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NUCLEAR REGULATORY COMMISSION ISSUANCES

August 1984

U.S. NUCLEAR REGULATORY COMMISSION

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

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

U.S. NUCLEAR REGULATORY COMMISSION

Prepared by the Division of Technical Information and Document Control,
Office of Administration, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555
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COMMISSION

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

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) **August 10, 1984**

The Commission determines that the circumstances in this case do not provide a basis for departure from its decision in *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-81-33, 14 NRC 1091 (1981), that (1) NRC regulations do not require consideration of the impacts of earthquakes on emergency planning, and (2) the determination of whether to amend the regulations to include the consideration of earthquakes should be addressed as a generic matter. The Commission decides to initiate such a rulemaking and, further, determines that the issuance of a full-power operating license in this proceeding need not be delayed until its conclusion.

OPERATING LICENSE: HEALTH AND SAFETY STANDARD

The Commission will not license a nuclear power plant unless it can make the statutorily required finding that operation of the plant will not result in undue risk to public health and safety.

DECISION

In CLI-84-4 (19 NRC 937 (1984)), the Commission requested the parties' responses to several questions bearing on whether the circumstances in this case warranted some specific consideration of the effects of seismic events on emergency planning. Responses were received from Pacific Gas and Electric Company (PG&E), the NRC staff, and Joint Intervenors.

After considering these responses, the Commission has determined that the information before it does not warrant departure from the decision in *San Onofre* that the NRC's regulations "do not require consideration of the impacts on emergency planning of earthquakes which cause or occur during an accidental radiological release," and that the determination of whether to amend the regulations to include the consideration of earthquakes should be addressed as a generic matter. *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-81-33, 14 NRC 1091, 1091 (1981).

Accordingly, for the reasons discussed below, the Commission has decided to initiate a rulemaking and has determined that the issuance of a full-power operating license need not be delayed until the conclusion of any such proceeding.

I.

The Commission's first question was whether emergency planning regulations can and should be read to require some review of the complicating effects of earthquakes on emergency planning for Diablo Canyon.

A. Parties' Views

PG&E and the NRC staff believe that the Commission should not read its emergency planning regulations and implementing guidance in NUREG-0654 so as to provide for any specific consideration of the complicating effects of earthquakes on emergency response, even in California. For the NRC staff, this appears to present a change from its previous view, expressed most clearly in 1981 in the *San Onofre* proceeding, that some limited consideration of the effects of earthquakes on emergency response was warranted in areas of high seismic activity, especially California.

PG&E's essential argument is that the Commission's emergency planning regulations implicitly include the complicating effects of earthquakes as part of the overall consideration of four classes of Emergency Action Levels established in NUREG-0654. In PG&E's view, consideration of the effects of earthquakes on emergency planning is subsumed within the consideration given to the effects of other natural phenomena having similar effects on emergency planning. PG&E is concerned that the explicit consideration of the effects of earthquakes on emergency planning will distort or preferentially align emergency plans to concentrate on earthquake-related emergencies. Therefore, PG&E believes that it would be redundant and contrary to established planning guidance to require an emergency plan to include consideration of specific accident sequences such as those associated with earthquakes.

The essential argument of the NRC staff is that there is an acceptably low risk to public health and safety associated with not requiring emergency plans to explicitly consider the complicating effects of earthquakes. This staff position is based on its belief that contemporaneous occurrence of an earthquake and a radiologic release has too low a probability to warrant mandatory consideration.[1]

Joint Intervenors take the contrary view that the NRC's regulations and implementing guidance require some consideration of the complicating effects of earthquakes on emergency response for the same reasons that the NRC staff has considered the effects of other natural phenomena on emergency plans.

B. Analysis

The Commission agrees with the NRC staff's analysis in this case. The focus of the emergency planning controversy among the parties is on the possible need to consider the contemporaneous occurrence of an earthquake and radiologic release from the plant. For earthquakes up to and including the Safe Shutdown Earthquake (SSE), the seismic design of the plant was reviewed to render extremely small the probability that such an earthquake would result in a radiologic release.[2] While a radiologic release might result from an earthquake greater than the SSE, the probability of occurrence of such an earthquake is extremely low.[3] In

[1] The details of the staff's position were described in its memorandum to the Commission of January 13, 1984 which was incorporated in CLI-84-4, *supra*.

[2] Indeed, Diablo Canyon has been subjected to special, unprecedented reviews of this issue.

[3] Joint Intervenors have recently moved the Appeal Board to reopen the record on the seismic design bases for Diablo Canyon to consider new seismic information. PG&E has opposed that request. Both

(Continued)

addition, as the NRC staff noted in its January 13, 1984 memorandum to the Commission on the generic subject of earthquakes and emergency planning, for those risk-dominant earthquakes which cause very severe damage to both the plant and the offsite area, emergency response would have marginal benefit because of its impairment by offsite damage. Thus, the Commission agrees with the NRC staff's conclusion that the expenditure of additional resources to cope with seismically caused offsite damage under those circumstances is of doubtful value considering the modest benefit in overall risk reduction which could be obtained.

There remains only the possibility of a contemporaneous occurrence of both a radiologic release from the plant caused by an event other than an earthquake, and an earthquake that would complicate emergency response. NUREG-0654 does call for some consideration of site-specific adverse or emergency conditions on emergency response. In prior cases, such frequently occurring natural phenomena as snow, heavy rain, and fog have been considered. With one exception, the focus has always been on frequently occurring natural phenomena.[4] The Commission believes, based on the information provided by the parties, that earthquakes of sufficient size to disrupt emergency response at Diablo Canyon would be so infrequent that their specific consideration is not warranted.

The Commission's view that it need not give specific consideration to the complicating effects of earthquakes on emergency planning in this case is bolstered by the following consideration. Specific consideration has been given in this case to the effects of other relatively frequent natural phenomena. The evidence includes the capability of the emergency plan to respond to disruptions in communication networks and evacuation routes as a result of fog, severe storms and heavy rain. In the extreme, these phenomena are capable of resulting in area-wide disruptions similar to some of the disruptions which may result from an earthquake. Testimony in the Diablo Canyon record indicates that adverse weather conditions such as the effect of heavy fog could increase evacuation time to approximately 10 hours. Thus, while no explicit consideration has been given to disruptions caused by earthquakes, the

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parties rely on conflicting expert opinions. The Commission has considered in CLI-84-13, 20 NRC 267 (1984), whether this new information warrants a stay and for the reasons stated there, has concluded that the new information does not require a revision of the seismic design basis of Diablo Canyon at this time. The Commission believes that the license condition requiring PG&E to complete a seismic evaluation of the site by 1988, as new scientific data become available, is the appropriate method for considering such new information.

[4] The one exception is Trojan, for which consideration has been given to the effects of volcanic eruption due to the expectation that another explosion is imminent at Mt. St. Helens.

emergency plans do have considerable flexibility to handle the disruptions caused by various natural phenomena which occur with far greater frequency than do damaging earthquakes, and this implicitly includes some flexibility to handle disruptions by earthquakes as well.

II.

The Commission's second question was whether, even though the regulations do not require it, there are special circumstances for the purposes of 10 C.F.R. § 2.758 that would permit consideration of the effects of earthquakes on emergency planning for Diablo Canyon.

A. Parties' Views

Joint Intervenors argue that this case does present special circumstances. They rely on the proximity of the plant to the Hosgri Fault, the seismic redesign of the plant to accommodate earthquake-induced ground motion which may result from an SSE on that fault, and the conclusion by the Advisory Committee on Reactor Safeguards (ACRS) that the plant is designed to less conservative criteria than would have been applied to a new plant at that site.

The NRC staff and PG&E respond that Diablo Canyon has been redesigned to take into account its proximity to the Hosgri Fault, and, thus, is no different from any reactor which has been designed to accommodate its seismic environment.

B. Analysis

The Commission notes that the important safety issue for any plant located in a region potentially affected by seismic activity is not the location of the facility *per se* but the probable consequences of such location for the plant in question. The Commission will not license a plant unless it can make the statutorily required finding that operation of the plant will not result in undue risk to public health and safety. Necessarily, this includes a determination that the seismic design is adequate. Such a finding is not undermined by the circumstances that more conservative criteria might have been applied to a new plant. The issue is whether operation of the plant as designed will result in undue risk to public health and safety. The Commission's seismic design criteria have been fully addressed for Diablo Canyon and the Commission has determined that the seismic design of the plant presents no undue risk. ALAB-644, 13 NRC 903 (1981).

What remains is the argument that the likelihood of the simultaneous occurrence of an earthquake and a radiologic release from other causes is especially high for this site. The Commission must disagree. The resources, time, and attention devoted to seismic design in this case have been unprecedented, and the information before us does not support the conclusion that the chance of such a simultaneous occurrence is substantially greater than for numerous other nuclear plant sites.

In particular, the Commission takes note of its Appeal Board decision, ALAB-644, *supra*, which concluded that the record does not bear out the claim that the Diablo Canyon site is one of "high seismicity," i.e., an area having a high frequency of seismic events. This conclusion was based on record evidence by Drs. Anderson and Trifunac who plotted for the years 1950 through 1974 the known epicenters in the central California coastal region, centered around Diablo Canyon, between 33° and 37° north latitude and 119° to 123° west longitude. That plot, and the calculated low-recurrence rate of an earthquake of the magnitude assigned the operating basis earthquake (OBE), indicate that the region is at most one of moderate seismicity. Earthquakes of greater magnitude than the SSE would occur with much lower frequency than the OBE. Thus, there has been no showing by Joint Intervenors of special circumstances warranting waiver of the regulations to allow specific consideration of the effects of earthquakes on emergency planning at Diablo Canyon.

III.

The Commission finds that the information and argument presented by the parties in response to the questions posed in CLI-84-4, *supra*, lead to the conclusion that there is no present need to reconsider the *San Onofre* decision.[5]

Nevertheless, we believe that further generic rulemaking exploring the effects of earthquakes on emergency planning could be useful. In particular, the Commission believes that it will be useful to address whether the potential for seismic impacts on emergency planning is a significant enough concern for large portions of the nation to warrant the amendment of the regulations to specifically consider those impacts. The chief focus of the rulemaking proceeding will be to obtain additional information to determine whether, in spite of current indications to the

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[5] In view of the answers to the first two questions, the third question regarding the specifics of any further consideration of the effects of earthquakes on emergency planning need not be addressed.

contrary, cost-effective reductions in overall risk may be obtained by the explicit consideration of severe earthquakes in emergency response planning. In addition, rulemaking would allow a greater spectrum of public participation in the resolution of this matter on a general, as opposed to plant-specific, basis.

We previously indicated in *San Onofre* that this matter would be considered on a generic basis. Some time ago the NRC staff advised us that, in its view, generic consideration was not necessary. However, we were diverted from this issue by the press of other important Commission business, and we took no action in response to that advice. In retrospect, since we disagree with the NRC staff's view, we should have acted sooner and initiated rulemaking. The need to address this issue in this case has again focused our attention on this matter. By this order we are indicating our desire to initiate rulemaking shortly, and directing the NRC staff to give priority attention to the matter.

Commissioner Zech participated only in the portion of the order which concerns the initiation of a rulemaking proceeding.

The additional views of Chairman Palladino and Commissioner Bernthal and the dissenting views of Commissioner Asselstine are attached.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 10th day August 1984.

ADDITIONAL VIEWS OF CHAIRMAN PALLADINO

I agree with the Commission's opinion. I believe that the Commission has adopted a reasonable approach to the question of earthquakes and emergency planning, one which will produce an informed Commission consideration of the policy issue, will not prejudice procedural rights, and will not pose undue risk for the health and safety of the public in the vicinity of Diablo Canyon, as well as other potentially affected plants.

Although the question before the Commission in this case might be characterized as a question of interpretation of NRC emergency planning regulations, I view the issue as a policy question that has generic dimensions. NRC regulations simply do not address earthquakes and emergency planning. Further, at least two other plants in California (San Onofre and Rancho Seco) could be affected by the answer to the outcome of our consideration and other plants outside of California might be affected.

NRC can address a policy question by either adjudication or rulemaking. In this instance, rulemaking offers the opportunity for broader and deeper public input. I believe that the Commission could benefit from public comment on issues such as the following: what is the range of probabilities of a coincidental earthquake and radiological emergency and how does this range compare with that for other natural phenomena that could affect emergency response? To what extent does emergency planning under current NRC regulations provide a sufficient planning base to handle the complicating effects of earthquakes? What benefits of significance for emergency preparedness would be expected to result from the consideration of the complicating effects of earthquakes? Further, if the outcome of the rulemaking is that more should be done, then the new requirements can be applied to Diablo Canyon.

It appears to me that the essential arguments in the dissenting opinion are pertinent to the policy question we will address by rulemaking, and have application to *all* California plants (and possibly to plants elsewhere) and not just Diablo Canyon. The assertions (and counterassertions) of facts and their significance for the policy question can also be examined in the rulemaking and, thus, need not be accepted or argued solely on the basis of the assertions alone. *All* Commissioners have approved this rulemaking and I, for one, have not "already decided the issue."

Rulemaking does not assure Joint Intervenors in this case an opportunity for a formal adjudicatory hearing, but it does provide them an adequate opportunity to be heard. Further, the Joint Intervenors had no assurance of a formal hearing in the Diablo Canyon operating license proceeding. Their hearing rights depended upon their raising an issue that was cognizable in an NRC hearing. The Commission ruled in *San Onofre* (CLI-81-33, 14 NRC 1091 (1981)) that the matter of complicating effects of earthquakes on emergency planning could not be raised in individual cases, and it reaffirmed the *San Onofre* ruling in this case after providing *all* parties, including the Joint Intervenors, with an opportunity to submit written briefs.

While the delay on the Commission's part in addressing the generic policy question is regrettable, it would be speculative to conclude that the delay prejudiced the rights of the Joint Intervenors in the Diablo Canyon proceeding. The outcome of a more timely generic proceeding might have been a final rule that the complicating effects of earthquakes need *not* be considered.

Operation of the Diablo Canyon plant during the interim while the Commission conducts rulemaking does not, in my judgment, pose a significant risk to the public. The probability of an earthquake that would impede emergency response action is exceedingly small for that period of time.

COMMISSIONER BERNTHAL'S ADDITIONAL VIEWS
(Revised August 13, 1984)

The Commission has been remiss in not dealing with this issue earlier, as it had indicated 3 years ago it would. Be that as it may, the question today is how best to proceed, in a manner that assures adequate protection of public health and safety, and is equitable and fair to the parties concerned.

My support of the Commission's order rests on a massive record compiled by the Licensing and Appeal Boards. That record includes the technical judgment of the best seismologists in this country. Their judgment is that the seismic design basis of this facility is adequate to prevent a radiological release from the most severe earthquake that could reasonably be postulated in the vicinity of Diablo Canyon. The complex basis for this conclusion is entirely consistent with the simple, factual, 200-year recorded history of seismic activity in the vicinity of the plant.

As for the probability of a random simultaneous occurrence of (1) an earthquake which could disrupt emergency planning, and (2) an accident severe enough to result in a radiological release *from other causes*, the comments of the parties in response to CLI-84-4 provided no basis for the notion that such an eventuality ought to be taken into account in emergency planning either generically or for Diablo Canyon specifically. My judgment in this regard is supported by the 200-year record of seismic events in the Diablo Canyon area which indicates that there have been only two events in all of that time which had the potential for *any*, let alone major, disruption of emergency response activities.[1]

[1] "Earthquake History of the United States," Publication 41-1, 1982 Reprint with Supplement.

Common perceptions and "gut" feelings might seem to argue that, because a plant is located in California, it must be unique. But the numbers for actual California sites, and for the seismic design bases required of all plants to deal with their particular seismic environments, require us to move beyond subjectivity and to consider the facts. The Appeal Board's conclusion, based on a careful examination of the record, that this particular EPZ area is of "low-to-moderate seismicity," was not casually derived, and is consistent with the history of recorded seismic activity in this limited geographical area.

It clearly makes sense to consider, in emergency response planning, hurricane-type events and fog conditions in California or blizzards in the northern half of the United States, since these events occur on at least an annual basis and have widespread and certain effects on road systems and other facilities which must be utilized should an emergency occur at a nuclear facility. But the actual record of seismic activity in the vicinity of Diablo Canyon, at least, convinces me that earthquakes need not be similarly treated in this case. Nor do I find, from all of the information before me at the present time, any basis to reconsider the *San Onofre* decision.

The hazards of earthquakes, tornados, hurricanes, and fogs rarely choose to conform themselves to State boundaries. California has no monopoly on seismic activity. Three of the four most severe earthquakes ever recorded in the continental United States occurred in the eastern half of the country. Further, there may be reasoned arguments which are possible, but which have not been made by the parties to the *Diablo Canyon* proceeding, to support the specific consideration of seismic effects on emergency planning in the areas surrounding nuclear facilities. Therefore, out of an abundance of caution, I have agreed that the Commission should get on with the generic proceeding it committed to initiate in the *San Onofre* decision so that this issue may finally be laid to rest.

DISSENTING VIEWS OF COMMISSIONER ASSELSTINE

The Commission's performance in its handling of this issue — the complicating effects of earthquakes on emergency planning — is most disappointing. In its apparent determination to avoid adjudicating an issue that the agency itself has acknowledged to be material to emergency planning, the Commission has repeatedly changed its mind about how to treat this issue only to end up right back where it started 3 years ago — promising a generic rulemaking. In the meantime, the Commission's only accomplishment has been to deny parties the right to adjudicate the issue and to delay any action on this issue until the only two plants, Diablo Canyon and San Onofre, for which this issue probably has any real significance have been licensed.

I cannot agree with the Commission's decision or its reasons for reaching that decision. The Commission's decision ignores fundamental principles of emergency planning, offends common sense, and abuses the legal process. I would recognize the obvious — that earthquakes ought to be considered for plants located in areas of high seismicity such as California, and let the parties adjudicate the specifics in individual cases. I would provide the parties to the Diablo Canyon proceeding an opportunity for a hearing and let them litigate whether the Diablo Canyon emergency plan is flexible enough to deal with the complicating effects of earthquakes on emergency planning.

History

The history of the Commission's handling of this issue shows exactly why the Commission's decision today is so disturbing. Rather than simply allowing the issue to be considered by a licensing board, a step that probably would have added about a week of hearing time to the *San Onofre* and *Diablo Canyon* proceedings, the Commission has instead followed a tortuous path from adjudication to generic rulemaking to case-by-case consideration, to generic adjudication, only to end up right back at generic rulemaking.

In early 1981 the staff took the position in the *San Onofre* proceeding that consideration of the complicating effects of earthquakes up to the Safe Shutdown Earthquake (SSE) was appropriate. The staff disagreed, however, when the Licensing Board tried to raise *sua sponte* the issue of the effects of earthquakes exceeding the SSE. The Commission on its own motion ordered the Licensing Board not to consider "the impacts on emergency planning of earthquakes which cause or occur during an accidental radiological release." *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-81-33, 14 NRC 1091, 1091 (1982). The Commission determined that its regulations did not require such consideration and concluded that whether the regulations should require such consideration was a generic issue to be decided by rulemaking. *Id.* at 1091-92.

Based on the *San Onofre* decision, the Licensing Board in the Diablo Canyon operating license proceeding refused to allow any consideration of the effects of earthquakes on emergency planning at the Diablo

Canyon site. There was, therefore, no opportunity to litigate any issue connected with the complicating effects of earthquakes on emergency planning.

