practice as manifested by the actions of the courts and the federal agencies which deal with those laws: relief in an antitrust case must be effective to redress violations and to restore competition.[246]

2. **Ownership Access to Farley.** In the earlier portions of our decision, we determined that the applicant enjoyed a dominant position in all three product markets. We also determined that the applicant had acted inconsistently with the antitrust laws and the policies thereunder in seven different instances, including its refusal to share ownership of the Farley plant with AEC. We found that this refusal to share in the ownership of Farley was in furtherance of the applicant's long held objective of preserving the dominant power which it enjoyed in all aspects of the electric power business in central and southern Alabama. Upon full consideration of the situation and the requirements and objectives of the Act, the conclusion we must reach is clear: To eliminate the concerns and to strengthen free competition in private enterprise, the license to the applicant for the construction and operation of the Farley plant must, as a minimum, include conditions providing (1) AEC with an opportunity to obtain a proportionate share in the ownership of the plant and (2) reasonable transmission or wheeling services as may be needed by AEC and MEUA.

In lieu of an ownership share in Farley, we considered a license condition — such as that imposed by the Board below — requiring the

[244]Report of the Joint Committee on Atomic Energy on S4141, S. Rep. No. 91-1247, 91st Cong., 2d Sess., p. 14 (1970), discussed in *Midland, supra*, 6 NRC at 926-27.

[245]S. Rep. No. 91-1247 (see fn. 244) at p. 31. In placing the responsibility on the Commission to fashion the appropriate remedy where the antitrust situation was found wanting, these same Congressional authors recognized that "there is not a clear boundary between antitrust considerations in relation to the strengthening of free competition in free enterprise and measures to accomplish such objective for reasons other than the antitrust laws or underlying antitrust policy." Rather than trying to legislate the boundaries of the antitrust considerations, the Joint Committee left it to the Commission to decide. In the Joint Committee's words: "the Commission will have to exercise discretion and judgment." *Id.* at p. 15.

[246]*Davis-Besse, supra*, 10 NRC at 292.

applicant to offer to AEC a share in Farley in the form of unit power. We reject that alternative. We find it would neither strengthen free competition in the applicant's market area nor eliminate the antitrust concerns which we found to exist in that market.

In a unit power arrangement, the purchaser is charged for all of the owner's costs of providing that power, including the costs of capital, of construction, and of fuel and operation. Where the owner is a private utility such as the applicant here, the charge to the purchaser includes a rate of return on the owner's investment.[247] This means that were AEC to purchase power from the applicant on a unit power basis, it would lose the benefits of the advantageous financing otherwise available to it for the capital costs attributable to its share of the plant. Due to its cheaper capital costs, primarily through the availability of low-cost loans, AEC could save approximately 7 mills per KWH through ownership access to Farley as opposed to unit power access.[248] It also has certain tax advantages over investor-owned utilities.

The availability of low cost loans to rural electric cooperatives such as AEC is not without good reason. Historically, these cooperatives were established to serve rural areas where the population is widely-dispersed and the customers have relatively low power demands. Consequently, they were faced with higher costs in bringing power to their customers in comparison to their investor-owned or municipal counterparts whose service areas were generally comprised of more densely populated areas.[249] Recognizing this factor, Congress enacted legislation to provide capital at low interest rates to enable electric cooperatives to provide service to its customers at rates comparable to those enjoyed by the others.[250]

In the circumstances of this case, we cannot perceive how a unit power arrangement would promote free competition, let alone "eliminate the concerns." Rather, a unit power arrangement would deprive AEC of its financing advantages — the very advantages Congress thought necessary for cooperatives such as AEC to operate effectively.

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[247]See fn. 7, *supra*.

[248]By AEC's estimate, its cost of a Kw of power, if it owned 4% of the Farley plant, would amount to 18.9 mills under a joint ownership arrangement, while by the same estimate, applicant's cost of producing power at Farley — the unit power cost to AEC — was placed at 26.2 mills. Rogers, Tr. 27,459-62.

[249]As a result, rural rates for retail use of power historically have been higher than urban rates. St. John, Tr. 4654.

[250]Rural Electrification Act of 1936, 7 U.S.C. §§ 901 *et seq*. See also House Report No. 93-91, the House Committee report on the House version of the bill which became P.L. 93-32 establishing a Rural Electrification and Telephone Revolving Fund. U.S. Code Cong. and Admin. News, p. 1365 (1973).

In this regard, the Licensing Board concluded that a "consideration of AEC's tax and other advantages is irrelevant for all purposes under the facts of the instant case." The Board thereupon purported to adopt the Department's suggestion that "one takes his competition as he finds them."[251] Notwithstanding this pronouncement, the action of the Board in ordering unit power did not leave AEC in its normal competitive position; its real effect was to deprive AEC of its normal financing advantages in connection with the power it would obtain from the Farley plant. These tax and other financing advantages were accorded the cooperatives by the Congress as a matter of governmental policy.[252] Absent a showing that these advantages serve to operate in derogation of the antitrust laws and the policies underlying them, we know of no sound reason why we should act to keep AEC from enjoying them.[253]

Generally, the antitrust laws seek to prevent the unreasonable use of market power to gain additional market power.[254] In this case, it can be expected that the addition of Farley to the applicant's generating capacity will over the years increase applicant's existing market dominance. Thus, a key consideration here is the action we must take to forestall that expectation from becoming reality. We find that, of the types of arrangements for access to generating capacity generally found in the electric industry, ownership access is likely to be the most effective way of accomplishing this result, because this arrangement will enable AEC to compete more effectively. As a part-owner, AEC will be able to take advantage of the lower interest and tax benefits available to it for financing its share of the plant which will, in turn, translate to lower costs for its share of the output from Farley. In the words of one witness, "there is a very substantial and meaningful difference between Alabama Power Company's costs and AEC's costs on an ownership basis, no matter whose figures you use."[255] And this observation should hold relatively true even if all parties' costs increase with time.[256]

We thus render explicit that which implicitly follows from the considerations we have just outlined: No less than a proportionate sharing of the ownership of the Farley plant by the applicant and AEC will suffice to accommodate the objectives of strengthening free competition in private

[251]5 NRC at 1497.
[252]*Midland, supra,* 6 NRC at 1019.
[253]We note in passing that the applicant enjoys special privileges accorded by other governmental entities, and is protected against competition from REA cooperatives in much if not most of its service territory.
[254]See, *e.g., United States* v. *Griffith,* 334 U.S. 100, 107-08 (1948).
[255]Rogers, Tr. 27,461.
[256]*Ibid.*

enterprise and eliminating the concerns which arise from our adverse antitrust findings related to the applicant's past conduct.[257]

3. **Public Interest Considerations.** In exercising our judgment in the foregoing respect, we have not overlooked the public interest factors with which the Licensing Board found the antitrust values must be harmonized. We agree with that Board's finding of the need for power[258] and the concomitant decision not to withhold the issuance of a license to the applicant for the construction and operation of the plant. But as regards the other public interest factors considered by the Licensing Board, we do not find cause to follow its lead.

One of these public interest considerations related, in the words of the Licensing Board, to the "grandfathered" nature of the antitrust review.[259] The Licensing Board noted that the applicant had filed its original application for a construction permit on October 10, 1969, and an amendment for authority to construct a second unit on June 26, 1970, both prior to the December 1970 amendments to Section 105c. Notice of the antitrust hearing was not issued by the Commission until June 28, 1972. The Licensing Board found equities flowing to the applicant from this sequence of events.

We fail to find in the "grandfathered" situation any justification for striving to achieve in any less than full measure the antitrust goals embedded in the Atomic Energy Act. Even though the license applications were filed prior to the enactment in 1970 of the current antitrust review provisions found in Section 105c, applicant must be presumed to have known that the antitrust laws would apply to their fullest to any license issued by the Commission. Section 105a of the Act, which was unaffected by the 1970 amendments, made this clear. Indeed, concern with the competitive aspects of licensing in the nuclear area went back to the original atomic energy legislation enacted in 1946.[260] In these circum-

[257]Of course, these same reasons cause us to reject out-of-hand applicant's argument that the remedy need only be the sale of wholesale power.

[258]5 NRC at 1500.

[259]5 NRC at 1498.

[260]Section 7(c) of the Atomic Energy Act of 1946 formerly provided that:

> Where activities under any license might serve to maintain or to foster the growth of monopoly, restraint of trade, unlawful competition, or other trade position inimical to the entry of new, freely competitive enterprises in the field, the Commission is authorized and directed to refuse to issue such license or to establish such conditions to prevent these results as the Commission, in consultation with the Attorney General, may determine. The Commission shall report promptly to the Attorney General any information it may have with respect to any utilization of fissionable material or atomic energy which appears to have these results.

stances, we discount this "grandfather" situation as a mitigating factor in our decision.

We also depart from the Licensing Board's consideration of the "alleged cessation of anticompetitive conduct as a mitigating factor."[261] According to the Board, "[t]here is no evidence that established conduct inconsistent with the antitrust laws beyond early 1972."[262] This observation is not altogether true. In at least one instance, the applicant's anticompetitive behavior extended until 1976, when it finally agreed to remove Section 4.2 from its contract with SEPA.[263] That provision, which in essence required SEPA's preference customers to purchase all of their supplemental power needs from the applicant, had been held by the Board to be anticompetitive.[264]

But an even more fundamental reason exists for our position. The fact that a transgressor has ceased its anticompetitive activity, especially when such cessation occurs after the onset of legal action,[265] in and of itself provides no justification for dispensing with otherwise appropriate remedial requirements. As the Supreme Court admonished in *United States* v. *Oregon State Medical Society:*

> It is the duty of the courts to beware of efforts to defeat injunctive relief by protestations of repentance and reform, especially when abandonment seems timed to anticipate suit, and there is probability of resumption.

343 U.S. 326, 333 (1952).

4. **Basis for Allocation.** Our decision calling for a proportionate ownership of Farley by AEC brings up the matter of how its share should be determined. The Licensing Board had devised an allocation formula, albeit in terms of unit power shares, "based on a ratio of (a) the aggregate coincident demand of all wholesale-for-resale members of AEC in Alabama *during the hour of peak demand on the electric system of [the applicant]* in 1976 to (b) the sum of such coincident demands of AEC and the territorial peak-hour demands of [the applicant] (excluding therefrom the peak-hour demands imposed by members of AEC upon the electric system of [the applicant], *during the hour of peak demand on [the applicant's] electric system in 1976."* (Emphasis added.)[266]

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[261] 5 NRC at 1500-1501.
[262] *Id.* at 1501.
[263] Stipulation by parties, Tr. 28,317-19.
[264] 5 NRC at 933-37.
[265] As noted above, the notice of hearing was issued in mid-1972; the trial commenced in December 1974.
[266] 5 NRC at 1507.

AEC accepts that "participation should be on the basis of the proportion of AEC's on- and off-system wholesale loads in central and southern Alabama to the total loads of both parties in such area."[267] However, it points out that the peak demands for each of AEC's on-system and off-system members and for applicant do not occur simultaneously.[268] The result of the Licensing Board's allocation formula, says AEC, enables the applicant to retain a disproportionate share of the facility.[269] AEC suggests instead that the ratio should be pegged to the load of AEC's on-system and off-system members and of the applicant *at the time of their respective peak loads.*

We agree with this position of AEC. Basing the allocation formula on the time of applicant's peak demand skews the result in its favor. A more equitable division of ownership would result if the shares were to be determined by the respective peak demands of AEC and the applicant occurring during 1976. The license condition we impose is based accordingly.

5. **Access to Transmission Services.** This brings us to the second of the license conditions we have determined are necessary in the circumstances of this case. It is evident that AEC needs access to the applicant's transmission system to make effective use of its share of the output from Farley.[270] It needs these services to transmit the power both to AEC's on-system and off-system members. Because AEC's on-system members are not interconnected directly to the off-system members, AEC also needs transmission services from its on-system members to its off-system members. But the need by AEC for transmission services is not limited to the power from Farley. To enable AEC to plan for and use in the most efficient manner all of the power to which it may have access — whether by self-generation or by purchase — it needs the transmission services of the applicant.[271] Without access to these transmission services, AEC's system would be an island to itself, isolated from other power sources or systems. Indeed, because it is not interconnected with all of its members, AEC is even now dependent on the applicant to bring power to AEC's off-system members.

AEC must have access to other sources of wholesale power as well as markets for any excess power it may have. The applicant enjoys such access through its interconnections with the Southern pool and, through that pool, with other nearby utilities. Through this access, the applicant is in a

[267] AEC Brief, 69.
[268] *Ibid.*
[269] *Ibid.*
[270] Rogers, Tr. 27,357.
[271] *Ibid.*

position to coordinate the various factors of production to produce, buy or sell reliable firm wholesale power under optimum conditions. Without equivalent access AEC would be unable to utilize fully its share of the power from Farley, hindering its ability to compete effectively against the applicant. Such a situation is unlikely to lead to a significant attenuation of the applicant's dominant position in central and southern Alabama, let alone strengthen free competition in private enterprise.

6. **MEUA's Remedy.** Our dissatisfaction with some of the Licensing Board's findings relating to MEUA perforce required us to reexamine the decision below to deny MEUA any remedy in this proceeding. As mentioned earlier, that decision was based on a finding that "there is no significant actual or prospective competition between [MEUA and applicant] at the retail distribution level,"[272] a finding we cannot accept. (See pp. 1060-1066, *supra*). Our disagreement with the decision below also presents us with an apparent due process problem: because the Licensing Board determined that MEUA was not entitled to any remedy, it excluded MEUA from offering evidence at the Phase II remedy hearing.[273]

In the circumstances, we could remand the case to the Licensing Board to allow MEUA an opportunity to present evidence on the subject of remedy. We do not, however, believe such a course is either necessary or desirable. In the first place, our views on remedy are shaped largely by our findings concerning the "situation inconsistent." Defining that situation was the purpose of the Phase I hearing, a phase in which MEUA participated actively. Second, MEUA was allowed to and did make an offer of proof at the Phase II hearing. We have carefully reviewed the offer[274] and find nothing therein which would, if developed more fully, cause us to change our opinion on remedy.

As we have said, our choice of remedy is dependent on the situation inconsistent with the antitrust laws. We think it important to place that situation as it affects MEUA in its proper perspective. We have found that MEUA and applicant compete at retail. We have found that applicant, by virtue of its dominant control of generation and transmission facilities in central and southern Alabama, has monopoly power in the retail market. And we have found that applicant has placed anticompetitive restrictions on MEUA's right to pursue other bulk power supply options.

On the other hand, we have found many of MEUA's allegations unsubstantiated by the evidence. In particular, we believe MEUA's role in the wholesale market is that of a buyer, and not in any real sense of a

[272] 5 NRC at 961.
[273] Tr. 27,189; 27,204.
[274] Tr. 27,437 - 27,445.

potential seller. We do not believe anticompetitive contractual restrictions have played a large part in MEUA's failure to develop other bulk power supply alternatives; we think MEUA would have continued as a wholesale customer of applicant regardless of the restrictions. Finally, we see no evidence that MEUA has been harmed in its retail role by any anticompetitive behavior on the part of applicant or that applicant has wrongfully attempted to limit MEUA'S retail business. The evidence shows that applicant has monopolized the wholesale market; it does not show that the applicant has unlawfully monopolized the retail market or sought to do so.

In sum, our analysis of the situation relative to MEUA finds it limited to the restrictions placed on MEUA's ability to look elsewhere than to the applicant for sources of bulk power. MEUA is plainly entitled to a remedy that eliminates these restrictions. This includes both the removal of any offensive contractual provisions still in force between applicant and any member of MEUA and the use of applicant's transmission facilities (where available and with appropriate compensation) to enable MEUA to deal with other suppliers of bulk power.