After the Diablo Canyon Board's decision, the staff on June 22, 1982, issued a memorandum which stated that it was the staff's technical judgment that a generic rulemaking was not necessary because of the very low likelihood of earthquakes in most parts of the country. However, the staff took the view that for California and other areas of high seismic risk in the Western United States explicit, site-specific consideration of the effects of earthquakes on emergency planning is necessary. As the staff explained:

> It is the judgment of the staff that for most sites earthquakes need not be explicitly considered for emergency planning purposes because of the very low likelihood that an earthquake severe enough to disturb onsite or offsite planned responses will occur concurrently with or cause a reactor accident. Planning for earthquakes which might have implications for response actions or initiate occurrences of the "Unusual Event" or "Alert" classes in areas where the seismic risk of earthquakes to offsite structures is relatively high may be appropriate (e.g., for California sites and other areas of relatively high seismic hazard in the western U.S.).

Memorandum to the Commissioners from William Dircks, Executive Director for Operations, dated June 22, 1982, entitled "Emergency Planning and Natural Hazards," at 1. The staff went on to say that it requests applicants for licenses for California facilities and the Federal Emergency Management Agency (FEMA) to consider earthquake effects in their emergency planning and review. Memorandum of June 22, 1982, Enclosure at 3-4. In fact, at both San Onofre and Diablo Canyon the staff required the license applicants to specifically consider this issue.

The Commission realized that this position by the staff seemed to contradict the Commission's *San Onofre* decision and thus cast doubt on the validity of the Licensing Board's ruling in the Diablo Canyon case. The Commission asked the staff to elaborate and in a further memorandum, the staff repeated its conclusion that "planning for earthquakes which might have emergency preparedness implications may be warranted in areas where the seismic risk to offsite structures is relatively high (e.g., California sites . . .)." Memorandum to Chairman Palladino from William Dircks, Executive Director for Operations, dated January 13, 1984, entitled "Emergency Planning and Seismic Hazards," at 2 n.2. The staff also stated that it thought current emergency planning review criteria were adequate for this. *Id.*

Given this position by the staff, the Commission decided to ask the parties to the Diablo Canyon proceeding whether and under what circumstances the effects of earthquakes on emergency planning should be considered for the Diablo Canyon plant. *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-84-4, 19 NRC 937 (1984). The Commission, referring to the staff's January 1984 memorandum, noted that the staff appeared "to believe that some specific consideration of the effects of seismic events on emergency planning may be warranted for plants located in areas of relatively high seismicity." CLI-84-4, *supra,* 19 NRC at 938.

In its response to the Commission's order, the staff attempted to reverse course. Staff counsel explained that while staff stated in its January 13, 1984 memorandum that "seismic events are considered and evaluated to a limited extent as part of our current emergency planning reviews, those staff reviews are informal and do not reflect a required licensing element which must be satisfied in order to warrant issuance of a license."[1] "NRC Staff's Memorandum Regarding Consideration of Effects of Earthquakes on Emergency Planning (CLI-84-4)," dated May 3, 1984, at 3 n.2.

Commission Decision

In its decision today, the Commission has concluded that there is no reason to depart from its decision in *San Onofre* that the NRC's regulations "do not require consideration of the impacts on emergency planning of earthquakes which cause or occur during an accidental release," for Diablo Canyon and that the determination of whether to amend the regulations to include the consideration of earthquakes should be addressed as a generic matter. *See* p. 250, *supra.* There are several problems with the Commission's decision and its underlying rationale.

The cornerstone of the Commission's decision is the Commission's conclusion that the probability of an earthquake disrupting an emergency response is so low that it need not be considered in emergency planning. The basis for the Commission's conclusion is its determination that for various reasons there is unlikely to be a radiological release and an earthquake at the same time. The Commission's arguments on this score ignore one of the fundamental precepts of emergency planning: we plan for low-probability occurrences because no matter how safe we try to make nuclear power plants there is always a possibility that some

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[1] The flaw in the staff's argument is obvious. Having acknowledged that it is concerned enough about the issue to require licensees to consider it, the staff cannot now argue that "informal" review by the staff is a satisfactory substitute for formal review in individual licensing proceedings. If the issue is material to the Commission's licensing decision, as the staff's own statements and actions concede, then the agency must admit that satisfactory resolution of the issue is a required licensing element.

event will occur which will require use of one or more aspects of emergency planning. The probability arguments used by the Commission are really arguments that we do not need *any* emergency planning, rather than that we need not consider earthquakes in emergency planning. The Commission simply asserts that there is a low likelihood of a release and an earthquake at the same time and assumes that that ends the inquiry.

Unfortunately, the Commission ignores the fact that safety calculations are subject to some uncertainties. The philosophy behind emergency planning is to recognize this uncertainty and to provide defense in depth in protecting the public. Indeed, the Commission's emergency planning regulations are founded on the judgment that adequate emergency planning is an essential element in protecting the public health and safety independent of the Commission's other regulations and safety reviews focusing on the design of the plant itself. Obviously, we do not plan for every conceivable but highly unlikely event. We should not, for example, waste resources planning for the effects of hurricanes on emergency responses in Kansas or for snow in Southern California. Instead, we plan to take into account the natural phenomena which present the more likely risks for a particular area. Thus, we consider hurricanes for plants in Florida, tornados for plants in the Midwest, and volcanic eruptions in the Pacific Northwest. By the same token, we should consider the complicating effects of earthquakes for plants in high-seismic-risk areas such as California.

The Commission tells us, however, that the probability of an earthquake disrupting an emergency response in an Emergency Planning Zone (EPZ) is too low even to be considered. To apply this argument to California, where almost 90% of the seismic activity in the United States occurs and where earthquakes which damage, obstruct or disrupt roads, buildings, bridges and communications networks occur with some regularity, simply ignores common sense. In support of this assertion, the Commission contends that the Diablo Canyon site is located in an area of low-to-moderate seismicity. This argument is based upon an analysis in the record of the recurrence rate for earthquakes in the central California coastal region for the years 1950 through 1974. What the Commission does not mention, however, is that the only plant in the country with a comparable SSE and OBE (Operating Basis Earthquake) — the key bases for the seismic design of the plant — is San Onofre (0.67*g* and 0.34*g*, respectively). In fact, the SSEs and OBEs for plants in other parts of the country are significantly lower (for other plants the SSE is typically 0.25*g* or less and the typical OBE is 0.11-0.12*g*, with the highest being 0.13*g*) than those for Diablo Canyon (SSE of 0.75*g* and

OBE of 0.20*g*). Clearly, by requiring the plant to be designed to withstand an earthquake with ground motions almost twice those of other plants in the country, the Commission explicitly made the technical judgment that the earthquake risk for the Diablo Canyon area is not comparable to other areas of the country, and is, in fact, much higher.[2]

Further, the Commission's argument must be considered in light of the other natural phenomena the Commission includes in its consideration of emergency planning. If the probability of an earthquake disrupting an emergency response in an EPZ in California is too unlikely to be considered, that probability must by definition be much lower than the probability of disruption caused by the other natural phenomena which the Commission does consider. It must, for example, be less likely than the probability that a tornado will disrupt an emergency response in an EPZ in the Midwest or that a hurricane will disrupt an emergency response in a California EPZ.

The probability that a tornado will travel through a particular 10-mile area and thereby initiate or disrupt response to an emergency at a nuclear plant must be quite low; yet, the Commission requires consideration of that issue for certain plants. Similarly, the probability of a hurricane striking the San Luis Obispo coastal area and initiating or disrupting an emergency response must also be quite low; yet, the Commission considered that very issue in the Diablo Canyon case. I see no factual basis for the Commission's assertion that earthquakes in California are so much more unlikely than either of these events that earthquakes need not be considered.

The Commission's order also misses another very important point. Emergency planning is not relevant only to accidents resulting in the offsite release of radiation. Emergency planning is also relevant for responses to emergencies which do not result in a radiological release, including emergencies initiated or complicated by earthquakes below the SSE. For example, whether or not an earthquake results in the offsite

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[2] Publicly available information compiled by the U.S. Geological Survey (USGS) would seem to indicate that earthquakes of sufficient magnitude to cause possible damage, obstruction or disruption to roads, buildings, bridges and communication networks occur throughout many parts of California, including the San Luis Obispo area, with some regularity. "Earthquake History of the United States," Publication 41-1, 1982 Reprint with Supplement. According to this information, four earthquakes have occurred in the immediate San Luis Obispo area since 1830, and at least one of these earthquakes has been of magnitude 7-8 on the Modified Mercalli scale. *Id.* at 138, 140, 141, 156, 162, 164. In addition, two other earthquakes, of magnitudes 6.5 and 7.5, have occurred within 50 miles of the Diablo Canyon site since 1922. "Earthquake Epicenter Map of California, 1900 through 1974," State of California, the Resources Agency, Department of Conservation 1978. This publicly available information, although not in the record of the *Diablo Canyon* proceeding, would also appear to contradict the Commission's assertions regarding the frequency of occurrence of earthquakes in the vicinity of the Diablo Canyon site which are sufficiently severe to cause damage to structures and disrupt communications. Much of this same information is also in the FSAR for Diablo Canyon, which is a part of the record in this proceeding.

release of radioactivity, an emergency plan must take into account the assurance of continued communication between a plant and offsite emergency response agencies, the ability to obtain damage estimates for the plant and the offsite transportation and communication facilities to provide data for decisions on appropriate responses, the availability of backup facilities to ensure continued functioning of an emergency response capability, and the ability to transport necessary personnel to a plant to deal with the emergency. In its June 22, 1982 memorandum to the Commission, the NRC staff recognized this:

> There is no explicit guidance in [the Commission's regulations] as to the extent to which adverse earthquake conditions are to be taken into account in emergency planning at particular sites. . . . The occurrence of earthquakes of a nature that could have implications for onsite or offsite response actions or initiate occurrences of the "Unusual Event" or "Alert" class is an adverse characteristic of the type discussed above.

Memorandum at 3-4. The staff went on to note that it asks applicants for licenses for California facilities and FEMA to consider such earthquakes (smaller than the Safe Shutdown Earthquake) in their emergency planning for this very reason.

The Commission simply ignores the fact that the staff has been requiring licensees for plants located in California to consider the effects of earthquakes on emergency planning. The staff has stated that while it does not think such consideration is necessary for plants in most areas of the country, "planning for earthquakes which might have emergency preparedness implications may be warranted in areas where the seismic risk to offsite structures is relatively high (e.g., California sites)."[3] Memoranda of June 22, 1982, and January 13, 1984. The complicating effects of earthquakes on emergency planning were formally considered by the staff in the *San Onofre* proceeding, and were informally considered by the staff for Diablo Canyon. By their own actions, the agency's technical experts have demonstrated that they consider this issue to be material to the Commission's licensing decisions in these two cases. Given the fact that the staff experts on this issue have been concerned

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[3] In its response to the Commission's order, staff counsel attempted to withdraw this conclusion. The fact remains, however, that staff has indeed been considering the complicating effects of earthquakes on emergency planning at California plants, including Diablo Canyon. Staff required PG&E to prepare a report on this issue. Presumably, the staff does not ask license applicants to look at issues which it thinks are irrelevant. Perhaps the staff's new position has something to do with the fact that for the only two plants located in "high seismic areas," the staff has now completed its review of seismic effects on emergency planning. This appears to be the only plausible reason for such a radical change in staff's position. Further, staff explained that what it really wanted was to consider this issue, but only "informally." *See* 261, *supra*.

enough to consider it, I see no basis for the Commission's argument that in the cases of Diablo Canyon and San Onofre, seismic effects on emergency planning are irrelevant. Since the issue is clearly material to the agency's licensing decision in those two cases, the Commission is required by law to grant the parties an opportunity to litigate that issue. *See Union of Concerned Scientists v. NRC*, 735 F.2d 1437 (D.C. Cir. 1984.)

Apparently recognizing the weaknesses in their low probability argument, my colleagues have also attempted to support their decision by arguing that the disruptions to emergency response caused by fog, hurricanes and heavy weather are similar to the disruptions which may result from an earthquake. Thus, the Commission argues, emergency plans *implicitly* have enough flexibility to deal with earthquakes as well. This is an interesting argument. Unfortunately, the Commission cannot point to any evidence in the record of this proceeding to support such a factual finding. Although the Diablo Canyon record includes information on natural phenomena other than earthquakes, there was no discussion in that record of earthquake effects, or whether the plans for dealing with other natural phenomena are flexible enough to implicitly include the effects of earthquakes. The Commission's conclusion seems, therefore, to be based on the Commission's intuitive feeling that the finding ought to be true rather than on any kind of factual record. This is precisely the type of factual question that should only be decided based upon a site-specific, factual record, developed and tested in a hearing (or at least after consideration of information in the record of a rulemaking specifically addressing this issue).

Finally, the Commission has decided that the regulations are not sufficiently clear on whether earthquakes must be considered in emergency planning and so intends to conduct a generic rulemaking on the issue. The Commission disagrees with the staff's view that a generic rulemaking is not necessary, although it offers no persuasive reason for rejecting the staff's technical judgment on this question. Unfortunately, the Commission's belatedly renewed promise of a generic rulemaking appears to be little more than window dressing. The Commission's justification for not considering seismic effects on emergency planning at Diablo Canyon clearly shows that it has already decided the issue. If the Commission will not require the consideration of earthquakes for plants located in an area of the country where 90% of the seismic activity occurs, it is unlikely to conclude that they must be considered for plants elsewhere. Since the Commission appears to have already decided this fundamental issue, it is unclear what it hopes to accomplish with such a rulemaking. I have agreed to the Commission's decision to conduct such a rulemaking, but only because some consideration of this issue is better than no consideration at all.

It is absolutely amazing, the lengths to which the Commission will go to avoid finding that a party is entitled to a hearing on an issue. In this case, the Commission has constructed an elaborate, but flawed, rationale in an attempt to explain why earthquakes need not be considered in emergency planning for Diablo Canyon. The Commission has then proceeded, as a factual matter, to consider the effects of earthquakes on emergency planning. As a last resort, the Commission has again promised to conduct a generic rulemaking on this issue, a promise that it made 3 years ago but did not keep. The unfortunate consequence of this delay has been to put the issue off until the two California plants have been licensed.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

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) **August 10, 1984**

The Commission determines to make effective, without prejudice to the pending appeals and petitions for review of the various licensing and appeal board decisions in this proceeding, the Licensing Board's fourth and final Partial Initial Decision authorizing the issuance of a full-power license for the Diablo Canyon Nuclear Power Plant Unit 1, LBP-82-70, 16 NRC 756 (1982), and, further, concludes that the license conditions imposed by the Board have been fulfilled and all other matters resolved so that the license may be issued.

MEMORANDUM AND ORDER

INTRODUCTION

This order concludes the Nuclear Regulatory Commission's process for determining whether to make effective the Atomic Safety and

Licensing Board's ("Licensing Board") fourth and final Partial Initial Decision (PID), LBP-82-70, 16 NRC 756 (1982) authorizing the issuance of a full-power license for the Diablo Canyon Nuclear Power Plant, Unit 1 ("Diablo Canyon" or "plant"), to Pacific Gas and Electric Company ("PG&E"), subject to the satisfaction of certain license conditions. Formal appeals and petitions for Commission review of the merits of various Licensing Board and Atomic Safety and Licensing Appeal Board ("Appeal Board") decisions for Diablo Canyon are still pending. This effectiveness decision is without prejudice to those appeals and petitions. 10 C.F.R. § 2.764.

In addition to reviewing the Licensing Board's decision and determining the status of the license conditions imposed in it, the Commission has considered several other issues, some of which arose as a result of the unique circumstances associated with this plant. The other matters considered by the Commission are: licensing issues which were not placed in controversy in the formal licensing hearings, including review of the concerns of Mr. Isa Yin regarding small-bore piping and pipe supports (Mr. Yin is an NRC inspector who was assigned to review some of the allegations regarding Diablo Canyon); issues related to the Independent Design Verification Program (IDVP) and determined by the NRC staff to require resolution prior to full-power operation; NRC staff evaluation of training and qualification of operators and shift supervisors; pending petitions for enforcement action pursuant to 10 C.F.R. § 2.206; allegations determined to require resolution prior to full-power operation; investigations by the Office of Investigations (OI) and the Office of Inspector and Auditor (OIA); recent Appeal Board decisions on motions to reopen the record, and on design quality assurance (DQA) and construction quality assurance (CQA); consideration of the effects of earthquakes on emergency planning; and Joint Intervenors' request for a stay of this licensing proceeding.

CONCLUSION

The Commission's decision on these issues is discussed below. In sum, the Commission has determined: (1) to make effective, without prejudice to the pending merits reviews, the Licensing Board decision authorizing issuance of the full-power operating license for Diablo Canyon; (2) that the license conditions imposed by the Licensing Board have been fulfilled; and (3) that all of the other matters listed above have been resolved adequately to authorize issuance of the full-power license for Diablo Canyon Unit 1. However, this Order shall not become effective, and no full-power license may issue, until 5:00 p.m., Eastern

Daylight Time, August 17, 1984. This delay is to allow orderly processing of any request for expedited judicial review.

DISCUSSION

I. Licensing Board Decision

In LBP-82-70, 16 NRC 756 (1982), the Licensing Board determined that a full-power operating license for Diablo Canyon could be issued upon the satisfaction of certain license conditions. Previous decisions by the Licensing Board and Appeal Board resolved other contested matters. The two remaining issues decided by the Licensing Board in LBP-82-70 were:

(1) the adequacy of the Diablo Canyon emergency plan; and

(2) whether the plant's pressurizer heaters, block valves and power-operated relief valves were required to be classified as safety-grade and provide adequate protection to the public health and safety as installed.

The Licensing Board found that PG&E's emergency plan would satisfy Commission regulations and be adequate upon completion of the following license conditions by the Director of Nuclear Reactor Regulation:

a. verification that deficiencies identified by FEMA in the San Luis Obispo County emergency plan have been corrected;

b. receipt of written acquiescence by the appropriate State jurisdictions binding them to participate in the Standard Operating Procedures required to be followed by Federal Regulations;

c. receipt of FEMA findings on the adequacy of the State Emergency Plan; and

d. verification that tone alerts or equivalent warning devices are operational in schools, hospitals and other institutions.

On August 2, 1984, the Director informed the Commission that all these license conditions were satisfied.[1]

As for the pressurizer heaters, power-operated relief valves and their associated block valves, the Licensing Board found that: (1) pressurizer heaters were not required to be safety-grade; (2) two of the three

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[1] In ALAB-776, 19 NRC 1373 (1984), the Appeal Board vacated the license condition requiring the Director to obtain FEMA findings on the adequacy of the State emergency plan, insofar as that license condition may have been interpreted to require completion of the formal FEMA review process under 44 C.F.R. § 350. To the extent that the Licensing Board may have had a less formal FEMA review in mind, the Board's condition has been satisfied by FEMA's letter of July 11, 1984. The merits review of ALAB-776 is pending before the Commission, and the Commission does not, at this point, express any view on the correctness of ALAB-776.

PORVs and associated equipment are safety-grade; and (3) adequate protection of public health and safety is provided by this equipment as installed. These decisions obviously support the issuance of a full-power license. The Commission finds nothing in the pending appeal which would support a stay of license issuance.

2. *Uncontested Licensing Issues*

a. *Conditions on the Low-Power License*

The low-power license for Diablo Canyon contained several license conditions required to be satisfied by PG&E prior to a full-power license decision. Seven of these conditions were a direct outgrowth of concerns raised by Mr. Yin. In response to his concerns, the NRC staff formed the Diablo Canyon Peer Review Group (Peer Review Group), which included senior staff engineers expert in piping, piping supports, and quality assurance. After meeting with Mr. Yin and PG&E, and after examining areas of the plant of concern to Mr. Yin, the Peer Review Group formulated the seven license conditions.

The license conditions addressed the following issues:

1. review of all computer calculations of small-bore piping supports;
2. review of rigid supports placed in close proximity to each other to assure that load sharing results in acceptable piping and support stress;
3. review of snubbers in close proximity to rigid supports to ensure adequate snubber function;
4. development of a periodic inspection program to ensure the maintenance of thermal gaps included in thermal analysis of piping;
5. establish procedures and schedules for the hot walkdown of the main steam piping system and document the results of such walkdown;
6. review, resolve and document certain piping design changes; and
7. demonstrate by report to the Commission that certain technical issues in the design of supports for small-bore and large-bore piping have been addressed.