In terms of access to the Farley nuclear facilities, we do not believe ownership access is warranted in the case of MEUA. MEUA has been able to compete effectively in the retail market in the past; we see no indication that an ownership interest is necessary to pry open the market. Nor is ownership access necessary to remedy the contractual limitations placed on MEUA's right to look for alternative suppliers. The municipals have purchased all their power requirements for decades; assuming power from Farley is fairly included in applicant's wholesale power mix, we fail to see how the nuclear facility will change in any way the situation at retail between applicant and MEUA. MEUA is entitled to enjoy any benefits of lower-cost nuclear power, but should be able to do so (and remain competitive) through the purchase of wholesale power from the applicant.[275]

Nothing in this decision, of course, prevents applicant from selling unit power or a portion of the Farley facilities to MEUA if the two parties so desire. We merely hold today that, in the circumstances of this case, where the two parties have fairly competed at retail for many years and where the Farley facilities will not impede MEUA's ability to continue doing so, the elimination of the situation in the retail market that is inconsistent with the antitrust laws can be accomplished without awarding the municipals the right to purchase a share of the Farley plant.

[275] See excerpts from the legislative history of Section 105c at 5 NRC at 1491-96.

CONCLUSION

The conditions appended to this decision shall be incorporated in the applicant's licenses in lieu of the present antitrust conditions; all exceptions not addressed herein have either been denied or found immaterial to our decision; the Licensing Board's decision is *modified* in accordance with the foregoing opinion and is *affirmed as modified.*

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Bishop
Secretary to the
Appeal Board

APPENDIX

License Conditions Approved by the Appeal Board

The following license conditions are made a part of any licenses issued to the applicant for the Joseph M. Farley Nuclear Plant, Units 1 and 2:

1. Licensee shall recognize and accord to Alabama Electric Cooperative the status of a competing electric utility in central and southern Alabama.

2. Licensee shall offer to sell to AEC an undivided ownership interest in Units 1 and 2 of the Farley Nuclear Plant. The percentage of ownership interest to be so offered shall be an amount based on the relative sizes of the respective peak loads of AEC and the Licensee (excluding from the Licensee's peak load that amount imposed by members of AEC upon the electric system of the Licensee) occurring in 1976. The price to be paid by AEC for its proportionate share of Units 1 and 2, determined in accordance with the foregoing formula, will be established by the parties through good faith negotiations. The price shall be sufficient to fairly reimburse Licensee for the proportionate share of its total costs related to the Units 1 and 2 including, but not limited to, all costs of construction, installation, ownership and licensing, as of a date, to be agreed to by the two parties, which fairly accommodates both their respective interests. The offer by Licensee to sell an undivided ownership interest in Units 1 and 2 may be conditioned, at Licensee's option, on the agreement by AEC to waive any right of partition of the Farley plant and to avoid interference in the day-to-day operation of the plant.

3. Licensee will provide, under contractual arrangements between Licensee and AEC, transmission services via its electric system (a) from AEC's electric system to AEC'S off-system members; and (b) to AEC'S electric system from electric systems other than Licensee's, and from AEC'S electric system to electric systems other than Licensee's. The contractual arrangements covering such transmission services shall embrace rates and charges reflecting conventional accounting and ratemaking concepts followed by the Federal Energy Regulatory Commission (or its successor in function) in testing the reasonableness of rates and charges for transmission services. Such contractual arrangements shall contain provisions protecting Licensee against economic detriment resulting from transmission line or transmission losses associated therewith.

4. Licensee shall furnish such other bulk power supply services as are reasonably available from its system.

5. Licensee shall enter into appropriate contractual arrangements amending the 1972 Interconnection Agreement as last amended to provide for a reserve sharing arrangement between Licensee and AEC under which

the Licensee will provide reserve generating capacity in accordance with practices applicable to its responsibility to the operating companies of the Southern Company System. AEC shall maintain a minimum level expressed as a percentage of coincident peak one-hour kilowatt load equal to the percent reserve level similarly expressed for Licensee as determined by the Southern Company System under its minimum reserve criterion then in effect. Licensee shall provide to AEC such data as needed from time to time to demonstrate the basis for the need for such minimum reserve level.

6. Licensee shall refrain from taking any steps, including but not limited to the adoption of restrictive provisions in rate filings or negotiated contracts for the sale of wholesale power, that serve to prevent any entity or group of entities engaged in the retail sale of firm electric power from fulfilling all or part of their bulk power requirements through self-generation or through purchases from some source other than licensee. Licensee shall further, upon request and subject to reasonable terms and conditions, sell partial requirements power to any such entity. Nothing in this paragraph shall be construed as preventing applicant from taking reasonable steps, in accord with general practice in the industry, to ensure that the reliability of its system is not endangered by any action called for herein.

7. Licensee shall engage in wheeling for and at the request of any municipally-owned distribution system:

(1) of electric energy from delivery points of licensee to said distribution system(s); and
(2) of power generated by or available to a distribution system as a result of its ownership or entitlement* in generating facilities, to delivery points of licensee designated by the distribution system.

Such wheeling services shall be available with respect to any unused capacity on the transmission lines of licensee, the use of which will not jeopardize licensee's system. The contractual arrangements covering such wheeling services shall be determined in accordance with the principles set forth in Condition (3) herein.

The Licensee shall make reasonable provisions for disclosed transmission requirements of any distribution system(s) in planning future transmission. By "disclosed" is meant the giving of reasonable advance notification of future requirements by said distribution system(s) utilizing wheeling services to be made available by Licensee.

*"Entitlement" includes but is not limited to power made available to an entity pursuant to an exchange agreement.

8. The foregoing conditions shall be implemented in a manner consistent with the provisions of the Federal Power Act and the Alabama Public Utility laws and regulations thereunder and all rates, charges, services or practices in connection therewith are to be subject to the approval of regulatory agencies having jurisdiction over them.

Cite as 13 NRC 1115 (1981) LBP-81-16

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Marshall E. Miller, Chairman
Dr. Emmeth A. Luebke
Dr. Oscar H. Paris

In the Matter of **Docket Nos. 50-250-SP**
50-251-SP
(Proposed Amendments to Facility Operating License to Permit Steam Generator Repairs)

FLORIDA POWER AND LIGHT COMPANY
(Turkey Point Nuclear Generating Station, Units 3 and 4) **June 19, 1981**

Upon consideration of detailed information submitted by the parties concerning the handling, storage, transportation or other disposition to be made of low level solid waste that might be generated by the licensee's proposed steam generator repairs, the Licensing Board concludes that the impact of a hurricane or tornado on the low level waste produced and stored temporarily on site in connection with those repairs will not pose a significant radiological hazard to the public. The Board also reaffirms its previous grant of summary disposition of all contentions involving the proposed repairs and authorizes the Director of Nuclear Reactor Regulation to issue the appropriate license amendments.

FINAL ORDER

In the Board's May 28, 1981 Memorandum and Order granting summary disposition of all contentions, the parties were directed to address the facts regarding the handling, storage, transportation or other disposition of low-level solid waste to be generated by the proposed steam generator repairs. The parties were also asked to state their positions on whether the Board can or should take any action regarding this matter, including the imposition of license amendment conditions (pp. 42-43).

The Staff replied on June 12, 1981, taking the position that the Board possesses the authority to impose appropriate license conditions, but that the onsite storage of repair - generated low-level solid waste does not pose an undue risk to public health and safety, even in the event of a hurricane or tornado. The affidavit of the project manager, Marshall Grotenhuis, was filed in support of this conclusion.

The Licensee, Florida Power and Light Company (FPL), filed an affidavit by Alan J. Gould, employed by it as a Power Resources Radwaste and Radiochemistry Specialist. Detailed facts and commitments concerning the handling, storage, transportation and disposition of low-level solid wastes were set forth, supporting the conclusion that even if all the drums in which such waste was stored were breached by a hurricane or tornado, the resultant doses would be below 10 CFR Part 20 limits.

The Intervenor took the positions that FPL should be required to submit an application under 10 CFR § 20.302 for its proposed disposal procedures, with opportunity for comment. The Staff should be required to prepare an FES on generic low level waste disposal, with the repairs delayed, the EPA appointed as the lead agency, and comments solicited from interested agencies. The Staff should be directed to contact appropriate State of Florida agencies concerning the completion of certain low level waste disposal studies. The repairs should be prohibited because there is no legal manner in which to dispose of such low-level wastes. Discovery should be reopened on this issue.

The Licensee has modified its low-level radioactive waste (LLW) management in view of the recent restrictions placed on storage at several burial sites, such as Barnwell, South Carolina.[1] Priority of offsite shipment is to be given to materials with higher specific radioactivity, while materials retained at site will contain relatively low concentrations of radioactivity. FPL may also obtain additional burial allocation from the "first come, first serve" pool at Barnwell. It is expected that an additional allocation of between 700 cu-ft to over 1,000 cu-ft will be available to FPL each month. FPL is seeking a permit for shipment of LLW to an alternate waste disposal facility.

The solid low-level waste generated by the steam generator repairs will be handled by the same procedures as low-level waste which is generated from routine plant operation and maintenance. These provisions include

[1]Affidavit of Alan J. Gould, dated June 12, 1981.

the compaction of dry radioactive compressible trash, such as rags, paper and clothing, using a highly efficient waste compacter in order to reduce the volume. Waste which contains a relatively high concentration of radioactivity is kept inside the Turkey Point Radwaste Building during the brief period (2 to 3 months) it is on site pending preparations for shipment and transportation. All solid low-level waste located on site will be monitored by portable monitors and swipe tests following approved procedures. Shipments offsite will comply with approved plant procedures and applicable Department of Transportation (DOT) and NRC regulations.

The low concentration LLW retained on site will be packaged as follows:

(1) Compressible trash is compacted into wooden boxes known as LSA boxes. These meet the criteria of a strong, tight package under 49 CFR Part 173. These boxes are lined with steel plates and plastic liners. The lids of filled boxes are nailed in place, and a steel lid cover is then nailed over the previous lid. The entire box is cross banded with five steel straps. A plastic cover is then placed over the entire box, and the box is rebanded with another five steel straps.
(2) Noncompressible solid waste with low concentration of radioactivity would normally be packaged in steel drums meeting DOT specifications for Type A packaging in accordance with 49 CFR Parts 173 and 178. Drum lids are clamped into place and held securely by a bolting ring.

LSA boxes with relatively low concentrations of radioactivity will be tied or banded together in blocks of four, providing a subassembly weighing approximately 16,000 lbs. They will stacked no more than two high. Plastic covers and/or tarps will be used to protect these containers from storms. Tie downs will be used for groups of these subassemblies to hold them in place in the event of hurricanes or tornadoes.

Drums containing LLW will be palletized and tied or banded together in groups of four. They will be stacked no more than two high. When stacked, the top and bottom subassemblies will be tied or banded together, providing an assembly weighing approximately 4,000 lbs. Tie downs will be used for these assemblies to hold them in place during storms. All of the drums which cannot be expeditiously shipped will be located within the Turkey Point 3 and 4 Radiation Controlled Area (RCA) at elevation 17.5 ft. MLW and will be appropriately secured.

The total estimated volume of LLW with low concentration of radioactivity that might be retained on site during the repairs, including the 1,312 drums now on site, is approximately 45,600 cu-ft. This represents a total estimated quantity of radioactivity of about 23.2 Ci. This could be reduced by additional Barnwell burial allocation or disposal at an alternate site.

The protective measures noted above make it extremely unlikely that the packages would be breached during a hurricane or tornado. The Gould affidavit cites analyses of a hypothetical LLW container breach that show the radioactive disposal consequences to the public are insignificant because the concentrations are so low to begin with.

The Staff provided an affidavit by Marshall Grotenhuis, dated June 9, 1981, on low level waste management. The handling, storage, transportation or other disposition of low-level solid waste from the steam generator repair will be the same as the processing of such waste during normal operation. With the unit under repair not producing waste from normal operation, the total waste from the plant (two units) is approximately the same during repairs as during normal two unit operation (FES, § 4.1.2.1.).

Reference was made to the Staff's accident analysis in FES, Sections 4.4 and 8.6.5, which considered a range of accidents and enumerated only the limiting cases. The hypothetical dispersal of LLW wastes was compared to the analysis of radioactive exposure consequences resulting from a steam generator lower assembly (SGLA) drop accident with the welded cover breaking loose. Dispersal of radioactivity into the atmosphere and by water pathway was considered.

The Staff estimated that a site boundary dose of 1.5 mrem could result from an accident which released into the atmosphere all of the radioactivity in the LLW from the repair of one unit. While there is no specific regulation governing a release of this type, we note that a release to the atmosphere giving a dose of 1.5 mrem is well within the limits set forth in 10 CFR Part 50, Appendix I, governing the design objectives for yearly doses produced by the normal operation of light-water-cooled power reactors.[2] The Staff also estimated the concentration of radionuclides which would result if all the LLW were washed into the cooling canals. The estimate of 1.4×10^{-5} uCi/cm^3 is within the limits set forth in 10 CFR Part 20, Appendix B, for releases to uncontrolled areas, as the Staff observed.

Based upon the foregoing considerations, the Board concluded that the impact of a hurricane or tornado on the LLW produced by the steam generator repairs and stored temporarily on site at Turkey Point will not pose a significant radiological hazard to the public.

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[2]The Staff suggested that the 1.5 mrem release to the atmosphere was acceptable because it was "clearly bounded by the SGLA breach accident, the limiting accident for purposes of evaluation" (Grotenhuis affidavit at 3). However, the SGLA drop accident was judged to be acceptable on the basis of a risk analysis, not on the basis of consequences alone (SER, Section 3.4.2; FES, Section 4.4). It is therefore deemed to be inappropriate and unacceptable for the Staff to attempt to compare the consequences of one accident with the risk analysis of another.

The Board has reviewed the 12 statements of position filed by the Intervenor, and considers that they are inapposite for the following reasons:

Position I. This Board was constituted to rule on the application for amendments to FPL's OL. We are not authorized to require FPL to apply for a license pursuant to 10 CFR 20.302

Position II. Irrelevant, considering Position I.

Position III. This is not a major federal action, and we have no jurisdiction over the EPA.

Position IV. Solid waste issue was addressed in the FES, Section 4.1.2.2.

Position V. Irrelevant considering Position IV.

Position VI. Irrelevant considering Position IV.

Position VII. The Board has no jurisdiction over disposition of low-level wastes not generated by the repair itself, nor of matters within the sole purview of the State of Florida.

Position VIII. Part 61B is in Rulemaking Status, and not cognizable in this proceeding. 10 CFR 40.11 and 70.11 are not relevant to power stations. 10 CFR 30.11 does not require this license to "reveal" whether it intends to apply for an exception.

Position IV. The shipment of waste from this site is governed by the appropriate regulations. The matter is not covered by the application for license amendments which is before this Board.

Position X. Licensee has not proposed that it become a waste storage facility. FPL has indicated that it has initiated plans to obtain additional burial allocation at Barnwell, on a "first come, first serve" basis.

Position XI. The question of LLW disposal for the life of the plant is beyond the scope of this proceeding.

Position XII. The Intervenor has had ample opportunity for discovery within the time frame of all issues including LLW and his failure to utilize it in a timely fashion cannot now be asserted for purposes of delay.

The Board has previously granted motions for summary disposition of all contentions, and therefore canceled the evidentiary hearing. Such action is reaffirmed fo: the reasons set forth in our Memorandum and Order entered May 28, 1981. The parties were therein also directed to file detailed information concerning the handling, storage, transportation of other disposition to be made of low level solid waste that might be generated by the proposed repairs. All of the parties have now submitted such information. For reasons discussed above, the Board has concluded that the impact of a hurricane or tornado on LLW to be stored at Turkey Point during the proposed repairs would not endanger the health and safety of the public.

ORDER

For all the foregoing reasons and based upon a consideration of the entire record in this matter, it is this 19th day of June, 1981

Ordered

That the evidentiary hearing previously scheduled concerning the Proposed Amendment to Facility Operating Licenses Nos. DPR-31 and DPR-41 to Permit Steam Generator Repairs, is permanently canceled, and the Director of Nuclear Reactor Regulation is authorized to issue appropriate license amendments to permit the proposed steam generator repairs of Turkey Point Nuclear Units 3 and 4, in accordance with the commitments made by the Licensee in its application and further described in the Affidavit of Alan J. Gould, dated June 12, 1981.