After a thorough review, the Peer Review Group and the Advisory Committee on Reactor Safeguards ("ACRS") found that PG&E had analyzed and resolved the issues in the license conditions adequately to permit full-power operation. These conclusions are set forth in staff's

Safety Evaluation Report Supplement ("SSER") 25. SSER 25 is discussed further under § 3, below.

At the August 2, 1984 public Commission meeting, Mr. Yin expressed his professional disagreement with the Peer Review Group's report on the adequacy of the resolution of certain design issues. The Commission explored with Mr. Yin and other members of the NRC staff the details of this differing professional judgment. Based on these discussions and the analyses in SSER 25, the Commission believes that the collective judgments by the Peer Review Group and ACRS are deserving of more weight than the views of Mr. Yin. Accordingly, the Commission accepts the judgments of the Peer Review Group and ACRS and believes that these matters have been resolved adequately for issuance of a full-power license.

Staff concluded in SSER-23 that PG&E had satisfied its requirements related to fire protection. Staff also reported in SSER 24 that PG&E's jet impingement evaluation, conducted in response to a condition imposed by the Appeal Board in ALAB-763, 19 NRC 571 (1984), was acceptable.

b. Other Issues

As with any full-power license, the license for Diablo Canyon contains several technical conditions which reflect the NRC staff's prelicensing technical review of issues relevant to full-power operation. For Diablo Canyon, the license conditions and the technical bases for them are contained in SSER 27. The Commission believes that SSER 27 adequately addresses the full-power issues considered by the staff.

3. Independent Design Verification Program

a. Large- and Small-Bore Piping

In SSERs 18, 19 and 20 the staff identified issues regarding the IDVP's review of the design of small- and large-bore piping and stated that those issues should be resolved prior to full-power operation. Those issues arose out of inspections performed in response to allegations concerning the control of design of pipes and piping supports. The principal issues identified by the staff were: (1) adequacy of the size of the sample used to determine the acceptability of small-bore piping designed in accordance with "span-rules"; (2) apparent inconsistencies between alleged deficiencies in Interim Technical Reports and the decision not to expand the IDVP; and (3) adequacy of the sample size and distribution used to analyze large-bore piping and its supports.

The NRC staff's procedure for resolving these issues is described in SSER 25. The Peer Review Group determined that piping designed using span-rules was acceptable, that well-founded judgmental factors had been applied to select the size and distribution of samples for review, that the number and types of samples were adequate to verify design methodology, that apparent deficiencies in the ITRs were found insignificant to the IDVP when viewed in light of the backup material, and that review of all small-bore, computer-analyzed supports showed that input errors had no impact on satisfying the licensing criteria. Accordingly, the Peer Group reaffirmed the IDVP's conclusion that the design of large- and small-bore piping had been verified. The Commission finds that the issues regarding the IDVP's review of large- and small-bore piping have been adequately resolved to permit full-power operation.

b. Other Issues

Supplements 18, 19 and 20 to the Safety Evaluation Report for Diablo Canyon also identified a number of other items requiring resolution prior to full-power operation. In Supplement 24 to the Safety Evaluation Report (SSER 24) the staff has reported that all these items have been resolved. The Commission has no reason to disagree with the staff's analysis.

4. Training and Qualification of Operators and Shift Supervisors

On July 13, 1984, the NRC staff reported to the Commission on the performance of operating crews and shift advisors[2] during startup and low-power testing. SECY-84-283 (1984). The report was based on observations and evaluations by various teams composed of members of the NRC staff expert in operator licensing, license qualification, and license examination. The teams concluded that:

1. PG&E has provided shift advisors that meet the Commission requirements for qualifications, training and experience;
2. the shift advisors are successfully working with operating shift crews;
3. operator crew performance during startup and low-power testing has been above average; and

[2] Shift advisors experienced with PWRs comparable to Diablo Canyon were provided for each operating shift to provide operating support until the operating crews attained experience with operating the facility.

4. licensee management is adequately involved in day-to-day operations.

On the basis of this report, the Commission concludes that the operating staff is capable of operating Diablo Canyon at full power.

5. Petitions Under 10 C.F.R. § 2.206

In recent months several petitions for enforcement action related to Diablo Canyon were filed. Essentially, these petitions were based on various allegations regarding construction practices and plant safety. These allegations are discussed below. At the August 2, 1984 public Commission meeting, the staff reported that it found nothing in the petitions that would warrant deferring the authorization of full-power operation.

6. Allegations Relevant to Full Power

As of July 8, 1984, there were over 1400 allegations regarding Diablo Canyon, although many (some 400) were duplications or small variations of others. All these allegations were filed since early 1983, some 10 years after PG&E filed its operating license application. In SSER 26, the staff reported that it considered 581 allegations formally resolved, and that in its view none of the other allegations required formal resolution prior to full-power operation.

All allegations were handled by the Diablo Canyon Allegation Management Program (DCAMP) described in SSER 21 and SSER 22. Under that program, the NRC staff has spent thousands of hours investigating and evaluating those allegations. All allegations were screened using criteria set out in SSER 22 for determining which allegations required resolution prior to full-power operation.

As a result of this screening, seven areas were identified in SSER 22 as requiring resolution prior to exceeding low power:

1. Operational Limits for the Component Cooling Water System;
2. Replacement of Welded High-Strength Bolts;
3. As-Built Drawings for Operations;
4. Completion of Systems Interaction Program and Modifications;
5. Evaluation of Coating Concern;
6. Piping and Supports and Related Design Issues; and
7. Residual Heat Removal Low Flow Alarm.

The detailed evaluations and resolutions of these allegation areas are contained in SSER 26. In addition, SSER 26 resolves a subsequently developed allegation area regarding bolted connections.

At the August 2, 1984, public Commission meeting, the staff reported that approximately 300 of the remaining allegations had been resolved satisfactorily and that the documentation of these resolutions would be available shortly. The staff also reported that resolution of all of the allegations required only very few (less than ten) physical changes to the plant. Some 500 allegations remain which have not been formally resolved. However, each of these has been reviewed under the SSER 22 screening criteria, and it has been determined that full-power operation can be authorized pending formal resolution.

Allegations of harassment or intimidation received special Commission attention. Relatively few (eight) individuals have made such charges, and staff concluded, based on its reviews, which included interviews of approximately 250 individuals on site and hundreds of interactions with others in the course of reviews of allegations, that there was no widespread pattern of harassment or intimidation sufficient to call the quality of the plant into question.

Based on our review of the information contained in SSER 26 and the information described above, as well as the other information provided at the August 2 meeting, the Commission believes that a full-power license need not be deferred pending the formal resolution of the outstanding allegations. Efforts to resolve all remaining allegations formally will continue.

7. *Investigations*

The Office of Investigations is still pursuing a number of allegations of wrongdoing related to Diablo Canyon, some related to harassment or intimidation of PG&E contractor quality inspectors. Staff informed the Commission at the August 2 meeting that these pending matters need not delay full-power authorization because, based on its screening of the allegations against the criteria of SSER 22, it found no significant technical problem or pervasive pattern of purposeful intimidation. At the same meeting, the Office of Inspector and Auditor (OIA) reported that allegations of wrongdoing by the staff had not been substantiated. The Commission also discussed with Mr. Ronald Smith, the OIA investigator, allegations regarding his conduct of the investigation.

Based on the written and oral report by the staff, the Commission concludes that authorization of the full-power license need not await resolution of pending investigations and that there is no reason to pursue further the allegations of staff wrongdoing.

8. *Adjudicatory Decisions*

In ALAB-756, 18 NRC 1340 (1983), the Appeal Board determined that Joint Intervenors and the Governor of California had failed to carry the heavy burden of showing that the formal adjudicatory record on construction quality assurance should be reopened. Petitions for Commission review of this decision were then filed. A majority of the Commission not having voted to review this decision, the petitions for review were deemed denied.

In ALAB-763, *supra*, the Appeal Board extensively reviewed contentions regarding alleged deficiencies in the design quality assurance program as reviewed by the Independent Design Verification Program (IDVP). The Appeal Board found that the IDVP had not uncovered any uncorrected deficiencies in design quality assurance requiring a reversal of the Licensing Board's previous decision on the adequacy of design quality assurance. The Commission is considering the petitions for review of this decision and the responses thereto. The decision in ALAB-763 obviously supports issuance of a full-power license, and the Commission sees nothing in the petitions for review that would warrant a stay of the full-power license pending further review.

In ALAB-775, 19 NRC 1361 (1984), the Appeal Board denied additional petitions by the Joint Intervenors and Governor of California to reopen the record on design and construction quality assurance. The Commission has not yet determined whether that Appeal Board decision warrants review. ALAB-775 also supports issuance of a full-power license, and the Commission sees no reason to stay the issuance of the full-power license pending further review.

9. *Effects of Earthquakes on Emergency Planning*

In a separate Decision, CLI-84-12, 20 NRC 249 (1984), the Commission concluded that its regulations do not require specific consideration of the effects of earthquakes on emergency planning, and that there are no special circumstances warranting waiver of the regulations to allow such consideration for Diablo Canyon. Rather, this issue would be pursued on a generic basis by rulemaking.

10. *Stay Requests*

a. *New Seismic Information*

By letter dated July 17, 1984, Joint Intervenors requested the Commission to delay indefinitely any vote on whether to authorize a full-power operating license for Diablo Canyon. The bases for Joint Intervenors' request were recent developments regarding the geology of the site at Diablo Canyon and new data associated with recent earthquakes in central California. This information has also been supplied to the Appeal Board in Joint Intervenors' motion to reopen the seismic record in this proceeding.

Subsequently, on July 25, 1984, Joint Intervenors moved the Appeal Board to stay the Diablo Canyon proceeding. That stay request incorporated Joint Intervenors' previous request of July 17, 1984, and raised other issues. By Order of July 27, 1984 (unpublished), the Appeal Board directed that stay request to the Commission.

The Commission has reviewed the parties' filings and determined, for the reasons discussed below, that a stay of the licensing proceeding is not warranted.

Before addressing the stay criteria, the Commission notes that it has recognized the growth of scientific knowledge in seismology and geology and the resulting potential need to reassess the seismic design basis of Diablo Canyon. The license for Diablo Canyon is conditioned on PG&E's completion of a seismic reevaluation by 1988. Of course, if new information developed in the interim requires more prompt action, that action will be taken. But the information presented now by Joint Intervenors does not warrant a stay.

Traditional stay analysis requires a movant to address several factors including, in particular, a demonstration of irreparable injury and probability of success on the merits. As applied to the new seismic information, this requires Joint Intervenors to demonstrate that the new information requires the conclusion that there is no longer reasonable assurance that the seismic design of Diablo Canyon is adequate, and that Joint Intervenors will be irreparably injured by permitting the plant to operate before the plant is abandoned or rebuilt in accordance with some modified design. A review of the information presented by Joint Intervenors shows that it does not meet the stay requirements.

Joint Intervenors rely on new data from the Morgan Hill earthquake of April 24, 1984. This earthquake resulted in the highest horizontal ground acceleration ever recorded, 1.29*g*, at a site on an abutment of the Coyote Dam near the southeast end of the rupture zone. Joint Intervenors contend that measurement of such a high ground acceleration for an earthquake of magnitude 6.1 shows that the anchor acceleration of 0.75*g*, taken as an important element of the seismic design basis for Diablo Canyon, is much too low for the Safe Shutdown Earthquake (SSE) of magnitude 7.5 assigned to the Hosgri Fault.

This conclusion does not necessarily follow from the data. As Joint Intervenors acknowledge, there is evidence in the record that two other earthquakes smaller than the SSE, the San Fernando Valley earthquake of 1971 and the Imperial Valley earthquake of 1979, both resulted in ground accelerations substantially higher than 0.75*g*. An acceleration of 1.25*g* was measured at the Pacoima Dam in 1971 and an acceleration of 0.81*g* was measured at Bond's Corner in 1979. The Appeal Board, in ALAB-644, 13 NRC 903 (1981), found that in both cases these anomalously higher acceleration values were distorted responses related to singularities in site geology. PG&E notes in its response to the stay motion that the acceleration at Pacoima Dam was almost as great as the acceleration measured at Morgan Hill and, thus, that the Appeal Board already took such high values of the acceleration into account when reviewing the seismic design basis of Diablo Canyon.

The Commission finds that the Morgan Hill data do not undermine the Appeal Board's analysis. As PG&E and the NRC staff point out, the new high value of ground acceleration observed at Morgan Hill was measured at a dam abutment, thus presenting a situation similar to that at the Pacoima Dam. Moreover, as discussed below, the "focusing" effect believed partially responsible for this high value of ground acceleration has already been found not to be present at Diablo Canyon. Under these circumstances, the Joint Intervenors have not established that they are likely to demonstrate a lack of reasonable assurance that the seismic design is adequate.

Joint Intervenors also rely on the conclusions of the United States Geologic Survey that the Morgan Hill earthquake demonstrated "focusing" and "high stress drop." These findings, Joint Intervenors contend, contradict the Appeal Board's conclusions that focusing and high stress drop were speculative phenomena.

But the Appeal Board did not merely dismiss focusing and high stress drop as speculative phenomena. For example, focusing was dismissed in part for Diablo Canyon because of site geology. The Appeal Board found that focusing would not be expected because the Diablo Canyon site had the wrong orientation to the Hosgri Fault and was too far from the source of the focussed motion. By contrast, the high ground acceleration associated with the Morgan Hill earthquake was measured at a site aligned with the unilateral rupture expansion and close to a secondary energetic source of seismic radiation. Thus, the Morgan Hill data do not undercut the Appeal Board's discussion of focusing.

As for high stress drop, there too the Appeal Board found that there were no indications of high-stress-drop regions on the Hosgri Fault, not that a high-stress-drop phenomenon does not exist. The Appeal Board's conclusion is based in substantial part on the determination that the Hosgri Fault would exhibit strike-slip/dip-slip motion rather than thrust motion. Joint Intervenors point out that recently published evidence by Crouch and others indicates that the Hosgri Fault may be a thrust fault and may be closer to the plant than previously believed.

The Commission was briefed on the Crouch data at a public meeting on whether to authorize the low-power license for Diablo Canyon. At that meeting, Mr. James Devine of the USGS expressed the opinion that even if the Hosgri Fault were a thrust fault, the seismic design basis for Diablo Canyon was probably adequate. As he put it, the new data were not "stop the presses" information. PG&E notes that at the Licensing Board hearings several experts testified that the Hosgri Fault had a component of reverse faulting and that expert testimony included a diagram showing the fault plane in the position predicted by the new information. PG&E also presents expert opinion that the Hosgri Fault is not substantially closer to the plant than previously believed. The NRC staff notes that the Newmark Spectrum for Diablo Canyon already accounts for the type of motion associated with a thrust rupture at depth which propagates up-dip.

At this point any uncertainty concerning the character of the Hosgri Fault should be resolved through the normal scientific peer review process.[3] Indeed, in a letter of June 20, 1984, the ACRS stated that the new data on the character of the Hosgri Fault do not require "immediate revision of the seismic design basis for Diablo Canyon."

Finally, Joint Intervenors contend that recent earthquake activity in California's central coastal region contradicts the Appeal Board's conclusion that the plant is situated in an area of low-to-moderate seismicity. PG&E has provided contrary expert opinion, and the staff notes that the six earthquakes referred to by Joint Intervenors occurred over a widely scattered area. Under these circumstances, Joint Intervenors have not demonstrated the necessary probability of success on the merits on this point.

b. Other Issues

Joint Intervenors' stay request of July 25, 1984, raises five other issues which have been raised before the Commission in earlier stages

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[3] This would include a reevaluation of the Safe Shutdown Earthquake should the character of the fault be definitively determined to be of the thrust variety. Pending such a reevaluation, there is no basis for the Joint Intervenors' assumption that an SSE of magnitude 7.5 would still be appropriate for a different type of fault motion.

of this proceeding. Because Joint Intervenors present no new perspectives on these issues, the Commission responds to them briefly below.

(i) Class Nine Accidents — Once again Joint Intervenors contend that the Commission violated the National Environmental Policy Act of 1969 and its own regulation by not explicitly considering class nine accidents in the Final Environmental Statement for Diablo Canyon. The Commission has replied to this argument most recently in its brief filed in the U.S. Court of Appeals in the D.C. Circuit in reply to Joint Intervenors' petition for review of the Diablo Canyon low-power license. *San Luis Obispo Mothers for Peace v. NRC* (No. 81-2034 and consolidated cases). Joint Intervenors have added nothing new to their argument that they are likely to prevail on the merits on this issue. The Commission finds that this issue does not warrant a stay of the full-power proceeding.

(ii) Earthquake Emergency Preparedness — As stated above in § 9, the Commission has addressed this issue by a separate decision.

(iii) Operator Training and Experience — As with Joint Intervenors' argument on class nine accidents, nothing new is presented on this issue. And as with class nine accidents, the Commission addressed this issue in its brief on the petition for review of the low-power license.

In any event, the circumstances regarding this issue have now changed radically so as to render it moot. By virtue of their operating the plant at low power, the operators now have extensive actual operating experience at the facility. Moreover, the staff has reported that the operators have discharged their responsibilities competently and safely and are capable of continuing to do so.

(iv) FEMA Finding on State Emergency Plan — As discussed above in § 1 regarding the Licensing Board's decision in LBP-82-70, the Director, NRR has reported that FEMA has made a finding that the California State Emergency Plan for Diablo Canyon is adequate. Accordingly, this issue cannot support a motion for a stay.

(v) Quality Assurance — Joint Intervenors' arguments here essentially repeat the arguments in their petitions for review of ALABs-756, -763 and -775. A Commission majority does not favor the petitions for review of ALAB-756. As for the petitions for review of ALAB-763 and ALAB-775, this is no different from the pendency of any exceptions before the Appeal Board when the Commission conducts an effectiveness review

> of a Licensing Board's decision. While the Commission has determined that the petitions for review of ALAB-763 and ALAB-775 do not raise issues warranting a stay, this determination is without prejudice to the Commission's ultimate disposition of the petition.

Joint Intervenors have also made no showing of irreparable injury. Their contention that operation of the plant will create a substantial risk is based on their conclusion that there is no longer any reasonable assurance that the seismic design of the plant is adequate. As discussed above, this conclusion is not supported.

CONCLUSION

For the reasons set out above, the Commission has determined that the full-power license for Diablo Canyon Unit 1 may be issued by the Director, NRR. However, this Order shall not become effective until 5:00 p.m., Eastern Daylight Time, August 17, 1984, to allow for the orderly processing of any request for expedited judicial review. Until then, no full-power license will be issued.

Commissioner Zech did not participate in this decision. An explanatory statement by Commissioner Zech is attached. Commissioner Asselstine dissents, and his separate statement is also attached.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 10th day of August 1984.

EXPLANATORY STATEMENT OF COMMISSIONER LANDO W. ZECH

The history of the licensing of the Diablo Canyon Nuclear Power Plant is complex and protracted. The record of the proceeding is voluminous. I have reviewed a considerable part of the record. I have visited the Diablo Canyon plant. I have talked to the utility management personnel, including some of the operators. However, the time available to me as a Commissioner has simply not been sufficient for me to satisfy myself that I have read, analyzed, and adequately reflected upon all the relevant material. If my vote were needed, either yea or nay, I believe I would need several more weeks before I could come to a decision. Therefore, I have concluded that I cannot vote today on the full-power license decision for Diablo Canyon.