It is further Ordered, in accordance with 10 CFR 2.760, 2.762, 2.764, 2.785 and 2.786, that this Final Order shall be effective immediately and shall constitute the final action of the Commission forty-five (45) days after the issuance thereof, subject to any review pursuant to the above-cited Rules of Practice. Exceptions to this Final Order may be filed within ten (10) days after service of this Final Order. A brief in support of any such exceptions must be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff). Within thirty (30) days of the filing and service of the brief of the Appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

THE ATOMIC SAFETY AND
LICENSING BOARD

Dr. Emmeth A. Luebke
ADMINISTRATIVE JUDGE

Dr. Oscar H. Paris
ADMINISTRATIVE JUDGE

Marshall E. Miller, Chairman
ADMINISTRATIVE JUDGE

Cite as 13 NRC 1122 (1981) LBP-81-17

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

John F. Wolf, Chairman
Mr. Glenn O. Bright
Dr. Jerry R. Kline

In the Matter of	**Docket Nos. 50-275-OL**
	50-323-OL
	(Low Power Test Proceeding)
PACIFIC GAS & ELECTRIC COMPANY	
(Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2)	**June 19, 1981**

The Licensing Board denies intervenors' motion to reopen the record for consideration of "Class 9" accidents at Diablo Canyon on the ground that no special circumstances exist to warrant such consideration.

MEMORANDUM AND ORDER DENYING JOINT INTERVENORS MOTION TO REOPEN ENVIRONMENTAL RECORD FOR CONSIDERATION OF CLASS NINE ACCIDENT

Following the occurrence of the Three Mile Island accident, Joint Intervenors filed on May 9, 1979 a motion with the Board to reopen the record for further consideration of "Class 9" accidents at Diablo Canyon. On May 24, 1979, the Staff proposed that the Board defer ruling on the motion pending completion of the Staff report on TMI and its specific implications to this case. On June 5, 1979, the Board agreed to the Staff's proposal. The Boards now finds that it has sufficient information to rule on the motion.

In November 1980, the Commission published NUREG-0737, "Clarification of TMI Action Plan Requirements," which sets forth those items of the total TMI-related actions contained in the Staff's TMI Action Plan, NUREG-0660, which have been approved by the Commission for implementation by licensees of operating reactors and Applicants for operating licenses. The Board has also received guidance from the Commission laying out the procedures to be used in our application of NUREG-0737 in our licensing process (CLI-80-42, December 18, 1980, and CLI-81-5, dated April 1, 1981). We have carefully reviewed these documents, and find that none of the requirements therein impact the Commission's interim policy on accident considerations. We therefore proceed with our analysis of the Diablo Canyon situation.

On May 16, 1980, the Commission issued a statement of interim policy which provided guidance on consideration of Class 9 accident analysis with respect to plants for which Final Environmental Statements had been issued. This guidance stated that in these cases consideration of Class 9 accidents need not be addressed absent a showing of special circumstances. The Commission noted that in the past the Staff has identified such special circumstances as falling within three categories: (1) high population density around the site; (2) a novel reactor design; or (3) a combination of a unique design and a unique siting mode. Diablo Canyon does not fall into any of these categories (cf. DD-80-22, 11 NRC 919 (1980)).

The Commission had earlier noted that in addition ot these three criteria that proximity of a plant to a "man-made or natural hazard" might also represent "the type of exceptional case that might warrant additional consideration" (Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), CLI-80-8, at 434-435 (March 21, 1980)). In response to this guidance, the Board believed that the known seismicity of the State of California might constitute such a natural hazard.

The Board conducted exhaustive hearings on the effects of seismic forces on the Diablo Canyon plants from December, 1978 through February, 1979. In our Partial Initial Decision issued September 27, 1979, we found that

> "... the evidence demonstrates that all structures, systems and components of the Diablo plant necessary for continued operation without undue risk to the health and safety of the public will remain functional and within applicable stress and deformation limits when subjected to the effects of the operating basis earthquake in combination with normal operating loads."

In June, 1980 the Atomic Licensing and Appeal Board reopened the record to receive new evidence not available to the Licensing Board at the time they issued their decision. After conducting a thorough in-depth review of both the new evidence and the evidence before the Licensing Board, the Appeal Board affirmed the Licensing Board's findings (ALAB-644, June 16, 1981). We must, therefore, conclude that even though Diablo Canyon is located in a region of known seismicity, the probability of it sustaining a "class nine" accident is no greater than for any other reactor. Thus no special circumstances exist, and the motion to reopen the record for consideration of class nine accidents is *denied.*

On the 19th day of June, 1981 it is

ORDERED

that the motion to reopen the record for consideration of class nine accidents is denied.

FOR THE ATOMIC SAFETY
AND LICENSING BOARD

John F. Wolf, Chairman
ADMINISTRATIVE JUDGE

Issued at Bethesda, Maryland
this 19th day of June, 1981

Cite as 13 NRC 1125 (1981) DD-81-9

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of | **Docket No. 50-322 (10 C.F.R. 2.206)**

LONG ISLAND LIGHTING COMPANY
(Shoreham Nuclear Power Station, Unit 1)

June 26, 1981

The Director of Nuclear Reactor Regulation denies a petition under 10 C.F.R. 2.206 which requested suspension of facility construction pending determination of an application for extension of the construction permit.

ADMINISTRATIVE PROCEDURE ACT: TIMELY APPLICATION FOR RENEWAL

Under the APA and Commission's regulations, a construction permit or other license generally remains effective until the Commission has finally determined the application.

ATOMIC ENERGY ACT: RESOLUTION OF SAFETY ISSUES

Institution of proceedings prior to consideration of an operating license is not mandated even if unresolved safety questions are raised after issuance of a construction permit. Continued construction does not itself pose any danger to public health and safety.

ATOMIC ENERGY ACT: SAFETY STANDARDS

An applicant will be required to take all actions necessary to ensure safety. Safety standards may not be compromised by consideration of the cost or difficulty associated with implementing measures required for safety.

RULES OF PRACTICE: PETITIONS UNDER 10 C.F.R. 2.206

To the extent a petition under 10 C.F.R. 2.206 raises matters which require resolution in other proceedings, those matters will be addressed in those proceedings and not under 10 C.F.R. 2.206.

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

In filings dated December 31, 1980, and January 23, 1981, the Shoreham Opponents Coalition (SOC) requested pursuant to section 189 of the Atomic Energy Act of 1954, as amended, and 10 C.F.R. 2.206 of the NRC's Rules of Practice that the Director of Nuclear Reactor Regulation institute a proceeding to determine whether good cause exists to extend the construction permit for the Shoreham Nuclear Power Station, Unit 1. SOC also requested "that, to protect public health and safety, the Shoreham construction permit be suspended pending the outcome of the hearing [on the construction permit extension]." Petition at 1 (Jan. 23, 1981). The Long Island Lighting Company (LILCO) had requested on November 26, 1980, an extension of Construction Permit No. CPPR-95 to March 31, 1983.[1] By separate memorandum, the NRC staff has made recommendations to the Commission with respect to SOC's request for a hearing on the extension of the construction permit.[2] The remainder of this decision is concerned with SOC's request that I suspend construction of the Shoreham facility pending the outcome of the proceeding on extension of the construction permit.

SOC claims that suspension of the permit should be ordered "to protect public health and safety". At no point in the petition does SOC give reasons why public health and safety would be threatened imminently if permit suspension were not ordered. To be sure, SOC lists a number of matters which it believes should be considered in connection with the application

[1] *See* Attachment A to Petition (Jan. 23, 1981). The construction permit would have expired on December 31, 1980. Under 10 C.F.R. 2.109, which derives from, section 9(b) of the Administrative Procedure Act, 5 U.S.C. 558(c), the permit remains in effect until the application for its renewal has been finally determined.

[2] A copy of this memorandum has been served with this decision on SOC and LILCO. SOC's petition lists a number of items which SOC believes should be litigated in a hearing on the construction permit extension or should be imposed as conditions on any permit extension. Because SOC has requested that these matters be litigated in the permit extension proceeding, the Staff will respond to these matters in the proceeding on permit extension, not under 10 C.F.R. 2.206. *See Pacific Gas & Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), CLI-81-6 (May 8, 1981).

for permit extension.[3] These matters concern, however, primarily issues that go to the question of whether LILCO should be granted an operating license for the Shoreham plant. Whether or not these matters are litigable in a proceeding on permit extension, they do not reveal any threat to public health and safety that stems from the facility's construction. Rather, SOC has alleged only that *operation* of the facility would be unsafe or environmentally unsound, because of the facility's siting, the risk of severe accidents, and the need for additional safety systems and analyses. Thus, the petition does not raise allegations that might provide a basis for suspension, perhaps even immediate suspension, of construction: *e.g.*, construction of the facility has been improper under existing requirements or implementation of the quality assurance assurance program has been inadequate.[4]

The only nexus between any of the matters raised by SOC and its request for immediate suspension of the permit is SOC's request that suspension of the permit be ordered pending a determination of the feasibility of evacuation after a severe accident during operation of the facility.[5] SOC's citation to a recent Appeal Board decision is inapposite as a basis for SOC's request. *Northern Indiana Public Service Co.* (Bailly Generating Station, Nuclear 1), ALAB-619, 12 NRC 558, 569-70 (1980). The *Bailley* decision suggests only that it may be appropriate to consider site suitability contentions in a proceeding on construction permit extension, not that suspension of construction pending resolution of such issues in the permit extension proceeding is appropriate. The feasibility of evacuation, as it relates to emergency planning, is relevant to the assessment of whether the plant should operate. Although that issue must be resolved before operation of the facility, evacuation considerations pose no imminent threat to public health and safety that would warrant immediate suspension of construction.

Suspension of construction is not mandated, therefore, by law or Commission policy. As noted above, a construction permit or any other Commission license generally remains effective under a timely application for renewal until the Commission has finally determined the application.[6] The permittee pursues construction work under a construction permit at its

[3]In part, the petition styles these matters as arguments for "revocation" of the construction permit. Petition at 4-20 (Jan. 23, 1981). However, SOC wants these matters litigated in the construction permit proceeding. If these matters are litigated in that proceeding and if SOC's views prevail, extension would be denied and thereby the permit would be terminated.

[4]*See Proposed General Statement of Policy and Procedure for Enforcement Actions*, § IV.C., 45 Fed. Reg. 66,754, 66,757 (Oct. 7, 1980).

[5]Petition at 20 (Jan. 23, 1981).

[6]10 CFR 2.109; 5 U.S.C. 558(c).

own risk pending approval of permit extension or of the application to operate the plant.[7] Even where unresolved safety questions are raised after issuance of the construction permit, institution of proceedings to suspend the permit is not required, because "permitting continued construction of the plant despite unresolved safety questions does not of itself pose any danger to the public health and safety".[8] Before LILCO may receive an operating license, it will be required to do anything necessary to ensure safe operation of the plant. The cost or difficulty associated with implementing needed actions to ensure safety are not relevant consideration to this agency. The safety standards which an applicant must meet to obtain an operating license are unconditional.[9] To the extent that SOC has raised matters which require resolution before an extension of the construction permit is granted or before an operating license is issued, these matters will be given appropriate consideration in those proceedings. I do not find further consideration of these matters appropriate at this time under 10 C.F.R. 2.206.[10]

As SOC's petition does not provide an adequate basis for immediate suspension of construction, SOC's petition to suspend is *denied*. The remaining matters in the petition concerning SOC's request under section 189 of the Atomic Energy Act for a hearing on permit extension are before the Commission for action. A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 C.F.R. 2.206(c). In accordance with 10 C.F.R. 2.206(c), this decision will constitute the final action of the Commission 25 days after the date of issuance, unless the Commission on its own motion institutes review of this decision within that time.

Harold R. Denton, Director

Office of Nuclear Reactor Regulation

Dated in Bethesda, Maryland
this 26th day of June, 1981

[7] *See Power Reactor Development Co. v. International Union of Electrical, Radio & Machine Workers*, 367 U.S. 396 (1961).

[8] *See Porter County Chapter of the Izaak Walton League, Inc. v. NRC*, 606 F.2d 1363, 1369 (D.C. Cir. 1979).

[9] *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 & 2), ALAB-623, 12 NRC 670, 677-78 (1980).

[10] *See Pacific Gas & Electric Co.*, (Diablo Canyon Nuclear Power Plant, Units 1 & 2), CLI-81-6 (May 8, 1981), *affirming* DD-81-3, Pt. I (March 26, 1981); *Commonwealth Edison Co.*, (Byron Station, Units 1 & 2), DD-81-5, Slip Op. at 2-4 (May 7, 1981).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

Victor Stello, Jr., Director

In the Matter of **Docket Nos. STN 50-546**
STN 50-547
(10 CFR 2.206)

PUBLIC SERVICE COMPANY OF INDIANA, *et al.*
(Marble Hill Nuclear Generating Station, Units 1 & 2) **June 26, 1981**

The Director of Inspection and Enforcement denies under 10 CFR 2.206 a petition which requested that the Director withdraw his authorization to permit certain construction on the Marble Hill project.

DIRECTOR'S DECISION UNDER 10 CFR 2.206

The Commission has referred a petition filed by Save the Valley on April 14, 1981 to the Director of the Office of Inspection and Enforcement for consideration under 10 CFR 2.206. In its petition, Save the Valley requested that the Commission review the Director's March 27, 1981 authorization that permitted resumption of certain concrete construction work on the Public Service Company of Indiana (PSI) Marble Hill project.[1] Save the Valley contends that the Director's action constituted an abuse of discretion for the following reasons:

[1]Counsel for Save the Valley styles the petition as a request for review under 10 CFR 2.206(c) of the Director's decision to permit resumed construction. The Director's authorization was not, however, an action pursuant to 10 CFR 2.206. The authorization did not involve a denial of a petition filed by Save the Valley and was made in accordance with the terms of the Director's *Graduated Rescission of Order Dated August 15, 1979* (May 15, 1980) which outlined the terms of rescission of the original *Order Confirming Suspension of Construction (August 15, 1979).*

(1) The independent engineering consultants to NRC have not issued a final written report on existing concrete at Marble Hill;

(2) Contrary to an alleged commitment by Region III, Save the Valley has not had an opportunity to review and comment on the final version of the consultants' report before resumption of construction;

(3) Sargent & Lundy's report for PSI does not meet the structural integrity criteria specified by the NRC; and

(4) The Director permitted resumption of construction without briefing the Commission in accordance with the Commission's Memorandum and Order of March 13, 1980 [CLI-80-10, 11 NRC 438 (1980)].

Save the Valley's contentions are without merit, and I have therefore denied its petition requesting that I halt renewed construction on the basis of Save the Valley's contentions.

Contrary to Save the Valley's impression, resumption of concrete placement and related construction activities was not dependent on the terms that Save the Valley suggests: i.e., on final, written findings and conclusions of NRC's independent consultants and receipt of Save the Valley's comments on the consultants' report.[2] Resumption of construction at Marble Hill has been predicated on PSI's satisfactory completion of the steps outlined in the Graduated Rescission of Order Dated August 15, 1979, which I issued on May 15, 1980.[3]

The Office of Inspection and Enforcement regards the work of its consultants as important to the overall confidence that can be attached to the quality of Marble Hill's construction. Accordingly, the consultants' interim reports and informally transmitted comments have been factored into the decision to permit resumption of concrete construction at Marble Hill. To suggest, as Save the Valley does, that receipt of the consultants' final conclusions was an absolute precondition to resumption of construction goes far beyond the intent of the rescission program and would not be warranted under the circumstances. The NRC consultants have reported to the inspection staff that the concrete at Marble Hill is of acceptable quality and that the methodology used in Sargent & Lundy's report for PSI meets the required level of 95% reliability and 95% confidence. This information was conveyed to Dr. Cassaro, technical adviser to Save the Valley.

[2]Region III staff made no commitment to Save the Valley to provide them an opportunity to review and comment on the final version of the consultants' report before resumption of construction.