DISSENTING VIEWS OF COMMISSIONER ASSELSTINE

I am unable to vote in favor of the issuance of a full-power operating license for Diablo Canyon Unit 1 at this time because of the Commission's treatment of two issues: the complicating effects of earthquakes on emergency planning, and the reevaluation of the adequacy of seismic design for small- and large-bore piping in the plant. The Commission's decision regarding the effects of earthquakes on emergency planning is being addressed in a separate order, and my views on the Commission's handling of this issue will be set forth in detail there. Suffice it to say here that this issue is material to the Commission's licensing decision in the Diablo Canyon case and that the Commission is compelled as a matter of law and logic to afford the parties to this proceeding an opportunity to litigate the issue prior to authorizing the issuance of a full-power license for the plant.

With regard to seismic design, the record of this proceeding, allegations filed by former workers at the site and subsequent NRC inspections, including those performed by NRC Inspector Isa Yin, all document a widespread quality assurance breakdown in the seismic design work for small-bore piping in the plant. This quality assurance breakdown raises serious questions regarding both the adequacy of quality assurance for other design activities for the plant and the adequacy of the Independent Design Verification Program (IDVP). Those questions are of special importance for the IDVP, which was established to verify that the seismic design problems that led to the Commission's suspension of the Diablo Canyon low-power license had been identified and corrected.

These questions existed at the time that the Commission authorized the reinstatement of the low-power license for Diablo Canyon Unit 1. When I voted to permit low-power operation, it was with the understanding that Mr. Yin and other elements of the NRC staff were in agreement on the measures needed to resolve those questions prior to a Commission decision authorizing full-power operation. I am particularly disappointed in the staff's subsequent handling of Mr. Yin's concerns. Given the special significance of seismic design for this plant and the extent of the quality assurance breakdown in the seismic design program for portions of the plant, it was incumbent on the NRC staff to make every effort to verify that all significant design errors had in fact been identified and corrected. Based upon the continuing concerns expressed by Mr. Yin regarding the adequacy of the staff's verification efforts and the extent of the seismic design quality assurance breakdown in the case, I am not yet satisfied that the Commission has the information needed to conclude, with a high degree of confidence, that all significant seismic design errors for this plant have been identified and corrected. The Agency's handling of these questions is particularly unfortunate since the adequacy of the seismic design of the plant is a matter of public concern and since it appears that an adequate design verification program to resolve Mr. Yin's concerns could be completed in a matter of a few weeks.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

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)

September 12, 1984

ORDER

Attached is an additional Statement of Commissioner Lando W. Zech, Jr., dated September 11, 1984 in this matter.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 12th day of September 1984.

STATEMENT OF COMMISSIONER LANDO W. ZECH, JR.
(September 11, 1984)

1. On August 17, 1984, a divided panel of the United States Court of Appeals for the District of Columbia Circuit ordered that the NRC's August 10, 1984, Order authorizing full-power operation of the Diablo Canyon Nuclear Power Plant (Unit 1) be stayed pending the Court's review. The Court's order cited one clause from my explanatory statement for not participating in a vote on the NRC's August 10, 1984, Order (CLI-84-13, 20 NRC 267). It appears that the Court's order may have misinterpreted the basis for my not participating in this decision. In view of the extreme importance of this matter to all of the interests involved, and to my personal responsibilities as a Commissioner, I want to leave no doubt at all on my position in this matter on August 10, 1984.

2. I did not participate in the Diablo Canyon vote in CLI-84-13. I was sworn in as a new Commissioner on July 5, 1984, a little more than 1 month prior to the August 10 decision. I explained in my statement:

> the time available to me as a Commissioner has simply not been sufficient for me to satisfy myself that I have read, analyzed, and adequately reflected upon all the relevant material. If my vote were needed, either yea or nay, I believe I would need several more weeks before I could come to a decision. Therefore, I have concluded that I cannot vote today on the full-power license decision for Diablo Canyon.

CLI-84-13, *supra,* 20 NRC at 281.

3. I did not say, and did not intend to say, that the much longer period of time to review the *Diablo Canyon* matter which was available to my colleagues prior to July 5, 1984, was not adequate. They all had much more than the "several more weeks" which I, as the newest Commissioner, said that I would need "before I could come to a decision."

4. I had absolutely no basis on August 10, 1984, to question the correctness of the decision reached on that date by a majority of my colleagues to authorize the full-power operation of Diablo Canyon Nuclear Power Plant (Unit 1). Any different interpretation of my explanatory statement by the Court in its August 17, 1984, order simply does not accurately reflect my position on August 10, 1984.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

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) **August 20, 1984**

The Commission decides not to review the Appeal Board's conclusions contained in ALAB-763, 19 NRC 571 (1984), concerning the adequacy of the operating license applicant's quality assurance program, except for a matter relating to the propriety of the Appeal Board's exclusion of certain contentions from the reopened hearing that was the subject of ALAB-763. The Commission indicates its agreement with the Appeal Board's exclusion of those contentions, but modifies the Board's reasoning for that action.

ORDER

The Commission has reviewed the petitions for review of the Atomic Safety and Licensing Appeal Board's decision in ALAB-763, 19 NRC 571 (1984), and has determined not to review that decision, subject to the following reservation. This reservation relates to the Appeal Board's rationale for excluding from the reopened hearing contentions by the Joint Intervenors and Governor of California on whether Pacific Gas and Electric (PG&E) has a quality assurance program for the design of structures, systems and components that are "important to safety" within the meaning of Appendix A to 10 C.F.R. Part 50.

The record clearly shows that as early as 1974, PG&E's Final Safety Analysis Report (FSAR) publicly disclosed PG&E's classification of equipment for the purposes of complying with the NRC's quality assurance requirements. Moreover, it has been several years since the possible distinctions between "safety-related" and "important to safety" were fully aired by NRC staff. Nothing in the events which have transpired since then constitutes new information regarding PG&E's scheme for classifying equipment for the purposes of complying with NRC regulations on quality assurance. Accordingly, as contended by the NRC staff below, the proposed contentions on PG&E's compliance with Appendix A were proffered grossly out of time.

The record also shows, as argued by the NRC staff below, that the proffered contentions lack the requisite specificity. *See* 10 C.F.R. § 2.714(a). The contentions do not identify any particular structures, systems or components for which it is claimed that the quality assurance program was not commensurate with their safety function.[1]

Under these circumstances, the Commission finds that the record clearly shows that the proposed contentions regarding PG&E's compliance with Appendix A to Part 50 were raised far too late and without the requisite specificity for their admission into the reopened proceeding. Accordingly, the Commission finds no reason to review the Appeal Board's determination not to admit those contentions, but deems the Appeal Board's decision to be modified to the extent necessary for consistency with this Order.

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[1] *See Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), CLI-84-9, 19 NRC 1323 (1984).

Commissioner Asselstine disapproved this Order. Commissioner Zech did not participate.

It is so ORDERED.

For the Commission

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, D.C.,
this 20th day of August 1984.

Cite as 20 NRC 288 (1984) CLI-84-15

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick N. Bernthal
Lando W. Zech, Jr.

In the Matter of

Docket Nos. PR-50
PR-51
(44 Fed. Reg. 61,372)

RULEMAKING ON THE STORAGE AND DISPOSAL OF NUCLEAR WASTE
(Waste Confidence Rulemaking)

August 22, 1984

The Commission sets out its findings in this waste confidence rulemaking proceeding called for by the Court of Appeals for the District of Columbia Circuit in *Minnesota v. NRC*, 602 F.2d 412 (1979). In general, the Commission finds that it can, with reasonable assurance, reach favorable conclusions with respect to the safe storage and disposal of high-level radioactive waste and spent fuel. Specifically the Commission finds reasonable assurance that: (1) safe disposal of high-level radioactive waste and spent fuel in a mined geologic repository is technically feasible; (2) one or more mined geologic repositories for commercial high-level radioactive waste and spent fuel will be available by the years 2007-09, and that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of existing commercial high-level radioactive waste and spent fuel originating in such reactor and generated up to that time; (3) high-level radioactive waste and spent fuel will be managed in a safe manner until sufficient repository capacity is available to assure the safe disposal of all high-level radioactive waste and spent fuel; (4) if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations; and (5) safe independent onsite or offsite spent fuel storage will be made available if such storage capacity is needed.

DECISION

1.0 INTRODUCTION

1.1 Initiation of the Waste Confidence Rulemaking Proceeding

In response to the remand of the U.S. Court of Appeals for the District of Columbia Circuit (*Minnesota v. NRC*, 602 F.2d 412 (1979)), and as a continuation of previous proceedings conducted in this area by NRC (44 Fed. Reg. 61,372), the Commission initiated a generic rulemaking proceeding on October 25, 1979. In its Notice of Proposed Rulemaking, the Commission stated that the

> purpose of this proceeding is solely to assess generically the degree of assurance now available that radioactive waste can be safely disposed of, to determine when such disposal or offsite storage will be available, and to determine whether radioactive wastes can be safely stored on site past the expiration of existing facility licenses until offsite disposal or storage is available.

The Commission also stated that in the event it determined that onsite storage of spent fuel would be necessary or appropriate after the expiration of facility licenses, it would propose a rule addressing the environmental and safety implications of such storage. The Commission recognized that the scope of this generic proceeding would be broader than the Court's instruction, which required the Commission to address the questions of whether offsite storage for spent fuel would be available by the expiration of reactor operating licenses and if not, whether spent fuel could continue to be safely stored on site (44 Fed. Reg. 61,373).

However, the Commission believed that the primary public concern was whether nuclear waste could be disposed of safely rather than with an offsite solution to the storage problem *per se*. Moreover, as stated in the *Federal Register* Notice on October 25, 1979, the Commission committed itself to reassess its basis for reasonable assurance that methods of safe permanent disposal of high-level waste would be available when they are needed. In conducting that reassessment, the Commission noted that it would "draw upon the record compiled in the Commission's recently concluded rulemaking on the environmental impacts of the nuclear fuel cycle (44 Fed. Reg. 45,362-74 [August 2, 1979])" (44 Fed. Reg. 61,373).

The Department of Energy (DOE), as the lead agency on nuclear waste management, filed its statement of position (PS) on April 15, 1980. Statements of position were filed by thirty participants by June 9, 1980, and were followed by cross-statements (CS) from twenty-one of the participants by August 11, 1980.

1.2 Establishment of the Working Group

On May 28, 1980, the Commission directed the staff to form a Working Group to advise the Commission on the adequacy of the record to be compiled in this proceeding, to review the participants' submissions and identify issues in controversy and any areas in which additional information would be needed. The Working Group submitted a report to the Commission on January 29, 1981. The report summarized the record, identified key issues and controversies, and commented on the adequacy of the record for considering the key issues. The participants were invited to submit comments on the adequacy of the Working Group's summary of the record and its identification and description of the issues. Such comments were made by twenty participants by March 5, 1981.

1.3 Commission's Order for Oral Presentations

The Commission found additional limited proceedings to be useful to allow the participants to state their basic positions directly to the Commissioners and to enable the Commissioners to discuss specific issues with them. In addition, the Commission invited comment on the following policy developments: (1) the Administration's announcement[1] of a policy favoring commercial reprocessing of spent fuel and instructing the Secretary of Energy to proceed swiftly toward deployment of a means of storing and disposing of commercial high-level radioactive waste, and (2) the submission of information to the Presiding Officer in

[1] Presidential Nuclear Policy Statement, October 9, 1981.

this proceeding by DOE on March 27, 1981, concerning the DOE decision to "discontinue [its] efforts to provide federal government-owned or -controlled away-from-reactor (AFR) [spent fuel] storage facilities." The participants were asked to comment on the significance to the proceeding of issues, particularly institutional concerns, resulting from these policy developments and to comment on the merits of DOE's new projection of spent fuel storage requirements and on the technical and practical feasibility of DOE's suggested alternative storage methods.

To implement the additional limited proceedings, the Commission consolidated the participants into the following identifiable groups: (a) Federal government, (b) State and local participants, (c) industry, and (d) public interest groups (Second Prehearing Memorandum and Order, November 6, 1981 (unpublished)). Prehearing statements (PHS) were provided by the consolidated groups, as well as by individual participants. The oral arguments were presented to the Commissioners on January 11, 1982.

The extensive record, comprised of all written and oral submissions, provides the primary basis for the Commission's decision regarding the safe storage and disposal of spent fuel and nuclear waste. However, while the Commission was preparing this Waste Confidence decision, the Nuclear Waste Policy Act of 1982 (NWPA) was enacted. The Commission found that this Act had a significant bearing on the Commission's decision, and the Commission has considered the NWPA in reaching its conclusions. The Commission believes that the NWPA had its most significant impact in narrowing the uncertainties surrounding institutional issues. Moreover, although the NWPA is intrinsically incapable of resolving technical issues, it will establish the necessary programs, milestones, and funding mechanisms to enable their resolution in the years ahead.

The Commission's preliminary decision in the Waste Confidence proceeding was served on the consolidated participants on May 17, 1983. However, the parties to this proceeding had not yet had an opportunity to comment on what implications, if any, the NWPA had on the Commission's decision. Further, the Commission's discussion of the safety of dry storage of spent nuclear fuel, in its preliminary decision, relied substantially on material not yet in the record. Therefore, the preliminary decision was issued as a draft decision. The Commission requested the consolidated groupings of participants to comment on either or both of these issues. In addition, the Commission found that onsite storage after license expiration might be necessary or appropriate, and therefore, in accordance with its notice initiating this proceeding, it proposed a rule to establish how the environmental effects of extended onsite storage would be considered in licensing proceedings (48 Fed. Reg. 22,730 (1983)), as amendments to 10 C.F.R. Parts 50 and 51.

Subsequently, in response to public comments on the proposed amendments to 10 C.F.R. Part 51, the Commission reopened the comment period to address the environmental aspects of the fourth finding of the Commission's Waste Confidence decision, on which the proposed amendment to Part 51 is based (48 Fed. Reg. 50,746 (1983)). Public comments were requested on: (1) the environmental aspects of the fourth finding — that the Commission has reasonable assurance that, if necessary, spent fuel can be stored without significant environmental effects for at least 30 years beyond the expiration of reactor operating licenses at reactor spent fuel storage basins, or at either onsite or offsite independent spent fuel storage installations; (2) the determination that there are no significant nonradiological consequences which could adversely affect the environment if spent fuel is stored beyond the expiration of operating licenses either at reactors or at independent spent fuel storage installations; and (3) the implications of comments on items (1) and (2) above for the proposed amendment to 10 C.F.R. Part 51.

After reviewing these additional comments, the Commission found no reason to modify its fourth finding or the supporting determination. The analysis of comments, together with the Commission's response is summarized in the Addendum to the Commission's decision.

The Commission notes that two relevant developments have occurred subsequent to the closing of the record in the Waste Confidence proceeding. They are the publication of DOE's draft Mission Plan for the Civilian Radioactive Waste Management Program (April 1984) and the Commission's concurrence in DOE's General Guidelines for Recommendation of Sites for Nuclear Waste Repositories (July 3, 1984). These developments are a matter of public record, and in the case of the Commission's concurrence was the conclusion of a separate public proceeding. The Commission has considered the effects of these developments on its previously announced decision in this proceeding and determined that these developments do not substantially modify the Commission's previous conclusions.

The decision is summarized as five Commission findings in § 2.0. The detailed rationale for these findings, including references to the record developed in this proceeding, is contained in the Appendix to this document. The Commission considers these five findings to be a response to the mandate of the U.S. Court of Appeals for the District of Columbia Circuit and, in addition, a generic determination that there is reasonable assurance that radioactive waste can and will be safely stored and disposed of in a timely manner.

In keeping with its commitment to issue a rule providing procedures for considering environmental effects of extended onsite storage of spent fuel in licensing proceedings, final amendments to 10 C.F.R. Parts 50 and 51 are being issued simultaneously with this decision.

2.0 COMMISSION FINDINGS[2]

1. The Commission finds reasonable assurance that safe disposal of high-level radioactive waste and spent fuel in a mined geologic repository is technically feasible.

2. The Commission finds reasonable assurance that one or more mined geologic repositories for commercial high-level radioactive waste and spent fuel will be available by the years 2007-09, and that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of existing commercial high-level radioactive waste and spent fuel originating in such reactor and generated up to that time.

3. The Commission finds reasonable assurance that high-level radioactive waste and spent fuel will be managed in a safe manner until sufficient repository capacity is available to assure the safe disposal of all high-level radioactive waste and spent fuel.

4. The Commission finds reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations.

5. The Commission finds reasonable assurance that safe independent onsite or offsite spent fuel storage will be made available if such storage capacity is needed.

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[2] All findings by the Commission in this proceeding are limited to the storage and disposal of high-level radioactive waste and spent fuel generated by nuclear power reactors required to be licensed under §§ 103 or 104b of the Atomic Energy Act of 1954 (42 U.S.C. §§ 2133 and 2134(b)), and to facilities intended for such storage or disposal. The Commission's findings in this proceeding do not address the storage and disposal of high-level radioactive waste or spent fuel resulting from atomic energy defense activities, research and development activities of the Department of Energy, or both. This is consistent with the Nuclear Waste Policy Act of 1982, § 8(c).

3.0 FUTURE ACTIONS BY THE COMMISSION

The Commission's Waste Confidence decision is unavoidably in the nature of a prediction. While the Commission believes for the reasons set out in the decision that it can, with reasonable assurance, reach favorable conclusions of confidence, the Commission recognizes that the possibility of significant unexpected events remains open. Consequently, the Commission will review its conclusions on waste confidence should significant and pertinent unexpected events occur, or at least every 5 years until a repository for high-level radioactive waste and spent fuel is available.

4.0 FOR FURTHER INFORMATION

Contact Dennis Rathbun or Clyde Jupiter, Office of Policy Evaluation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone (202) 634-3295, or Sheldon Trubatch, Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; telephone (202) 634-3224.

Commissioner Zech did not participate in this action.

For the Commission

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.,
this 22nd day of August 1984.

Addendum to the Commission's Waste Confidence Decision

INTRODUCTION

On May 17, 1983, the Commission issued its proposed decision in the Waste Confidence proceeding, and asked the consolidated groups of participants to comment on two aspects of the decision: the implications of the Nuclear Waste Policy Act (NWPA) for the decision and the Commission's discussion of the safety of dry storage of spent nuclear fuel, which relied substantially on material not in the record. The analysis of these comments is subdivided into several issue categories and presented, with NRC's responses, in Part I below. The membership of the consolidated groups responding to the Commission's request as well as the abbreviations used to identify the groups are provided in § 3 of Part I.

Subsequently, in response to public comments on the Commission's proposed amendment to 10 C.F.R. Part 51 (48 Fed. Reg. 22,730 (1983)), the Commission reopened (48 Fed. Reg. 50,746 (1983)) the comment period to address the environmental aspects of the fourth finding of the Commission's proposed Waste Confidence decision on which the proposed amendment to Part 51 is based. Public comments were requested on: (1) the environmental aspects of the fourth finding — that the Commission has reasonable assurance that, if necessary, spent fuel can be stored without significant environmental effects for at least 30 years beyond the expiration of reactor operating licenses at reactor spent fuel storage basins, or at either onsite or offsite independent spent fuel storage installations; (2) the determination that there are no significant nonradiological consequences which could adversely affect the environment if spent fuel is stored beyond the expiration of operating licenses either at reactors or at independent spent fuel storage installations; and (3) the implications of comments on items (1) and (2) above for the proposed amendment to 10 C.F.R. Part 51. The analysis of public comments and NRC's responses are presented in Part II of this addendum. The list of respondents to this reopened comment period and the abbreviations used to identify them are given in § 4 of Part II.

The Commission notes that two relevant developments have occurred subsequent to the closing of the record in the Waste Confidence proceeding. They are the publication of DOE's draft Mission Plan for the Civilian Radioactive Waste Management Program (April 1984) and the Commission's concurrence in DOE's General Guidelines for Recommendation of Sites for Nuclear Waste Repositories (July 3, 1984). These developments are a matter of public record, and in the case of the Commission's concurrence was the conclusion of a separate public proceeding. The Commission has considered the effects of these developments on its previously announced decision in this proceeding and determined that these developments do not substantially modify the Commission's previous conclusions.