[3]Mr. Dattilo, counsel to Save the Valley, received a copy of this letter and all other formal correspondence referred to in this decision.

A copy of the final report will be provided to Save the Valley and copies will be made available to other members of the public when the report is issued. In light of the consultant's preliminary findings and PSI's satisfactory completion of the required steps under the rescission program, my decision to permit resumed construction was appropriate. If the consultants' final report differs for some reason from their preliminary conclusions or if PSI does not demonstrate that it is satisfactorily implementing its construction program, I shall take appropriate action in accordance with the rescission program and my enforcement authority under 10 CFR Part 2 and the Commission's interim enforcement policy (45 FR 66754, October 7, 1980).

As its third basis for relief, Save the Valley contends that Sargent & Lundy's report for PSI regarding the integrity of concrete does not meet NRC structural integrity criteria of 95% reliability and 95% confidence. Save the Valley points to concerns raised by its technical advisor, Dr. Cassaro, in his letters to Region III dated September 26, 1980, March 4, 1981, and March 26, 1981. The concerns contained in Dr. Cassaro's September 26, 1980 and March 4, 1981 letters were responded to in detail in a letter to him dated March 20, 1980, from James G. Keppler, Director of NRC Region III. The staff has also had informal conversations with Dr. Cassaro. With respect to Dr. Cassaro's March 26 letter, in which he took exception to points in Region III's March 20 letter, NRC invited Dr. Cassaro and his associate to travel to Region III, at NRC's expense, to further discuss the basis of NRC's decision regarding concrete quality at Marble Hill. Since Dr. Cassaro declined the offer, Region III will provide him with a further written response. In any event, all of the concerns addressed in Dr. Cassaro's letters have been reviewed and, for the reasons discussed in the March 20 letter to him, I have concluded that the Sargent & Lundy report meets NRC criteria.

The Commission has informed Save the Valley that the fourth basis for relief, an alleged failure of the Director to brief the Commission, is without merit. In accordance with the Commission's Memorandum and Order of March 13, 1980, CLI-80-10, 11 NRC 438 (1980), the Director briefed the Commission on May 7, 1980, with respect to the intended course of action in permitting resumed construction at Marble Hill. Mr. Dattilo, counsel for Save the Valley, attended that briefing.

For the reasons stated in this decision, Save the Valley's petition is denied. A copy of this decision will be filed with the Secretary for the Commission's review in accordance with 10 CFR 2.206(c). As provided in 10 CFR 2.206(c), this decision will become the final action of the agency 25 days after issuance unless the Commission determines to institute a review of this decision within that time.

Victor Stello, Jr., Director
Office of Inspection and
Enforcement

Dated at Bethesda, Maryland
this 26th day of June, 1981

Cite as 13 NRC 1133 (1981) DD-81-11

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of **Docket Nos. 50-259**
50-260
50-296
(10 C.F.R. 2.206)

TENNESSEE VALLEY AUTHORITY
(Browns Ferry Nuclear Plant, Units 1, 2, & 3) **June 26, 1981**

The Director of Nuclear Reactor Regulation denies a request that the Director reconsider the issuance of amendments to the Browns Ferry licenses which permit limited storage of certain low-level radioactive wastes.

TECHNICAL ISSUES DISCUSSED: APPLICABLE STANDARDS FOR PROTECTION OF WASTE SYSTEMS

Waste systems are judged against General Design Criteria 60 and 61 of Appendix A to 10 C.F.R. Part 50.

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

In a letter dated October 28, 1980, Messrs. Thomas W. Paul, Stewart Horn and David Ely, on behalf of the Huntsville Chapter, Safe Energy Alliance of Alabama (SEAA), requested that NRC reconsider the issuance by the NRC of amendments Nos. 60, 55 and 32 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Unit Nos. 1, 2 and 3. These amendments were issued by the NRC on March 17, 1980 and authorized TVA to temporarily store low-level radioactive waste (LLRW) in an existing covered pavilion on the Browns Ferry site.

In their letter of October 28, 1980, the SEAA stated the reasons why we should reconsider the authorization and these are summarized below:

1. The area of northern Alabama where the BFNP is located is subject to frequent, destructive tornado activity.
2. On April 3-4, 1974, a series of tornadoes passed within 2 miles of the BFNP. Fifty-eight (58) 500 KV line transmission towers carrying power from BFNP were snapped. As a result of the loss of these power lines, one unit at BFNP was forced to shutdown since the plant was not able to distribute the total power capable of being generated by the plant.
3. During the April 3-4, 1974 outbreak of tornadoes, the rotational wind speed at some locations was estimated to be between 200 and 250 mph.
4. Despite this history of very recent serious tornado activity, TVA, in their application requesting authorization for temporary onsite storage of LLRW, had concluded - on a probabilistic basis - that design of the drum restraint and hold-down system for wind speeds of 95 mph was adequate, considering the relatively short period of time drums of LLRW might be stored in the building. Specifically, TVA concluded that the probability of a tornado with maximum wind speeds higher than the 95 miles per hour value striking the plant in any one year is 7×10^{-4}. TVA considered this small enough to be neglected.
5. Despite statements that temporary storage of LLRW in the building will comply with all applicable Commission regulations, the building and drum restraint system were not designed in accordance with 10 CFR 50, Appendix A, Criterion 2 - *Design bases for protection against natural phenomena.*

The NRC staff comments on the above points are summarized below:

1. The NRC staff, in conjunction with other government agencies, keeps track of all reported tornadoes. Alabama, along with most other southern, mideastern and midwestern states, is prone to be subject to frequent, severe tornadoes. Regulatory Guide 1.76 describes a design basis tornado acceptable to the Regulatory staff for each of three regions within the contiguous United States that structures, systems and components in a nuclear plant *important to safety* (emphasis added) should be designed to withstand. All of the United States east of the Rocky Mountains is classified as Region I. The recommended set of properties defining a design basis tornado in this Region I is the strictest for any region of the country.
2. On April 3-4, 1974, there was an outbreak of 148 tornadoes within a 24 hour period in 13 states and Canada. This is by far the largest number of tornadoes within a 24 hour period on record. At the height of activity, 15 tornadoes were on the ground simultaneously. As SEAA pointed out, over 300 people were killed. The tornadoes ranged from Mississippi, Alabama and Georgia in the south to Illinois, Indiana, Ohio and Michigan in the north. There were two approximately parallel tornadoes

that swept a path that extended from Mississippi, through northern Alabama and into Tennessee, both of which crossed the Tennessee River to the east of the BFNP in the general area between Athens, Alabama and Huntsville, Alabama. The first tornado was named First Tanner and the second tornado was dubbed Second Tanner. First Tanner touched down at 1820 hours CST and lifted off about 61 minutes later, traversing a path approximately 51 miles long, with a width of 1/8 to 1/4 mile on the average. Second Tanner touched down at 1930 hours, lasted for about 55 minutes and swept across a path approximately the same length and width as First Tanner. Tornadoes are generally rated on a scale of 1 to 5, based on windspeed, path length and path width, with a rating of "5" being the most severe. There was a short section in the overall path of the First Tanner tornado north of Wheeler Reservoir and east of the BFNP assigned a damage category "5". As pointed out by SEAA, this tornado knocked-out the 500 KVa transmission system, causing a shutdown of Unit 1; Unit 2 was undergoing preoperational testing at the time and Unit 3 was still under construction. At no time did the loss of offsite transmission lines affect the capability to safely shutdown the reactor facility and maintain it in a safe shutdown condition. Browns Ferry Unit 1 resumed partial operation the next day when the 500 KVa West Point line was restored to service.