PART I: ANALYSIS OF THE CONSOLIDATED GROUPS' COMMENTS ON THE COMMISSION'S WASTE CONFIDENCE DECISION AND NRC RESPONSES

1. Effect of the Nuclear Waste Policy Act on the Commission's Decision

A. General

(1) Summary of Comments

The Consolidated Industry Group agreed with the Commission's view that the NWPA contains provisions pertinent to all of the major elements relevant to mined geologic disposal of high-level radioactive wastes (Industry at 3). The Industry Group called attention to the comprehensive nature of the NWPA which authorizes DOE to undertake steps leading to the construction, operation and maintenance of a deep geologic test and evaluation facility; requires DOE to prepare a waste management mission plan; establishes a prescribed schedule for repository siting, construction and operation; defines the decisionmaking roles of affected States and Indian tribes in repository site selection and evaluation; provides for the continuity of Federal management of the nuclear waste program and continued funding; and facilitates the establishment of an overall integrated spent fuel and waste management system. The Industry Group suggested that these features of the Act should increase the Commission's confidence that waste can and will be disposed of safely. The Group pointed out that the Act also contains special procedures to facilitate the licensing of spent fuel storage capacity expansion and transshipments; directs DOE research, development and cooperation with utilities in developing dry storage and rod compaction; and provides for federally supplied interim storage capacity to supplement that of industry (Industry at 4-8).

The Industry Group believed that the NWPA's enactment — in and of itself — provides a sound basis for confidence that institutional difficulties can and will continue to be resolved. At the same time, Industry stated that the NWPA's enactment was not essential for the Commission to reach an affirmative decision in this proceeding (Industry at 9).

In contrast, the Consolidated Public Interest Group (CPIG) believed that the NWPA provides an insufficient basis for the Commission's decision in this proceeding with respect to the availability or timing of a nuclear waste repository. The CPIG contended that the NWPA contains many areas of ambiguity, and gave as examples:

(i) Section 114(a) of the NWPA requires DOE to make a recommendation to the President for the first repository site, accompanied by the preliminary comments by the Commission concerning the suitability of three alternative candidate sites for licensing under 10 C.F.R. Part 60. DOE interprets this section to require such preliminary comments *before* site characterization begins. . . . The Commission staff interprets that section . . . to require a judgment of suitability under 10 C.F.R. Part 60 *after* site characterization has occurred.

(ii) DOE originally interpreted § 112(f) to permit continuation of ongoing site characterization at Hanford before completion of the DOE siting guidelines. DOE now concedes that such site characterization work must await completion of an environmental assessment prepared in accordance with final DOE siting guidelines.

(CPIG at 2-3).

(2) NRC Response

The Commission has considered the effect of enactment of the Nuclear Waste Policy Act of 1982 and concludes that the Act provides support for timely resolution of technical uncertainties and reduces uncertainties in the institutional arrangements for the participation of affected States and Indian tribes in the siting and development of repositories and in the long-term management, direction and funding of the repository program. The bases for the Commission's conclusion are set forth in the decision and will not be repeated here. The passage of the Act provides evidence of a strong national commitment to the solution of the radioactive waste management problem.

The Commission recognizes the possibility of differing interpretations regarding the implementation of the NWPA. With respect to CPIG's discussion of § 114(a), the Commission is unaware of any differences between DOE and NRC in the interpretation of this section of the Act. We note that DOE's recommendation of a repository site to the President would necessarily be made after DOE's preliminary determination that three sites are suitable for development. DOE and NRC now agree that the preliminary determination of site suitability for the alternative sites should be made following site characterization (Commission's Final Decision on the U.S. Department of Energy's General Guidelines for the Recommendation of Sites for Nuclear Waste Repositories (July 3, 1984)).

Concerning § 112(f), DOE has continued site characterization at Hanford during formulation of the siting guidelines; in accordance with the views of the States and environmental groups, DOE has deferred drilling of the exploratory shaft pending the completion of the guidelines, submission of the site characterization plan to NRC and preparation of an environmental assessment of site characterization activities.

B. Technical Aspects

(1) Summary of Comments

The Consolidated Industry Group believed that the Act contained provisions pertinent to all of the major elements relevant to disposal (Industry at 3). The Consolidated Public Interest Group, on the other hand, contended that the NWPA did not resolve technical uncertainties concerning repository development and safety (CPIG at 5). The Consolidated State Group did not believe that the NWPA supported a finding of confidence because it failed to resolve technical questions and merely set target dates for deciding on the site of the first waste repository. The State Group noted that if technical problems are not resolved by the dates proposed by Congress, the milestone dates will have to be postponed. The State Group contended too that, although the Act authorizes DOE to conduct research on unresolved technical issues, the research could uncover additional problems (States at 2). However, DOE pointed out that the NWPA provides for a focused, integrated and extensive research and development program for the deep geologic disposal of high-level waste and spent fuel. DOE believed that § 215 of the Act enhances confidence in the timely availability of disposal facilities by authorizing a research facility to develop and demonstrate a program for waste disposal. DOE also stated that the schedule for a Test and Evaluation Facility would require the *in situ* testing described in § 217 of the Act to begin not later than May 6, 1990, thus allowing for research and development results to be incorporated in the repository which is scheduled to open in 1998 (DOE at 11, 12).

(2) NRC Response

As the record of this proceeding shows, there are no known technical problems that would make safe waste disposal impossible. Clearly, further engineering development and site-specific evaluations will be required before a repository can be constructed. The Commission did not propose to rely on the NWPA as the basis for resolving technical uncertainties. Rather, the Commission found that the NWPA provides a framework for facilitating the solution of the remaining technical issues. Title II of the Act authorizes DOE to undertake steps leading to the construction, operation and maintenance of a deep geologic test and evaluation facility and to conduct the necessary research and development as well as to establish a demonstration program. The schedule set forth in the Act is consistent with the objective of assuring repository operation within the time period discussed in the Waste Confidence decision. The "Mission Plan" which is required by the Act will provide an effective management tool for assuring that the many technical activities are properly coordinated and that results of research and development projects are available when needed.

C. *Institutional Aspects*

(1) Summary of Comments

The Consolidated State Group believed that the NWPA failed to resolve institutional questions. The States argued that their cooperation cannot be assumed in the event that the general public in the vicinity of a proposed site is opposed to the location. Further, the States contended that, if a site is vetoed by a host State or Indian tribe, there is no assurance that Congress will vote to override the veto. Moreover, if the veto is overridden, a legal challenge is likely and the outcome is uncertain (States at 3).

The Consolidated Public Interest Group also believed that the NWPA has not significantly reduced institutional uncertainties regarding participation and objections of affected States and Indian tribes. As examples of institutional difficulties, CPIG pointed out that State officials and Indian tribes still have concerns regarding the adequacy of time to monitor and comment upon agency proposals, the lack of agency response to their concerns, and inadequate funding to support their full participation. Further, CPIG noted that the Act (§ 115) provides States and Indian tribes with strong new authority to veto the siting of a repository within their borders (CPIG at 5).

DOE, on the other hand, believed that §§ 116 and 117 of the NWPA would reduce Federal-State institutional uncertainties (DOE at 9).

(2) NRC Response

It would be unrealistic to expect that the NWPA will resolve all institutional issues. However, it does provide specific statutory procedures and arrangements for accomplishing such resolution. The right of affected States and Indian tribes to disapprove a site designation under the NWPA might create uncertainty in gaining the needed approvals. Nevertheless, the NWPA's establishment of a detailed process for State and tribal participation in the development of repositories and for the resolution of disputes should minimize the potential for substantial disruption of plans and schedules. The Commission does not expect that the NWPA can eliminate all disagreement about development of waste repositories. However, in providing for information exchange, financial and technical assistance to affected groups, and meaningful participation of affected States and tribes in the decisionmaking process, the Act should minimize the potential for direct confrontations and disputes.

D. Funding Aspects

(1) Summary of Comments

The Consolidated Industry Group expressed its general belief that the NWPA assures adequate funding for interim storage and disposal of radioactive waste (Industry at 6, 7). Similarly, DOE believed that the funding mechanism provided by the NWPA should largely remove uncertainties in assuring adequate resources to complete the program (DOE at 10, 11). On the other hand, the Consolidated States Group contended that, since the law can be changed at any time, the NWPA assures neither an adequate level of funding nor a prolonged congressional commitment (States at 4).

(2) NRC Response

The Commission believes that the general approach prescribed by the NWPA is to operate DOE's radioactive waste program on a full-cost-recovery basis. It seems clear that Congress intended to establish a long-term program for waste management and disposal, with built-in reviews and adjustments of funding as necessary to meet changing requirements. In this regard, the Act provides that DOE must annually review the amount of the established fees to determine whether collection of the fees will provide sufficient revenues to offset the expected costs. In the event DOE determines that the revenues being collected are less than the amount needed to recover costs, DOE must propose to Congress an adjustment to the fees to ensure full cost recovery. The Act also provides that, if at any time, the monies available in the waste fund are insufficient to support DOE's nuclear waste program, DOE will have the authority to borrow from the Treasury. The Commission believes that the long-term funding provisions of the Act will ensure adequate financial support for DOE's nuclear waste program for FY 1984 and beyond.

The Commission believes that uncertainties regarding the adequacy of financial management of the nuclear waste program have also been reduced by the NWPA requirement that an Office of Civilian Radioactive Waste Management be established within the Department of Energy. This Office is to be headed by a Director, appointed by the President with Senate confirmation, who will report directly to the Secretary of Energy. Further, the Act stipulates that an annual comprehensive report of the activities and expenditures of the Office will be submitted to Congress and that an annual audit of the Office will be conducted by the Comptroller General, who will report the results to Congress.

Some concern has been expressed that the Congress may amend the funding provisions of the NWPA and thereby undermine the financial stability of the Federal radioactive waste management program. Commenters have not provided any basis for this belief. The Commission considers this possibility to be most unlikely. It is reasonable to assume that the long-range public health and safety and political concerns which motivated the Congress over the past several years to pass the NWPA will continue to motivate the Congress in considering amendments to the NWPA.

E. Schedule

(1) Summary of Comments

DOE contended that the NWPA provides additional assurance that a repository will be available by 1998. As the basis for this belief, DOE stated that §§ 111 through 125 of the NWPA provide specific schedules and reporting requirements for the timely siting, development, construction, and operation by 1998 of a repository for high-level waste and spent fuel (DOE at 6). DOE believed that these schedules and reporting requirements will ensure that deadlines are met. The Commission notes that DOE recognizes that there has been a delay of about 1 year in its schedule for meeting early milestones such as publication of its siting guidelines; nevertheless, DOE continues to maintain that its date for completion of repository development will be met (DOE draft Mission Plan for the Civilian Radioactive Waste Management Program, April 1984).

The Consolidated Public Interest Group, however, did not believe that the provision of specific dates in the NWPA gives assurance that they will be met. CPIG cited, for example, the delay in preparing DOE's site-selection guidelines, which were due by June 1983, and were expected to be delayed further (CPIG at 4).

Further, the CPIG contended that a date for the availability of a repository is not certain since both the President and the NRC have explicit authority to reject any or all site proposals that are submitted to them (CPIG at 4). Also, CPIG believed that the legislation contemplates the possibility of delay beyond statutory deadlines and NWPA's legislative history indicates that the timing of repository availability remains uncertain (CPIG at 5).

(2) NRC Response

One of the primary purposes of the NWPA is "to establish a schedule for the siting, construction, and operation of repositories that will provide reasonable assurance that the public and the environment will be adequately protected from the hazards posed by high-level radioactive waste and such spent nuclear fuel as may be disposed of in a repository." (§ 111(b)(1)). The Commission believes this purpose will be achieved.

As the Commission noted in the proposed decision, the Congress would not be able to legislate the schedules for the accomplishment of fundamental technical breakthroughs if it believed that such breakthroughs were necessary. They are not necessary. Rather, it is the Commission's judgment that the remaining uncertainties can be resolved by the planned step-by-step evaluation and development based on ongoing site studies and research programs. The Commission believes the Act provides means for resolution of those institutional and technical issues most likely to delay repository development, both because it provides an assured source of funding and other significant institutional arrangements, and because it provides detailed procedures for maintaining progress, coordinating activities and rectifying weaknesses.

The Commission believes that the milestones established by the Act are generally consistent with the schedules presented by DOE in the Waste Confidence proceeding and that those milestones are generally reasonable. Achievement of the scheduled first date of repository operation is further supported by other provisions of the Act which specify means for resolution of issues most likely to delay repository completion. One of the earlier milestones — publication of DOE's general guidelines for the recommendation of sites for a repository — was about a year behind schedule and the Commission was concerned that this delay could result in corresponding delays in DOE's nomination of at least five sites for characterization work. However, DOE has indicated in its draft Mission Plan (April 1984) that the subsequent milestones have been scheduled to provide completion of the first repository by 1998.

The Commission believes that the timely attainment of a repository does not require DOE's program schedule to adhere strictly to the milestones set out in the NWPA over the approximately 15-year duration of the repository development program. Delays in some milestones as well as advances in others can be expected.

The Commission has no evidence that delays of a year or so in meeting any of the milestones set forth in the NWPA would delay the repository availability date by more than a few years beyond the 1998 date specified in the NWPA. The Commission found reasonable assurance that a repository would be available by 2007-09, a decade later than that specified in the NWPA, and a date which allows for considerable slippage in the DOE schedule. The Act also requires that any Federal agency that determines that it cannot comply with the repository development schedule in the Act must notify both the Secretary of Energy and Congress, provide reasons for its inability to meet the deadlines, and submit recommendations for mitigating the delay. The Commission notes that the Act also clarifies how the requirements of the National Environmental Policy Act are to be met. These provisions of the Act, as well as the provisions for research, development and demonstration efforts regarding waste disposal, increase the prospects for having the first repository in operation not later than the first few years of the next century.

The repository development schedule may have to accommodate such contingencies as vetoes of proposed repository sites, prolonged public hearings, protracted litigation, possible project reorientation, or delay in promulgation of siting guidelines. The schedule now incorporated into the Act allows substantial time for these possibilities.

2. Discussion of the Safety of Dry Storage

A. Summary of Comments

DOE believed that the availability of dry storage techniques provides further reasonable assurance of the ability to safely store nuclear wastes at least 30 years beyond the expiration of reactor operating licenses. DOE stated that the citations quoted in the Commission's rationale are reliable and representative of the literature in the area, and that the Commission's technical judgment on dry storage conforms with DOE's experience and is accurate and correct (DOE at 16). The Consolidated Industry Group also stated that the pertinent points in the Commission's discussion appear to be adequately supported with appropriate references (Industry at 10, 11).

In further support of the safety of dry storage, DOE cited the following:

- Extensive worldwide experience shows that dry fuel handling and storage is safe and efficient. Irradiated fuel has been handled, shipped, and safely stored under dry conditions since the mid-1940s. All types of irradiated fuel have been handled dry at hot cells, where a variety of phenomena have been observed in detail. The passive nature of most dry storage concepts contributes to the safety of interim storage by not requiring active cooling systems involving moving parts (DOE at 16).
- Regarding specific experience, DOE stated that reactor fuel has been successfully stored in dry vaults licensed under Part 50 at the Hallam sodium-cooled graphite research reactor in Nebraska and the Fort St. Vrain HTGR prototype facility in Colorado. In addition, dry storage of zircaloy-clad fuel has been successfully conducted in drywells and in air-cooled vaults at DOE's Nevada Test Site. There is favorable foreign experience with dry storage at Wylfa, Wales in Great Britain, at Whitesell in Canada, in the Federal Republic of Germany, in France where vault dry storage of vitrified waste is routine, and in Japan, where a dry storage vault has been recently constructed (DOE at 17).
- To date, all dry storage tests have indicated satisfactory storage of zircaloy-clad fuel without cladding failure over the temperature range of 100°C to 570°C, in inert atmospheres. Existing data which support the conclusion that spent fuel can be stored safely in an inert atmosphere for at least 30 years is being augmented by additional ongoing research (DOE at 17, 18).

None of the consolidated groups of participants offered comments which were critical of the Commission's discussion of the safety of dry storage.

B. NRC Response

The Commission is confident that dry storage installations can provide continued safe storage of spent fuel at reactor sites for at least 30 years after expiration of the reactor operating licenses.

3. List of Respondents

CONSOLIDATED PARTICIPANTS AS RESPONDENTS TO THE COMMISSION'S WASTE CONFIDENCE DECISION

1.	Department of Energy	(DOE)
2.	Consolidated States Representative[1]	(States)
3.	Consolidated Public Interest Representative[2]	(CPIR)
4.	Consolidated Industry Representative[3]	(Industry)

PART II: COMMISSION CONSIDERATION OF ADDITIONAL COMMENTS ON ITS FOURTH FINDING

1. Introduction

On November 3, 1983, the Commission reopened the comment period in this proceeding to receive comments on: (1) the environmental aspects of its fourth finding — that it has reasonable assurance that, if necessary, spent fuel can be stored without significant environmental effects for at least 30 years beyond the expiration of reactor operating licenses at reactor spent fuel storage basins, or at either onsite or offsite independent spent fuel storage installations; (2) the determination that there are no significant nonradiological consequences which could adversely affect the environment if spent fuel is stored beyond the expiration of operating licenses either at reactors or at independent spent fuel storage installations; and (3) implications of comments on items (1) and

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[1] The Consolidated States Group consists of the Attorney General of the State of New York, Minnesota (by its Attorney General and the Minnesota Pollution Control Agency), Ohio, South Carolina and Wisconsin. The remaining participants previously consolidated in the States Group have not joined in these comments.

[2] The Consolidated Public Interest Group is represented here by the Natural Resources Defense Council, Inc., the New England Coalition on Nuclear Pollution, the Sierra Club, the Environmental Coalition on Nuclear Power, Wisconsin's Environmental Decade, Mississippians Against Disposal, Safe Haven, Ltd., John O'Neill, Jr., and Marvin Lewis.

[3] The Consolidated Industry Group is represented by: American Institute of Chemical Engineers; American Nuclear Society; Association of Engineering Geologists; Atomic Industrial Forum; Bechtel National; Consumers Power; General Electric; Neighbors for the Environment; Scientists and Engineers for Secure Energy; Tennessee Valley Authority; the Utilities group (Niagara Mohawk Power Corporation, Omaha Public Power District, Power Authority of the State of New York, and Public Service Company of Indiana, Inc.); and the Utility Nuclear Waste Management Group–Edison Electric Institute. In order to emphasize the independent nature of its participation, the American Nuclear Society has chosen to proceed separately. ANS continues to protest its assignment to the Consolidated Industry Group and has offered separate comments on the Commission's Waste Confidence decision. Since only the consolidated groups of participants were invited to comment on the proposed decision, the ANS's separate comments are not discussed here. Further, TVA, as a Federal agency, wishes to stress the independent nature of its participation.

(2) above for the proposed amendment to 10 C.F.R. Part 51 (48 Fed. Reg. 50,746).

The Commission has considered those comments and, for the reasons discussed below, finds no reason to substantively modify its fourth finding or other related aspects of its decision in this proceeding. The Commission has, however, made revisions in its fourth finding to clarify its original intent.

Thirteen comments were received. Seven commenters identified various reasons which they believed argued against the finding.[4] Six commenters supported the finding.[5] In addition to the issues on which the Commission specifically requested comments, some commenters raised additional issues regarding the Commission's compliance with the National Environmental Policy Act (NEPA).

2. Environmental Aspects of Extended Storage of Spent Fuel

A. Radiological Consequences of Spent Fuel Storage

The Commission's proposed fourth finding stated:

> The Commission finds reasonable assurance that, if necessary, spent fuel can be stored safely without significant environmental effects for at least 30 years beyond the expiration of reactor operating licenses at reactor spent fuel storage basins, or at either onsite or offsite independent spent fuel storage installations.