3. There is no question that the Browns Ferry site is located in an area occasionally traversed by tornado storms. Wind speeds in excess of 40 mph are occasionally reported but wind speeds in excess of 80 mph are rare. During the design of the Browns Ferry facility, we thoroughly evaluated the meteorological conditions at the site. We have rereviewed the straight-line winds and tornado winds that structures at the Browns Ferry site might possibly be subjected to. A determination of the wind hazard probability for a given site consists of separate estimates of windspeed as a function of recurrence interval (or probability per year) for straight-line winds and tornado winds. The two sets of data are not from the same statistical population and, thus, cannot be combined into a single data set. Two curves arise: (1) determination of the expected value of the fastest mile per hour wind using the windspeed data collected at a given site; this curve is generally accepted to be of the extreme value type I distribution; (2) determination of the expected value of windspeeds arising from tornadoes which involves tornado occurrence rates, path length and width, and some measure of the intensity (strength) of the individual tornadoes that comprise the data set for a given meteorologically and topographically homogeneous region. The two curves are not identical. For low probabilities (<1 ×

10^{-4}/yr), tornado windspeeds are greater than those projected from the straight-line wind data; for high probabilities, the straight-line winds are greater than tornadic winds for a given probability. For a site such as Browns Ferry, Alabama, the straight-line winds dominate the probabilities through about 100 mph corresponding to 1×10^{-4}/yr. For a 95 mph windspeed, the probability for this to be from straight-line winds is as above, but for it to be from tornadoes the probability decreases to 5×10^{-5}/yr. Thus, the probability of seeing 95 mph from straight-line winds is higher than seeing 95 mph in a tornado in this area. This is explained, in part, by the fact that tornadoes must occur first in order for 95 mph winds to exist from them; and the tornado occurrence rate in this area is about 1×10^{-4}/yr. In other words, the probability that a tornado will strike the facility is about once every 10,000 years. The probability of a structure at the Browns Ferry site being subjected to a wind speed of a certain velocity can be approximated from the following:

Mean Recurrence Interval	Expected Probability	Windspeed, mph	Type of Wind
10 years	10^{-1}	60	Straight wind
100 years	10^{-2}	70	Straight wind
1000 years	10^{-3}	85	Straight wind
10,000 years	10^{-4}	100	Straight wind
100,000 years	10^{-5}	150	Tornado wind
1,000,000 years	10^{-6}	210	Tornado wind
10,000,000 years	10^{-7}	260	Tornado wind

4. General Design Criterion 2 of Appendix A to 10 CFR Part 50, requires, in part, that structures, systems and components in a nuclear plant

important to safety (emphasis added) be designed to withstand the effects of natural phenomena, such as tornadoes, without loss of capability to perform their safety function. For BFNP, and other nuclear plants, structures and equipment whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the facility and the removal of decay heat are classified as Class I structures. Class II structures and equipment are defined as those which are necessary for station operation but are not essential to a safe shutdown. The classification of structures and equipment - and the basis therefore - is discussed in TVA's Final Safety Analysis Report (FSAR) for the BFNP and in the Commission's Safety Evaluation Report dated June 26, 1972. We have concluded that the structures and equipment at BFNP are appropriately classified. Class I structures at BFNP are designed for normal dead and live loads, 100 mph wind, 300 mph tornado wind and 3 psi pressure drop, operating and design basis earthquakes of 0.1g and 0.2g maximum ground accelerations, respectively. Soil, hydrostatic and missile loads have also been included. Facilities or structures that are used solely for the storage of LLRW are not classified as Class I structures and are not required to be designed to these loads. In light of the limited hazard involved with these wastes, see paragraph 6, we believe that the pavilion need not be designed for any particular loading.

5. The applicable regulatory standards for protection of waste systems are 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 60 and 61, which provide:

   "The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences" and The ... radioactive waste and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed ... with suitable shielding for radiation protection and with appropriate containment, confinement and filtering systems.

   Your petition does not raise any issue with respect to normal operations and for the reasons discussed in paragraph 6, below, we believe that the storage activity is adequately protected against postulated accidents, including those resulting from postulated tornados.

6. The possible reoccurrence of a tornado at BFNP was considered in TVA's application and the NRC's safety evaluation related to the

amendments in question. Such consideration is reflected in the condi-tions associated with the temporary storage of LLRW in the pavilion

(a) Only dry, compacted or noncompacted trash may be stored in the pavilion. Spent ion exchange resins or evaporator bottoms (which might contain liquids and which are the only wastes that usually contain any significant amount of radioactivity) are not authorized to be stored in the pavilion.
(b) The amount of radioactivity in any drum of waste stored in the pavilion is limited to 0.5 curies. The total amount of radioactivity that may be stored in the pavilion is limited to 1320 curies. The contact radiation dose rate at the surface of any drum must be less than 0.7 R/hour.
(c) All containers of trash placed in the temporary storage facility are to be held secure at all times by means of an installed restraint system This system has been designed to hold all containers secure during all severe environmental conditions up to and including the design basis event. The design basis event used by TVA was a basic wind velocity of 95 miles per hour with a 100 year recurrence frequency

As a prudent measure, TVA has adopted very low limits on the amount of radioactivity to be stored in each container and committed to installing a drum restraint system. The restraint system consists of heavy metal grates placed over a section of drums, with the grates anchored to the concrete slab. The restraint system would likely keep any drums from being carried offsite under all meteorological conditions except for the most severe postulated tornado.

The NRC staff had considered the potential impact if a drum (or drums) of LLRW stored temporarily in the pavilion were carried offsite by a tornado.

In this unlikely event, the radiological consequences of such an event are not likely to exceed the 10 CFR Part 20 annual exposure limit of 500 mrem Even in the most conservative case with a member of the public in direct contact with the surface of a drum with the highest allowable dose rate of 700 mrem/hr, it is unlikely the duration of the exposure in such close contact would be sufficiently long to exceed the 500 mrem limit. In practice most drums to be placed in the storage facility will not have the maximum 700 mrem/hr dose rate on contact. In addition, containers of waste are required to be labelled as containing radioactive material and such labelling, when seen by members of the public, is expected to cause a person to increase his (her) distance from the container. In the unlikely event a container or containers are carried offsite by a tornado, efforts to recover the container(s) will be initiated as quickly as possible by utility and local and state officials, limiting the time any member of the public might be exposed to radiation from the container(s).

If a container were to rupture, the possible exposure to a member of the public would likely be even less than the case where the container remained intact. The type of waste to be stored in the temporary facility is dry trash that is usually relatively uniformly contaminated with radioactive material. Thus, if the waste is scattered, the possible direct exposure from any one piece or several pieces of the waste is likely to be smaller than from a full container. Inhalation doses from a ruptured container would be small because of the small fraction of respirable sized particles of radioactive material released from the container and the dilution in air that would occur between the point of container rupture and the breathing zone of a downwind individual.

Based on the above, we have reevaluated the safety aspects of temporarily storing LLRW in the existing pavilion on the Browns Ferry site and particularly the effect on public health and safety from potential tornadoes striking the building. We have concluded that although the pavilion and drum restraint system are not designed to withstand the most severe potential tornadoes that might strike the temporary LLRW storage facility, the potential hazard to public health and safety from drums of waste being carried offsite and/or their contents being dispersed would be small. As discussed above, the storage of LLRW in the pavilion is intended to be a temporary measure until the waste can be shipped to a licensed disposal facility or stored onsite in NRC approved longer-term storage facilities.

Considering that the probability of a tornado with wind speeds greater than 95 mph striking the Browns Ferry site is in the order of once every 20,000 years, the restrictions on the type and activity levels of LLRW that can be stored in the pavilion, and our evaluation of the potential consequences to public health and safety if a tornado were to strike the temporary storage facility, I have concluded that the issuance of the amendments authorizing TVA to temporarily store LLRW in the onsite pavilion was a reasonable and safe action and that there are no safety reasons for modifying our previous determination.

Based on the foregoing discussion, I have determined that there exists no basis for reconsidering the issuance of Amendment Nos. 60, 55 and 32 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68. The request of Messrs. Thomas W. Paul, Stewart Horn and David Ely, on behalf of the Huntsville Chapter, Safe Energy Alliance of Alabama, is hereby denied.

A copy of this determination will be placed in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C. 20555, and at the Local Public Document Room for the Browns Ferry Nuclear Plant located at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of this document will also be filed with the Secretary of the Commission for its review in accordance with 10 CFR 2.206(c) of the Commission's regulations.

In accordance with 10 CFR 2.206(c) of the Commission's Rules of Practice, this decision will constitute the final action of the Commission 25 days after the date of issuance, unless the Commission on its own motion institutes the review of this decision within that time.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

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