The public was invited to submit additional comments on the environmental aspects of this finding. Those comments, and the Commission's responses to them, are set out below.

The State of Minnesota ("Minnesota"), through its Attorney General, and the Sierra Club believe that an event at the spent fuel pool for Prairie Island Nuclear Generating Station ("Prairie Island") indicates that irradiated spent fuel assemblies are degrading rapidly with time. In December 1981, during a fuel transfer operation at Prairie Island, the top nozzle assembly separated from the remainder of a spent fuel assembly due to stress corrosion cracking of the spent fuel assembly while it was in the spent fuel pool. Minnesota and the Sierra Club acknowledge that this separation was an isolated event; over 5000 similar spent fuel

[4] Department of Law of the State of New York, Marvin Lewis, Sierra Club, Safe Haven, Ltd., Attorney General of the State of Minnesota, Department of Justice of the State of Wisconsin and Natural Resources Defense Council, Inc.

[5] Scientists and Engineers for Secure Energy, Inc., American Institute of Chemical Engineers, American Nuclear Society, Utility Nuclear Waste Management Group-Edison Electric Institute, and U.S. Department of Energy.

assemblies have been moved successfully at other plants. These commenters also acknowledge that television examination showed no corrosion cracking of similarly designed fuel assemblies at other nuclear power plants: Zion, Trojan, Kewanee and Point Beach. They also acknowledge that even though the water contaminant contributing to stress corrosion cracking has never been identified, the possibility that it may have been sulfates has led the Commission to suggest that Prairie Island monitor the sulfate levels of its spent fuel pool.

However, the Sierra Club contended[6] that the NRC staff essentially ignored the opinion of Mr. Earl J. Brown, an NRC engineer, that sulfate contamination is a generic problem at pressurized water reactors (PWRs). The Sierra Club also believes that television inspection of spent fuel assemblies in spent fuel pools cannot reveal the initial signs of stress corrosion cracking. For these reasons, the Sierra Club and Minnesota believe that there is no assurance that spent fuel can be stored safely in spent fuel pools for 30 years after reactor shutdown or for 60 years after irradiation.

The NRC investigated the Prairie Island event and found it to be an isolated event without generic impact. The staff also concluded that if a fuel assembly were to drop due to top nozzle failures, such an event would not lead to a criticality hazard in a spent fuel pool and that such an accident would result in radiation levels at the site boundary well within the limits in 10 C.F.R. Part 100. The NRC Staff Assessment Report ("SAR") and associated memoranda, although already publicly available in the Commission's Public Document Room, have been added to the docket of this proceeding. That SAR concluded that the event was caused by intergranular stress corrosion cracking due to an unidentified corrodant temporarily present in the spent fuel pool.

As for the Sierra Club's specific comments, the staff recognized that sulfate contamination was suspected to have contributed to the corrosion and recommended that licensees administratively control sulfate level concentrations in spent fuel pools. Such monitoring had been recommended by Mr. Brown as the only action that should be taken in response to the incident. Although Mr. Brown stated that in his opinion the event was a "potential" generic issue for PWRs, subsequent staff investigation revealed that the event was an isolated incident. The staff

[6] Sierra Club also stated that the staff did not consider an Oak Ridge report (ORNL-3684, November 1964) which identified water vapor as contributing to corrosion of the type of steel used in spent fuel assemblies. That report is not germane to light water reactor fuel because it addressed the sensitization of stainless steel in a high-temperature, gas-cooled reactor environment, which is very different from the environment of a light water reactor. Refer to the discussion in § 2.4A of the Appendix to the Commission's decision.

also considered the properties of the steel used in the spent fuel assemblies and acknowledged that they could have contributed to the event. However, the absence of any similar events for 5000 other spent fuel assemblies indicated that the type of steel was not critical. Accordingly, the Commission finds no basis for reconsidering the Safety Assessment Report's finding that the Prairie Island event was an isolated incident and recommendation that sulfate control was an adequate response, or for altering its conclusion concerning the potential environmental impacts of stored spent fuel.

Wisconsin, Safe Haven, Ltd., and NRDC contended that the environmental effects of extended spent fuel storage are site-specific and should be considered on a case-by-case basis.[7] Safe Haven believes that the individuality of each plant and its environmental surroundings necessitate separate evaluations of extended storage of spent fuel, but identified no site-specific factors which would result in significant environmental impacts. NRDC listed some site-specific factors: geology, hydrology, seismicity, ecological factors and individual proposals for spent fuel management and storage. However, NRDC did not suggest how these factors could lead to significant site-specific environmental impacts that would preclude the Commission from making a generic finding. Similarly, Wisconsin listed as relevant factors proximity to population centers, highways, geologic faults, dams, floodplains or shorelines affected by erosion, but offered no suggestion of how these factors could affect the Commission's generic determination. For example, there has been no discussion of why the Commission's seismic design requirements, though site-specific, are not generically adequate to assure that spent fuel can be stored for up to 30 more years in a spent fuel pool designed to withstand the largest expected earthquake at each reactor site. Mr. Marvin Lewis contended that the fourth finding had no basis because the Commission had little or no experience with storing spent fuel for 30 years or with storing fuel that could be up to 70 years old. Mr. Lewis also asserted that the pyrophoricity of the zircaloy tubes containing spent fuel for 30 years presents an unknown fire danger. This comment is based on a private communication to Mr. Lewis regarding the condition of the spent fuel at Three Mile Island, Unit 2. By the terms of that letter, any fire danger associated with pyrophoricity of zircaloy arises from the accident conditions at TMI-2. NRC has previously studied the

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[7] Safe Haven also suggested that a full environmental and safety review should accompany any utility's proposed plans submitted pursuant to 10 C.F.R. Part 50 (§ 50.54(aa)) for extended storage of spent fuel. The Commission will treat its review of any such utility proposal in accordance with the established procedures for considering any application for a license amendment.

effects of loss of water from pools on the temperature of stored spent fuel (NUREG/CR-0649, "Spent Fuel Heatup Following Loss of Water During Storage," March 1979). While this study noted that oxidation could become self-sustaining for temperatures in the neighborhood of 850-950°C (NUREG/CR-0649, at 13), the study shows that such oxidation can only occur for extreme temperature conditions and for spent fuel that has been stored for a relatively brief storage period. In order for rapid oxidation to occur, the age of the spent fuel (30,000 MWD/MT burnup) would have to be in the range of less than 10 days to less than 2 years, depending on the density at which it is stored (*see* NUREG/CR-0649, Figure 17, at 55). Moreover, one must assume a continuing oxygen supply adequate to sustain the oxidation. Any damaged spent fuel such as that from TMI-2, would be canned to avoid particulate loss and would have already aged several years. Neither the heat load leading to temperatures capable of initiating rapid oxidation nor the presence of an adequate supply of oxygen to sustain a pyrophoric reaction would seem to be present in any storage configuration or under conditions that would receive NRC approval. While it is correct that spent fuel has not been stored for over 30 years, the record shows that utilities have successfully stored spent fuel for over 20 years, and that there are no known physical processes which would indicate that it is impractical to extrapolate that experience to make predictions about the behavior of spent fuel for 70 years of storage.

The Utility Nuclear Waste Management Group — Edison Electric Institute and the U.S. Department of Energy referred to several documents in the record which show that the relatively low energy content of spent fuel and the relatively benign static environment of spent fuel storage render insignificant the radiologic impacts arising from the extended storage of spent fuel. As discussed in more detail below, these documents also show that there are no significant nonradiologic environmental impacts arising from such extended storage. Under these circumstances, the Commission finds that it has sufficient experience with spent fuel storage to predict spent fuel behavior during 70 years of storage and to find that such storage will not result in significant environmental effects.

B. Nonradiological Consequences of Spent Fuel Storage

The Commission's fourth finding rested in part on the Commission's determination that there are no significant nonradiological consequences due to the extended storage of spent fuel which could adversely affect the environment. The public was invited to comment also on this finding and to provide a detailed discussion of any such environmental impacts. Mr. Marvin Lewis asserted that the continuous storage of spent fuel under water for 30 years or more requires unprecedented institutional guarantees. He also noted that there had been no consideration of financial, economic and security implications of storage for 30 or more years. Mr. Lewis did not expand upon these assertions to explain how they would result in significant nonradiological environmental consequences. In any event, the more than 20 years of experience with storing spent fuel demonstrates that storage of spent fuel for 30 years or more does not require unprecedented institutional guarantees or raise unique questions regarding finances, economics or the security of extended spent fuel storage. Further, the Commission will require all reactor licensees, 5 years before expiration of their operating license, to provide a plan for managing the spent fuel prior to disposal. Moreover, the record documents referred to by UNWMG-EEI, DOE and AIF show that there are no significant nonradiological environmental impacts associated with the extended storage of spent fuels. The amount of heat given off by spent fuel decreases with time as the fuel ages and decays radioactively. No additional land needs to be devoted to storage facilities because reactor sites have adequate space for additional spent fuel pools or dry storage installations. The additional energy and water needed to maintain spent fuel storage is also environmentally insignificant. No commenter has challenged these assessments of environmental impacts and the Commission has no reason to question their validity. Under these circumstances, the Commission has no reason to reassess its prior determination that extended storage of spent fuel will present no significant nonradiological consequences which could adversely affect the environment.

3. Commission Compliance with NEPA

Several participants challenged the Commission's compliance with NEPA. The States of New York ("New York") and Wisconsin contend that since its inception, this proceeding has focused on the availability and safety of spent fuel storage, and has been conducted outside the scope of NEPA. New York supports this contention with the following quote from the F st Prehearing Conference Order (February 1, 1980) (unpublished):

> This rulemaking proceeding does not involve a major federal action having a significant impact on the environment, and consequently an environmental impact statement is not required by NEPA . . . .

New York asserts that this statement caused the participants not to consider NEPA in their filings. Accordingly, New York believes that the Commission cannot now transform the Waste Confidence Proceeding into a NEPA proceeding. In New York's view, joined by the Natural Resources Defense Council, Inc. ("NRDC"), NEPA required the Commission to prepare an environmental impact statement ("EIS") or environmental assessment to consider the environmental impacts of spent fuel storage at reactor sites beyond the expiration dates of reactor licenses. The Utility Nuclear Waste Management Group-Edison Electric Institute ("UNWMG-EEI") believes that it has been clear from the outset of this proceeding that the Commission intended to develop environmental regulations appropriate to the issues considered here. UNWMG-EEI cites several factors in support of its position: (1) this proceeding was the direct outgrowth of a NEPA case, *Minnesota v. NRC*, 602 F.2d 412 (D.C. Cir. 1979); (2) the Notice of Proposed Rulemaking explicitly stated a Commission intent to deal with environmental aspects of spent fuel storage; (3) the proceeding was docketed under Part 51, the Commission's regulations implementing NEPA; (4) the Commission stated that it would draw on the record of the rulemaking on environmental impact of the nuclear fuel cycle (Table S-3) and included in the NRC Data Bank for this proceeding sources of information on the environmental impacts of spent fuel storage; and (5) several participants included in their statements information pertaining to the environmental impacts of spent fuel storage.

The Commission believes that from the very beginning of this proceeding, participants were on notice that environmental aspects of spent fuel storage were under consideration. The notice initiating this proceeding stated, in pertinent part:

> If the Commission finds reasonable assurance that safe, offsite disposal for radioactive wastes from licensed facilities will be available prior to expiration of the facilities' licenses, it will promulgate a final rule providing that the *environmental and safety implications of continued onsite storage after the termination of licenses* need not be considered in individual licensing proceedings. In the event the Commission determines that onsite storage after license expiration may be necessary or appropriate, it will issue a proposed rule providing *how that question will be addressed.*
>
> * * *
>
> Based on the material received in this proceeding and on any other relevant information properly available to it, the Commission will publish a proposed or final rule in the *Federal Register*. Any such final rule will be effective thirty days after publication.

44 Fed. Reg. 61,372, 61,373-74 (1979). (Emphasis supplied.)

It is clear from this notice that if the Commission found that onsite storage after termination of reactor operating licenses would be necessary or appropriate, then it would propose a rule for dealing with the question of environmental and safety implications of continued onsite storage. New York's reference to the statement in the First Prehearing Conference Order is inapposite. That statement addressed the issue of whether a decision in this proceeding would be a proposal for major federal action having significant impact on the environment so as to require an EIS. The Presiding Officer found that the decision itself would not require an EIS. His decision in no way implied a change in the scope of the proceeding as announced in the notice initiating it.

There is also nothing about the Commission's fourth finding which requires an EIS. Neither New York nor NRDC has explained how this finding is a major Federal action having a significant impact on the human environment. The finding provides a basis for a rule that provides that environmental impacts from extended storage of spent fuel are so insignificant as not to be required to be included in an impact statement. The validity of such a rule depends on the procedures used to promulgate it and the record supporting it. An EIS is not required because such a rule itself has no environmental impacts, significant or otherwise.[8] To require an EIS here would be essentially to require an EIS to show that no EIS is required. Clearly such a result would be incorrect. Accordingly, the Commission finds that NEPA does not require an EIS to support the fourth finding.

4. List of Respondents

RESPONDENTS TO THE COMMISSION'S NOVEMBER 3, 1983, ORDER (48 FED. REG. 50,746) TO REOPEN THE PERIOD FOR LIMITED COMMENT ON THE ENVIRONMENTAL ASPECTS OF THE COMMISSION'S FOURTH FINDING IN THE WASTE CONFIDENCE PROCEEDING

1.	Attorney General of the State of New York	(N.Y.)
2.	Marvin Lewis	(Lewis)
3.	Sierra Club Radioactive Waste Campaign	(Sierra)
4.	Scientists and Engineers for Secure Energy, Inc.	(SE2)

[8] *See*, for example, *Natural Resources Defense Council, Inc. v. NRC*, 547 F.2d 633, 653 n.57 (D.C. Cir. 1976), *rev'd on other grounds, sub nom. Vermont Yankee Nuclear Power Corp. v. NRC*, 435 U.S. 519 (1978).

5. Safe Haven, Ltd. (S.H.)
6. American Institute of Chemical Engineers (AICE)
7. Atomic Industrial Forum, Inc. (AIF)
8. Utility Nuclear Waste Management Group-Edison Electric Institute (UNWMG-EEI)
9. Natural Resources Defense Council, Inc. (NRDC)
10. Attorney General of the State of Wisconsin (Wis.)
11. U.S. Department of Energy (DOE)
12. American Nuclear Society (ANS)
13. Attorney General of the State of Minnesota (Minn.)

APPENDIX

RATIONALE FOR COMMISSION FINDINGS IN THE MATTER OF THE WASTE CONFIDENCE PROCEEDING

Table of Contents

1.0 INTRODUCTION

The rationale for the five Commission findings resulting from the Waste Confidence proceeding is summarized below. This rationale is based principally on the record of the proceeding which includes participants' position statements, cross-statements, prehearing and oral statements (in the discussion below, the participants are identified by the citations defined in the Reference Notation at the end of this document). The Commission also relied on the provisions of the Nuclear Waste Policy Act of 1982 (NWPA), and other substantive material not originally included in the record relating to the discussion of the safety of dry storage of spent nuclear fuel in the Commission's Fourth Finding; the NWPA and the dry storage material have now been incorporated into the record along with the relevant comments of participants in this proceeding.

The Commission notes that two relevant developments have occurred subsequent to the closing of the record in the Waste Confidence proceeding. They are the publication of DOE's draft Mission Plan for the Civilian Radioactive Waste Management Program (April 1984) and the Commission's concurrence in DOE's General Guidelines for Recommendation of Sites for Nuclear Waste Repositories (July 3, 1984). These developments are a matter of public record, and in the case of the Commission's concurrence was the conclusion of a separate public proceeding. The Commission has considered the effects of these developments on its previously announced decision in this proceeding and determined that these developments do not substantially modify the Commission's previous conclusions.

2.0 RATIONALE FOR COMMISSION FINDINGS

2.1 First Commission Finding

The Commission finds reasonable assurance that safe disposal of radioactive waste and spent fuel in a mined geologic repository is technically feasible.

The Commission finds that safe disposal of high-level radioactive waste and spent fuel is technically possible and that it is achievable using existing technology. Although a repository has not yet been constructed and its safety and environmental acceptability demonstrated, no fundamental breakthrough in science or technology is needed to implement a successful waste disposal program. Those participants who questioned the availability of a repository did not contend that fundamental scientific breakthroughs were required, but questioned whether technical problems could be resolved in a timely manner. The record supports the conclusion that the safe disposal of high-level radioactive waste and spent nuclear fuel from licensed facilities can be accomplished.

The Department of Energy's (DOE) position is that disposal in mined geologic repositories can meet the goal of providing safe and effective isolation of radionuclides from the environment (DOE PHS at 2, 4; Tr. at 11). A number of participants stated that waste containment and isolation from the biosphere are scientifically feasible (USGS PS at 4; NRDC PS at 9; UNWMG-EEI PS, Doc. I at 22, Doc. II at II-6; Consolidated Industry Group Tr. at 16; Consolidated States Group Tr. at 98). This view is consistent with the conclusions of the *Report to the American Physical Society by the Study Group on Nuclear Fuel Cycles and Waste Management* (50 Rev. Mod. Phys. (No. 1, Pt. II), S6 (January 1980)) and the "Report to the President of the Interagency Review Group on Nuclear Waste Management" 38 (Final Report, March 1979).

The conclusion that safe radioactive waste disposal is technically feasible is based on consideration of the basic features of repository design and the problems to be solved in developing the final design. A mined geologic repository for disposal of high-level radioactive waste, as developed during the past three decades, will be based on application of the multi-barrier approach for isolation of radionuclides. The high-level radioactive waste or spent fuel is to be contained in a sealed package and any leakage from the package is to be retarded from migrating to the biosphere by engineered barriers. These engineered barriers include backfilling and sealing of the drifts and shafts of the mined repository. We believe that the isolation capability and long-term stability of the geologic setting provide a final barrier to migration to the biosphere.

The selection of a suitable geologic setting is one of the key technical problems which DOE must solve. Other problems include development of waste packages that can contain the waste until the fission product hazard is greatly reduced and engineered barriers that can effectively retard migration of radionuclides out of the repository. The Commission recognizes that these three problems are not the only ones which DOE's program must solve, but they are critical components of the multi-barrier approach for nuclear waste isolation. Much of the discussion in this proceeding has focused on these problems. We have reviewed each of these issues and have concluded that they do not present an insoluble problem which will prevent safe disposal of radioactive waste and spent fuel.

A. The Identification of Acceptable Sites

There is general agreement among the participants that the period during which the wastes must be isolated from the biosphere is at least several millenia and that such prolonged isolation can be achieved in a deep mined repository provided the geologic setting is suitable. The geologic setting is the "final" isolating barrier. If the waste package and engineered barriers fail to perform as expected, the geologic barrier must prevent harmful quantities of radioactive materials from entering the human environment.

The Commission believes that technically acceptable sites exist and can be identified. In many locations in the continental United States there are geologic media potentially suitable for a waste repository. These media occur in large, relatively homogeneous and unfaulted formations and have properties (e.g., mechanical strength, thermal stability, impermeability to water) which qualify them as potential host rocks for radioactive wastes. The potential host rocks include those being investigated by DOE — that is, domed salt, bedded salt, tuff, basalt, granite, and shale (DOE PS at II-70 to II-80). Thousands of square miles of the United States are underlain with formations containing extensive masses of such potential host rocks. Moreover, more than one-half of the United States is underlain with rock that has been stable against significant deformation and disruption for over 10 million years. The potential sites being investigated by DOE are in regions of relative tectonic stability (USGS PS at 19, 23, 24, 25, 26, 28; Tr. at 236).

Host rock suitability and formation stability are not the only relevant technical factors to be considered in repository site selection. Geohydrologic conditions — particularly the absence of significant groundwater flow from the repository to the biosphere — must be favorable for effective isolation of the wastes (USGS PS at 11). DOE's investigations reveal that the hydrologic characteristics of a major portion of the sites underlain with stable formations of potential host rock appear to be suitable for repository location (Tr. at 236; DOE PS at II-77).

These general conclusions about the extent of potential repository sites are based on the results of DOE's site exploration program (DOE PS, Appendix B) and the extensive body of earth-sciences information available at the United States Geological Survey — the Federal agency principally concerned with earth-sciences issues and, under a DOE-USGS Memorandum of Understanding, a primary source of geologic, hydrologic and mineral resource data for the National Waste Terminal Storage program (USGS PS at 2 and Appendix A; DOE PS at III-44).

DOE's site exploration efforts are focused on four host rocks (domed salt, bedded salt, basalt, and tuff) in six regions (Gulf Interior, Paradox Basin, Permian Basin, Salina Basin, DOE Hanford Site, DOE Nevada Test Site). (DOE PS, Appendix B). Although investigations of granite sites in the U.S. have been limited, DOE is developing data on the potential of granite as a host rock in collaboration with foreign investigators. A Swedish-American cooperative program (DOE's Lawrence Berkeley Laboratory is the U.S. principal in the program) has involved a series of *in situ* tests in a granite formation conducted at the Stripa mine in Sweden. The investigations included determinations of thermally induced stresses and deformations in the granite rock mass. Another cooperative study at Studsvik in Sweden involved experiments in nuclide migration in fractured subsurface crystalline rocks (DOE PS at II-258).

Some participants objected to the fact that most of DOE's site exploration involved federally owned or controlled areas, arguing that this would result in ignoring sites that were technically better (NRDC PS at 17; Tr. at 206). This objection, apparently based on the assumption that

Federal lands investigated were limited in area and geologic diversity, is not supported by the record. The Federal lands being investigated by DOE are extensive and geologically diverse; moreover, they are more readily accessible to DOE and some of them, such as the Nevada Test Site, have been previously subjected to extensive geologic assessment. These latter factors are significant advantages (DOE PS, Appendix B; UNWMG-EEI CS at IV.B-4). Although, as the United States Geological Survey pointed out, there may be advantages from a purely earth-science viewpoint in examining all parts of the country for their potential as repositories, time and resource limitations require that site exploration efforts be concentrated in limited regions fairly early so that detailed site-specific characterization efforts can be undertaken in a timely way (USGS PS at 17).

A specific site has not yet been identified as technically acceptable, and investigations of potential sites have shown some to be unsuitable. This does not necessarily mean that DOE's site-selection program will be unsuccessful in identifying technically acceptable sites. The elimination of some sites is to be expected in a pursuit of the site-selection program and is not, as some participants implied, an indication that suitable sites cannot ultimately be found.

Although the record of this proceeding does not show that DOE has progressed far enough in site characterization to confirm the existence of an acceptable site, the record does indicate that DOE's site characterization and selection program is technically sound. The data obtained in each stage of the screening process are analyzed and compared against criteria that must be satisfied for adequate performance of the total isolation system. DOE's program is providing information on site characteristics at a sufficiently large number and variety of sites and geologic media to support the expectation that one or more technically acceptable sites will be identified (DOE PS at III-8 to III-24; CS at II-140). As discussed above, DOE's site-screening efforts have concentrated on a diverse set of potentially suitable geologic media and are directed to an examination of large areas of the country on both federally owned and nonfederal lands (USGS PS at 17).

The technology for site identification is particularly well advanced (UNWMG-EEI PS at III.A-1). The record describes numerous site characterization techniques, both remote sensing and *in situ*, which are being used to evaluate sites (DOE PS at II-84 to II-103). The location and demonstration of acceptability of repository sites are problems which can be solved by the investigative and analytical methods now available (AEG PS at 1). Site-selection criteria are being refined (DOE PS at II-80 to II-83; 48 Fed. Reg. 5671 (1983)) and the technology exists for site characterization (DOE PS at II-84 to II-103). Areas have been found where most natural geologic and hydrologic processes operate at rates favorable to long-term containment in a mined repository (DOE PS at II-128; Consolidated Industry Group PHS at 9).

The Commission recognizes that there are gaps in the current state of knowledge about potential repository sites and geologic media, and about geochemical processes which affect radionuclide migration (e.g., CEC PS at 17, 54; NRDC PS at 18, 50, 64; NY at 38, 80; USGS CS at 5, 6). The gaps include a lack of a detailed understanding of such relevant processes as sorption of radionuclide-bearing molecules by the geologic media, leaching of the wastes by groundwater, and radionuclide migration through subsurface formations. Some participants contend that these gaps and uncertainties in knowledge make it difficult to predict on the basis of any effort less than a detailed onsite investigation whether a candidate repository site will be technically suitable (e.g., NRDC PS at 18, 50, 53; ECNP PS at 3, 4; NECNP PS at 20, 21, 22).

The Commission recognizes that detailed site characterization is necessary to confirm that a proposed site is indeed suitable. The Commission does not believe, however, that all uncertainties must be resolved as a precondition to repository development. The performance of a repository may be bounded by using conservative values for controlling parameters, such as waste form solubility, groundwater travel time and retardation of radionuclides. Furthermore, bounding analyses can be useful to take residual gaps in knowledge and uncertainties into account. If it can be established that a repository can perform its isolation function using established, conservative values for the controlling parameters, then it is not necessary to resolve uncertainties in the range of values these parameters may exhibit (DOE CS at II-[illegible], II-84, II-130, III-9, III-12).

The statements of those participants who are pessimistic about timely accomplishment of disposal tend to assign equal importance to all areas of uncertainty. Hence, they contain few attempts to assess the consequences of gaps in knowledge or to project the benefits of expected results from ongoing research and development efforts. It is the Commission's belief that the waste isolation system elements are adequately understood so that major unforeseen surprises in results of research and development are highly unlikely. This view is supported by USGS (USGS CS at 1-2).

A further concern of some participants is that, even if DOE were to identify a potentially acceptable repository site, the *in-situ* testing required to determine acceptability would breach the integrity of the candidate site (NY PS at 59, 63-65). If, for example, boreholes essential to characterize a potential site result in penetration of aquifers which are not amenable to effective sealing, this might make the site unacceptable (DOE PS at II-161 to II-164). However, no persuasive evidence was presented in the record to support the position that *in-situ* tests for site characterization work are likely to compromise the integrity of candidate sites. The Commission believes that *in-situ* tests can be successfully accomplished without adversely affecting site integrity for the following reasons. Many nondestructive, remote-sensing methods are available for determining site characteristics. Further, boreholes can be located in shafts or pillars of the future repository to minimize the possibility of leakage through them.

As discussed later, borehole sealing methods are expected to be adequate. The number of boreholes necessary to adequately characterize a site can be minimized by careful planning and by use of remote-sensing methods in conjunction with the drilling program (DOE PS at II-84 to II-103, II-181). Finally, the Commission believes that if a site is found to be sufficiently sensitive to the testing program that its integrity would be destroyed, then that site would necessarily be found unacceptable.

In summary, the Commission believes that technically acceptable sites for disposal of radioactive waste and spent fuel exist and can be found. There are a number of suitable host rock types to select from; many areas are underlain with massive, stable formations containing these host rocks; the areas being investigated by DOE contain such rock formations; and the uncertainties in knowledge of the earth and material sciences relevant to the identification of an acceptable repository site are not fundamental uncertainties that would prevent the identification of technically acceptable sites. Further, *in-situ* testing required to characterize a candidate site would not necessarily compromise its integrity.

B. The Development of Effective Waste Packages

1. Waste Package Considerations

An important technical aspect of safe waste disposal is to assure that the waste form and the balance of the waste package, including the primary container and ancillary enclosures, are capable of containing the radioactivity for a time sufficient for the hazard from fission-product activity to be significantly reduced (e.g., DOE PS at II-8). Decay heat, groundwater and nuclear radiation could cause the waste package components to interact with each other or with the host rock materials in such a way as to degrade the ability of the package to contain the radionuclides. These items are discussed below.

To assure long-term containment, DOE's conceptual design of a waste package is based on a defense-in-depth approach and involves a number of components including spent fuel, stabilizer (or filler), waste canister, overpack, and an emplacement hole sleeve. The stabilizer is intended to improve heat transfer from the spent fuel, to provide mechanical resistance to possible canister collapse caused by lithostatic pressure, and to act as a corrosion-resistant barrier between the spent fuel and the canister. Selection of canister overpack and emplacement hole sleeve materials will be based on tests of their chemical and physical integrity at various temperatures and levels of radiation and under various conditions of groundwater chemistry, as well as tests of their compatibility with each other and with the host rock materials under repository conditions. The canister, overpack, and sleeve should constitute relatively impermeable elements of the waste package. A variety of candidate materials is being considered for these elements. The various waste package components are to be combined in a conservative design that will compensate for the overall technical uncertainties in containment capability. The requirement for retrievability during some specified period after emplacement places conditions (e.g., ruggedness) on waste package design which are added factors to be considered in its development (DOE PS at II-129 to II-152, II-282).

It is apparent from the foregoing that the development of an effective waste package depends on obtaining engineering data on those materials that appear to be promising candidates for package components. DOE is studying over twenty-eight candidate materials for canisters and overpack (DOE PS at II-143). The DOE evaluation program indicates that many of these materials are promising. For example, iron alloys have demonstrated long-term durability (DOE PS at II-144, Ref. 383), and titanium alloys and nickel alloys show high resistance to corrosion (DOE PS at II-144, Refs. 315, 338, 342). Ceramics are resistant to chemical degradation and have many other desirable properties (DOE PS at II-145, Refs. 337, 347, 348 and 349). Preliminary analysis indicates that mild steel canisters with an appropriate backfill material would be a feasible waste package for either a salt or hard rock repository. For more demanding requirements, such as brine applications, the alloys of titanium, zirconium or nickel appear to represent alternate choices (DOE PS at II-150, Refs. 337, 382). The DOE program also includes experimental studies of the release of radioisotopes from spent fuel exposed to simulated repository conditions (e.g., salt brine and fresh water with varying dissolved oxygen content). The studies are being conducted under temperature and pressure conditions that bound and exceed repository conditions (DOE PS at II-139 to II-141).

Not all participants were optimistic about waste package development. One participant asserted that in spite of DOE's efforts to develop a package that would remain inert and stable under repository conditions, none had yet been found and the DOE program would not succeed in finding one (NRDC PS at 46). Other participants pointed to the limits of present knowledge, particularly about the leaching of radioisotopes from spent fuel in a groundwater environment, and concluded that it is not possible to select a waste form which will prevent radioisotopes from migrating to the biosphere (e.g., CEC PS at 51). They also pointed out that chemical and physical properties of spent fuel varied widely and depended on burnup, location within the reactor core, age, and physical integrity; design of a system of barriers to accommodate this heterogeneity within the context of a given geohydrologic environment would be a major undertaking (NY PS at 83).

The Commission recognizes the difficulties which must be overcome in developing a suitable waste package. A large body of experimental data must be accumulated and applied to a variety of candidate arrangements of waste package components. Suitably conservative assumptions must be postulated to define the repository conditions. Data from experiments of relatively short duration have to be used to predict behavior for much longer periods. It is common practice in materials research to perform short-duration experiments under physical or chemical conditions much more severe than those expected for the longer duration and, from known fundamental properties of the materials under investigation, to extrapolate the experimental data to predict long-term behavior. Conservatism can usually be assured by making the experimental conditions sufficiently severe.

The complex composition of the mixture of radionuclides in fission products and their basic chemical properties are known and have been the subject of investigation for more than three decades. The large body of published data on fission product chemistry and experience with fission product mixtures should provide considerable support for predicting the behavior of spent fuel and high-level radioactive waste in waste package designs.[1] The Commission, therefore, concludes that the chemical and physical properties of spent nuclear fuel and high-level radioactive waste can be sufficiently understood to permit the design of a suitable waste package.

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[1] Published compilations of such data, although not specifically included in the record of this proceeding, are well known to the nuclear science and engineering community. Examples are the three volumes of the National Nuclear Energy Series, C.D. Coryell and N. Sugarman, "Radiological Studies: The Fission Products," McGraw-Hill (1951); "Fuel Reprocessing," in *Reactor Handbook*, S.M. Stoller and R.B. Richards, Eds. (Interscience Publishers, Inc., New York, 1961), Vol. II, 2d ed.

The Commission also concludes that the DOE program is capable of developing a suitable waste package which can be disposed of in a mined geologic repository. This conclusion is based upon the large number of candidate materials being considered by DOE, the detailed evaluation of these materials to be conducted as part of the DOE program and the results of DOE's preliminary analysis of candidate materials, as described above (*see* § 2.1-B.1). The Commission's conclusion that the development of a suitable waste package is technically feasible is also consistent with other material in the record. For example, a study sponsored by the National Academy of Sciences (NAS) concluded that no insurmountable technical obstacles were foreseen to preclude safe disposal of nuclear wastes in geologic formations (UNWMG-EEI PS, Doc. 2, at II-6). The United States Geological Survey stated that a long-lived canister is within the capability of materials science technology to be achieved in the same time frame as repository site identification, qualification and development (USGS PS at 11). The National Research Council, after reviewing the Swedish waste disposal work (DOE PS at II-335, Ref. 380), concluded that the Swedish waste package could contain the radionuclides in spent fuel rods for hundreds of thousands of years (DOE CS at II-98).

2. *Effect of Reprocessing on Waste Form and Waste Package*

The waste form itself (spent fuel or other high-level waste) serves as the first barrier to radionuclide release and thus supplements the containment capability of the other components of the waste package as well as the repository's natural isolation capability. Throughout this proceeding it has been assumed that the waste form would be spent fuel discharged from light water reactors, with mechanical disassembly for volume reduction and packaging in a canister as the only potential modifications. The relevant properties of the spent fuel (irradiated uranium dioxide pellets and zircaloy cladding) are known. DOE's program has been directed toward providing data to determine the behavior of spent fuel as a waste package component under repository conditions. In its Position Statement DOE stated that the "representative case" to be considered in this proceeding is the disposal and storage of spent fuel from commercial reactors and that this does not foreclose "other approaches, such as the reprocessing of spent fuel and solidification of resultant nuclear wastes" (DOE PS at I-2).

On August 27, 1981, the Natural Resources Defense Council filed a Motion for Judgment requesting a prompt ruling that, on the basis of the present record, there is not reasonable assurance that offsite storage or disposal will be available by the year 2007-09. NRDC stated that, because the present Administration[2] had changed Federal policy towards commercial reprocessing of spent fuel (reprocessing was deferred "indefinitely" in April 1977 by the previous Administration), the disposal of spent fuel would be contrary to the present Administration's policy, and thus spent fuel was no longer a valid "reference waste form" for this proceeding. As a consequence, according to NRDC, DOE schedules and timetables, which were based on spent fuel storage and disposal, were irrelevant. The NRDC view was challenged by DOE as well as by seven participants representing utilities and the nuclear industry. The Commission took note of the NRDC filings and the responsive filings by other participants, considering them part of the record, and in its November 6, 1981 Second Prehearing Memorandum and Order asked the participants to address the significance of commercial reprocessing to the Commission's decision in the waste confidence proceeding. In response, the participants addressed this change in government policy in their prehearing statements filed in December 1981.

In response to those who argued that the change of reprocessing policy invalidated DOE's position, DOE stated that the program for development of the technology is not dependent on the waste form. Moreover, DOE pointed out that the purpose of this proceeding — "to determine whether there is at least one safe method of disposal or storage for high-level radioactive waste" is not changed by this Administration's support of reprocessing of spent fuel (DOE PHS at 2-3). Some participants who agreed with DOE commented that spent fuel disposal involves greater difficulty than disposal of solidified reprocessing waste because of its higher radioactivity and less easily handled form; in addition, they asserted that the removal of the uranium and most actinides by reprocessing would ease the requirements for safe long-term storage and simplify the waste disposal problem (UNWMG-EEI PHS at 16; SE2 PHS at 4). Others contended that spent fuel is a more difficult waste form because heat dissipation and packaging problems involved in disposal appear to be more severe than in disposal of solidified reprocessing waste (AIF PHS at 6; ANS PHS at 5).

The Commission recognizes that the proceeding has been primarily concerned with storage and disposal of spent fuel. However, the Commission does not believe that the possibility of future reprocessing, and the potential need to dispose of high-level radioactive waste resulting from reprocessing, significantly alters the technical feasibility or the

[2] The NRDC statement was based on DOE testimony before a congressional committee. The President's Nuclear Policy Statement of October 8, 1981, confirmed the DOE testimony.

schedule for developing a mined geologic repository and the design of its multiple barriers.

With regard to technical feasibility, the effect of spent fuel reprocessing on the commercial radioactive waste disposal problem is not a new consideration. The disposal of waste from reprocessing spent fuel has been studied for a longer time than the disposal of spent fuel. Until 1977, the commercial waste management program was directed primarily toward disposal of waste from spent fuel reprocessing, and those efforts have continued. A variety of waste forms has been studied (DOE PS at II-153 to II-160). Thus, considerable information is already available on the technical feasibility of developing a suitable waste form for reprocessed high-level radioactive waste. In fact, there is evidence that the disposal of reprocessed high-level waste may pose fewer technical challenges than the disposal of spent fuel (Tr. at 29). Moreover, commercial reprocessing of spent fuel cannot be undertaken in this country in the absence of a full NRC licensing review. That review will consider, among other things, the waste form to be produced by the reprocessing method and its implications for waste disposal. Unless the Commission determines that commercial reprocessing and management of its products assure adequate protection to the public health and safety and the common defense and security, spent fuel will continue to be the predominant commercial waste form available for disposal in a repository.

With regard to the impact on DOE's repository schedule, the Commission recognizes that DOE's waste package development program will eventually be affected to some extent by the nature of the waste form under development. However, the direction taken in research and evaluation of materials being conducted in the DOE program is expected to produce results which would be relevant to the waste package design, regardless of which waste form is used (DOE PS at II-141 to II-152, CS at II-96 to II-100). Moreover, the choice of waste form will not significantly affect other elements of the DOE repository program. The storage and disposal of reprocessed waste would involve substantially the same problems as those being addressed for spent fuel, and a change in waste form would not alter the site-selection program or the program for development of suitable engineered barriers (DOE PHS at 3). Thus, DOE's program is proceeding on a basis that would permit the disposal of either high-level waste or spent fuel. This approach is consistent with the recommendations of the Interagency Review Group in its March 1979 report to the President (IRG Final Report at 73) and with the direction in the Nuclear Waste Policy Act of 1982 (§ 111(a)(2)). Finally, as noted above, any decision to permit the commercial reprocessing of spent fuel will include consideration of the reprocessed waste form and its implications for waste disposal. For these reasons, the Commission concludes that the possibility of commercial reprocessing does not substantially alter the technical feasibility of, or the schedule for, developing a suitable waste package.

The Commission concludes that the basic knowledge of spent fuel and high-level waste and its behavior in a repository environment, together with DOE's ongoing development and testing program, are sufficient to provide assurance that a waste package can be developed that will provide adequate containment until the potential hazard from the fission product activity is sufficiently reduced.

C. The Development of Effective Engineered Barriers for Isolating Wastes from the Biosphere

1. Backfill Materials

In DOE's conceptual design, one engineered barrier consists of backfill materials for filling voids between canister, overpack, sleeve and host rock. The materials are chosen to retard radionuclide migration. The task is to design and test barrier materials which will be effective for very long periods of time. Candidate materials include bentonite, zeolites, iron, calcium or magnesium oxide, tachyhydrite, anhydrite, apatite, peat, gypsum, alumina, carbon, calcium chloride, crushed host rock, and others (DOE PS at II-147). Host rock or other materials would also be used to backfill drifts and shafts within the repository.

The California Department of Conservation (CDC) contends that repository shaft and borehole backfill material performance may be degraded as a result of increased temperature and other factors (CDC PS at 19-22). However, the expected temperature rise in the shaft backfill material will be only about 10°F, and will cause no significant degradation of the shaft backfill material (DOE PS at II-347, Ref. 527, NUREG/CR-0495). Other participants believe that there is inadequate information to permit development of long-lived engineered barriers that will effectively contain high-level radioactive wastes (NRDC PS at 18, 32; Ill PS at 3-4; NECNP PS at 18). CDC further contends that at this time, no information appears to have been developed that specifies the best type of backfill material to be used in particular geologic media (CDC PS at 19-22). However, the choice of backfill must take into account the rock media at the selected site as well as the waste package material. Thus, the backfill cannot be selected until a repository site has been selected. The NWTS program has as its objective, providing information on a practical range of options for backfill materials. Although a considerable amount of work remains to be done, an active research and development program on backfill materials is under way (DOE PS at II-147). Further, that program is providing information to evaluate the backfill material options, as well as to establish a basis for selection of a suitable material for the geologic media being considered. The Commission believes that this approach provides an adequate basis for concluding that effective backfill materials will be identified in a timely fashion.

In the National Waste Terminal Storage program, a wide range of candidate backfill materials has been and is continuing to be evaluated (DOE PS at II-129 to II-152). The DOE studies include measurements of the appropriate properties of backfill material including nuclide sorption capacities, capability to prevent or delay groundwater flow, thermal conductivity, mechanical strength, swelling, plastic flow and methods of backfill emplacement. Data on available candidate materials show significant radionuclide sorption capabilities, and sorptive properties can be maintained at elevated temperature and in the presence of radiation (DOE CS at II-98, II-99). Analyses indicate that several of the materials could provide adequate performance characteristics (DOE PS, Part II, Refs. 339, 340, 346, 372, 374, 376). As an example of the development of effective engineered barriers, the results of Swedish studies on radionuclide release in a repository were cited. The studies showed that a bentonite clay backfill, in conjunction with a thick copper canister (with spent fuel inside) could prevent the release of radionuclides to the host rock in the presence of granitic groundwater for thousands to hundreds of thousands of years. In the Swedish experiments, the clay barrier provided sorptive properties which were predicted to delay the breakthrough of various radionuclides for thousands of years and also served to chemically condition the groundwater, reducing its corrosive effect on the canister (DOE PS at II-145, II-148). The use of certain clays to retard the transport of radionuclides released by the waste package is applicable to repository designs here in this country. While DOE has not proposed using thick copper canisters as employed in the Swedish studies, this example of a durable combination of waste package and backfill material, which was demonstrated to be effective in isolating radionuclides for very long times, indicates that the basic approach is reasonable. The use of clays, combined with other appropriate materials, could provide an effective means for radionuclide retardation and corrosion control.

In sum, the Commission believes that DOE's ongoing developmental studies reported in this proceeding (DOE PS at II-129 to II-152) are technically sound and provide a basis for reasonable assurance that engineered barriers can be developed to isolate or retard radioactive material released by the waste package.

2. Borehole and Shaft Sealants

A major factor in repository performance is the effective sealing of boreholes and shafts during repository closure operations. All penetrations provide potential pathways for radionuclides to reach the biosphere or for groundwater to enter the repository. The penetrations must be sealed for an extended period of time. Further, the geology and hydrology at a particular site, as well as the expected temperature and pressure conditions during repository lifetime, must be understood in order to make a proper choice of the borehole and shaft sealing materials and to develop effective borehole and shaft seals.

Some participants concluded that current information concerning the technology for the sealing of the boreholes and shafts is inadequate. They also questioned the capability of the DOE program to develop sufficient information to allow effective seal design (CDC PS at 19-22; NRDC PS at 5). The views of several participants who expressed concern about sealing were reflected in the comments of CDC. The Commission's response to each of the points raised by CDC on borehole and shaft sealing issues is discussed below.

CDC indicated that since long-term effects of heat and radiation on seal materials were not a factor in past oil and gas borehole sealing experience, such experience is not applicable to repository sealing.[3] However, at distances of more than several feet from waste canisters emplaced in a repository, radiation exposures are small and the temperature rise at seals in the shafts and boreholes is insignificant for sealing purposes (DOE CS at II-108).

CDC also believes that the tests of cement seals with epoxy resins in bedded salt deposits discussed by DOE are insufficient to provide assurance of seal stability over a period of 10,000 years, especially when the effects of higher temperature and radiation are not included. As noted above, temperature and radiation effects on seals are expected to be negligible.

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[3] The Commission notes that the extensive oil and gas borehole sealing experience has not been concerned with very-long-term sealing. Therefore, DOE's sealing research and development must provide a basis to extend that experience for the development of long-term seals for a repository.

While these tests may not provide conclusive proof of performance for 10,000 years, they are expected to provide useful information for seal development.

CDC states that the results of field tests described by DOE as continuing over the next few years will not be completed in time to contribute to seal design criteria which are to be completed[4] in 1982. However, the final seal design for the selected site is scheduled for 2 years after a site is selected (DOE PS at II-184). Testing up to that date is expected to be useful in designing an effective seal.

CDC questioned whether tests of waste package system component interactions with the surrounding media in bedded salt described by DOE will be completed in time for location of a repository. However, the Commission finds no basis for this assertion in the record. The DOE program appears to be adequately addressing this issue. Studies are in progress to characterize further the interactions between candidate backfill-getter materials and waste container alloys. These studies include investigations of dry rock salt/metal interactions and high-intensity radiation/salt/brine/metal interactions. (DOE PS at II-149, II-150).

CDC asserts that DOE has not discussed designing backfill material and penetration seals to allow for safe reentry if retrieval should become necessary. However, the provision to retrieve high-level waste and spent fuel for a number of years after the repository is filled has been addressed by DOE (DOE PS at II-280 to II-283). Although it has not yet been established whether backfilling and sealing will be conducted before repository closure, these operations may be reserved until a final decision for closure is made. In any event, CDC provides no basis for concluding that providing for retrievability will necessarily create any major difficulties for the design of backfill material and penetration seals.

According to one participant,

> [t]here is no established way to seal a repository so as to prevent radionuclide release to the biosphere for the necessary period of time. DOE has termed the sealing problem a "key unknown" but there is no consensus that the technology which is currently anticipated will provide adequate seals for even a few decades.

(Consolidated States Group PHS at 8). Other participants maintained that seals must perform as well as the host rock in preventing radionuclide migration (NRDC PS at 55). The DOE position is that the seal should provide a barrier with sufficient integrity to ensure acceptable

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[4] DOE has published "Schematic Designs for Penetration Seals for a Reference Repository in Bedded Salt," ONWI-405, November 1982.

consequences, and sealing adequacy should be determined only on a site-specific basis (DOE CS at II-106). DOE asserted that its program will successfully resolve remaining uncertainties in repository sealing technology (DOE CS at II-106 to II-109).

DOE has been studying cement-based borehole plugging and has examined use of grout materials for application to the Waste Isolation Pilot Plant (WIPP) and other potential repository sites. Earth-melting technology for plugging in salt and use of compacted natural earth materials are also being investigated (DOE PS at II-183, CS at 106-09). There is a considerable body of experience in sealing subsurface formations in the oil, gas, and other mineral-extraction industries. However, related industrial experience and requirements for sealing a repository differ in one important respect: repository sealing must be effective for a very long time while most other sealing applications are for relatively short time periods (DOE PS at II-182). Future DOE effort will be needed to verify borehole seal performance and durability for each candidate medium. An important aspect of DOE's work is to determine the rate of degradation of seal performance as a function of time. DOE plans to determine seal performance specifications for a particular site on the basis of calculated predictions of radionuclide release and transport to the accessible environment (DOE PS at II-182). These predictions are expected to indicate that a site whose characteristics for waste isolation are clearly superior may not require sealing performance specifications as stringent as those for a less-favorable site.

Based upon the extensive experience with shaft and borehole sealing in other industries and DOE's detailed program for evaluating the long-term performance of seals, the Commission believes that there is a reasonable basis to expect that long-term effective borehole and shaft seals can be developed.

D. *Summary of Views on the Technical Feasibility of Safe Waste Disposal*

The Commission notes that participants in the Waste Confidence Rulemaking proceeding have generally agreed there are no known fundamental technical problems which would make safe waste disposal impossible. Where they differ is the extent to which the technical problems of disposal technology and siting have already been solved and the capability of DOE to solve them, and particularly to solve them by 2007-09 or by the expiration date of reactor operating licenses (e.g., NY PS at 3; NECNP PS at 171; Minn PS, Enclosure at 13-20).

The Commission believes that the record provides a basis for reasonable assurance that the key technical problems can be solved. Technically acceptable sites exist and can be found among the various types of geologic media and locations under investigation by DOE. Currently developed geophysical methods for site evaluation appear capable of adequately characterizing the site, and the residual uncertainties in earth sciences data do not seem to be an insurmountable impediment. Further, the Commission believes that the multi-barrier approach to waste package design is sound and that package development is being adequately addressed by DOE. DOE's development work on backfill materials and sealants provides a reasonable basis to expect that backfill materials and long-term seals can be developed. Reprocessing of spent fuel would only become a licensed commercial activity if disposal of reprocessing waste in a mined repository would be established as technically feasible. While the Commission recognizes that more engineering development and site-specific work on disposal technology will have to be conducted before a waste repository can be constructed and operated, the Commission concludes that it is technically feasible to safely dispose of high-level radioactive waste and spent fuel in a mined geologic repository.

2.2 Second Commission Finding

The Commission finds reasonable assurance that one or more mined geologic repositories for commercial high-level radioactive waste and spent fuel will be available by the years 2007-09, and that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of commercial high-level radioactive waste and spent fuel originating in such reactor and generated up to that time.

While the record of the proceeding supports a finding that disposal is technically achievable, the Federal government has, in the past, made inadequate progress in developing sound waste management policies and programs. The Commission notes that DOE has stated in its April 1984 draft Mission Plan that the first repository will begin operations in 1998, and that the second will start up in 2004. However, it is recognized that both technical and institutional issues contribute to uncertainties concerning DOE's ability to complete one or more mined geologic repositories for high-level radioactive waste by those dates. The technical issues concern DOE's ability to find technically acceptable sites in a timely fashion and the timely development of waste forms, packages, and engineered barriers. The institutional issues concern primarily Federal-State relations and the management and funding of the Federal program.

The Commission has considered the effect of enactment of the Nuclear Waste Policy Act of 1982 and concludes that the Act helps to reduce these scheduling and institutional concerns. The Act provides support for timely resolution of technical uncertainties by (1) establishing specific milestones for all the key tasks; (2) coordinating the activities of all the involved Federal agencies; (3) providing for firm schedules, and a mission plan for the accomplishment of the tasks; and (4) providing a mechanism for monitoring progress, for identifying failures to meet the schedules and the milestones, and for adjusting the future elements of the program in the event that such failures occur. In order to further enhance the resolution of technical uncertainties regarding rock thermal-geomechanics the Act provides for the establishment of a Test and Evaluation facility to carry out *in-situ* studies of rock at repository depth. The Act also reduces uncertainties in the institutional arrangements for the participation of affected States in the siting and development of repositories and in the long-term management, direction and funding of the repository program. The Commission's assessment of both the technical and institutional factors is discussed below.

A. *Technical Uncertainties*

The ability to construct and operate a mined geologic repository that will provide for the safe disposal of high-level radioactive waste and spent fuel by the years 2007-09 has been challenged by several participants. In addition to the institutional issues which must be resolved, interrelated technical problems have to be solved in a coordinated and timely fashion. The Department of Energy is confident the technical problems can be solved as scheduled in the National Waste Terminal Storage Program plans (DOE PS at III-86, CS at III-13; DOE draft Mission Plan, April 1984). Other participants conclude that because of unresolved technical problems, DOE's schedule cannot be met (e.g., Consolidated Public Interest Group PHS at 2-7; Consolidated State Group PHS at 1-13). For convenience, we consider the technical controversy in two categories: (a) finding technically acceptable sites in a timely fashion, and (b) the timely development of waste packages and engineered barriers.

1. Finding Technically Acceptable Sites in a Timely Fashion

To assure the adequacy of a candidate site requires extensive onsite investigations including drilling or excavating, as well as analyses and technical evaluations. Although DOE has not yet begun subsurface site characterization to enable identification of an acceptable site, the record does indicate that DOE's site screening and selection program is providing information on site characteristics at a sufficiently large number and variety of sites and geologic media to support the expectation that one or more technically acceptable sites will be identified.

DOE is investigating four geologic media at a number of sites: domed salt (Gulf Interior Region); bedded salt (Paradox Basin, Permian Basin, Salina Basin); basalt (DOE's Hanford Site), and volcanic tuff (DOE's Nevada Test Site). Investigations in a fifth media (granite) are planned, but sites have not yet been determined (DOE PS, Appendix B). Exploratory shaft excavation at three sites in different geologic media was to begin for basalt in April 1983, for volcanic tuff in October 1983, and for salt in December 1983 (Tr. at 241-42). However, the Nuclear Waste Policy Act of 1982 (NWPA) imposed new conditions which made it necessary to revise this schedule. The NWPA specified that DOE had to prepare environmental assessments for each of five nominated sites, from which three sites would be recommended to the President for characterization. DOE's preparation of environmental assessments and recommendation of three sites were to be accomplished in keeping with the provisions of the repository siting guidelines required by the NWPA. The Commission's concurrence in DOE's siting guidelines on July 3, 1984, enables DOE to proceed to nominate and recommend repository sites for characterization. DOE has recently published a revised schedule for site-selection milestones in its April 1984 draft Mission Plan. As described in its Mission Plan, the current status of DOE's site-selection schedule calls for the issuance of environmental assessments for five nominated sites and the recommendation of three of those sites for characterization by December 1984. DOE's schedule for work in the various geologic media is summarized below.

Salt: Resolution of the identified key screening issues in FY 84 is expected to permit nomination of a candidate salt dome site in December 1984. DOE is still choosing from among several salt domes in the Gulf Coast interior region (Tr. at 243-44; DOE draft Mission Plan, April 1984). For bedded salt, primary effort has been focused on the Palo Duro Basin in Texas, the Paradox Basin in Utah, and the Permian Basin, particularly the Delaware Basin in the Los Medanos area, the site considered for the proposed WIPP. The Bureau of Land Management issued the report "Environmental Assessment of DOE Proposed Location and Baseline Studies in the Paradox Basin, Utah-Final" UT-060-51-2-11, in July 1982. Each of the seven potentially acceptable salt sites has been evaluated for environmental conditions, and a site characterization plan is expected to be issued for salt in September 1985. DOE will start land access and permitting activities for salt after negotiating agreements with affected States and Indian tribes (DOE draft Mission Plan, April 1984).

Basalt: The basalt formations at the Hanford Reservation in the center of the Pasco Basin (Columbia Plateau, central Washington) are prime candidates for repository sites. DOE expects to issue a site characterization plan for basalt in January 1985 and start drilling for the exploratory shaft in March 1985 (DOE draft Mission Plan, April 1984).

Volcanic Tuff: The Nevada Test Site offers several suitable candidates for waste repository siting. The primary focus is welded tuff on Yucca Mountain, where DOE has begun a program of drilling and geophysical evaluation. DOE expects to issue a site characterization plan for tuff in March 1985 and begin shaft work in September 1985 (DOE draft Mission Plan, April 1984).

Granite: Granite and other crystalline rock media are being considered for the second repository (DOE draft Mission Plan, April 1984). DOE has conducted only limited investigations of granite at the Nevada Test Site (DOE PS at B-66, B-72), but is developing data on the potential of granite as a repository medium in collaboration with Swedish investigators (DOE PS at II-258). This project has already produced a large amount of rock thermal-mechanics data at repository depth for use in repository designs in granite media in this county (DOE PS at II-258 to II-260).

As indicated in our discussion of technical feasibility, the identification of technically acceptable sites is a key problem and the date of successful solution of this problem is a critical milestone in the repository program. Those participants who believe DOE could not meet its site-selection schedule asserted that determination of the acceptability of proposed repository sites requires information that will not be available when needed. They maintained that DOE's knowledge is seriously incomplete with respect to all of the potential sites considered to date. Further, they asserted that because new information could disqualify any of the potential sites, as it did at the Palestine dome, there is, as yet, no basis for reasonable assurance that an acceptable repository site will be available in the time period under consideration (NRDC PS at 44; NECNP PS at 24). The Commission recognizes that if the DOE program were further along, e.g., in the middle of exploratory shaft work, there would be much more site-specific information available (including the results of *in-situ* tests) and a firmer basis for assessing whether DOE's revised schedule can be met. However, the Commission can make a reasonable prediction with the information now before it.

Underlying the pessimism of some participants is apparently a belief that DOE's past record in solving technical problems undermines the possibility of finding confidence in DOE's ability to solve the waste disposal problems in a timely way. The Commission acknowledges that in the past the waste programs of DOE and its predecessor organizations have experienced difficulty in making timely progress toward a solution of the nuclear waste problem. However, the Commission need not rely on this past record in making its confidence determination. The DOE program is now adequately addressing the issues yet to be resolved in identifying an acceptable site, and DOE's schedule is a reasonable one (*see* the discussion in § 2.2-B.4, below). The qualifications and professional experience of the many scientists and engineers on the overview committees and peer review groups who advise and consult on the DOE program should provide confidence in DOE's efforts (DOE CS, Appendix D). The support of the USGS in the earth sciences field (USGS PS, Appendix A) clearly contributes to confidence that the technical problems associated with identifying an acceptable repository site will be solved. As noted before, no fundamental technical breakthroughs are necessary. Rather, completing the program is a matter of step-by-step evaluation and development based on ongoing site studies and research programs.

The Commission believes that the enactment of the Nuclear Waste Policy Act of 1982 provides impetus to that program and helps ensure that it will be completed on a schedule consistent with the Commission's findings. The Nuclear Waste Policy Act establishes a detailed step-by-step plan for developing a waste repository. The Act directs DOE to prepare a comprehensive Mission Plan which will establish programmatic milestones for research, development, technology demonstration and systems integration. The Act also requires the various Federal agencies involved in the program to coordinate their activities. Involved agencies must report their progress, or lack thereof, to Congress, explain any slip in schedule and set a new schedule for activities. Thus, the Act provides a framework and schedule for developing a repository.

The schedule set forth in the Act calls for the identification of adequate sites in time to meet the final decision date on construction authorization by the NRC and well before the time at which such action would be necessary to assure repository operation within the time period discussed in this decision. The time between sinking of an exploratory shaft and the completion of site characterization contemplated by the Act (§§ 112, 114) is 26 months, with an extension to 38 months under certain conditions; the DOE schedule for these activities is generally compatible with this schedule (*see* § 2.2-B.4, below).

The Nuclear Waste Policy Act also puts in place procedures (§§ 115, 116, 117, 118, 119) which the Commission believes will help to resolve potential institutional problems that might affect the schedule for site selection. These are discussed in detail hereafter. The Commission believes that the provisions of the Act should also provide resources (§§ 302, 303) to adequately fund the site selection and characterization work.

Given all of these considerations, the Commission concludes that there is reasonable assurance that technical uncertainties — unsolved technical problems and information gaps — will be removed in time for DOE to meet its proposed schedule. DOE's program is adequate and its schedule is reasonable. The Act provides a greater degree of confidence than existed previously that site selection will proceed within the general time frame that DOE has described in its position statement.

2. Timely Development of Waste Packages and Engineered Barriers

Some participants have expressed strong reservations concerning DOE's ability to develop waste forms, packages, and engineered barriers in a timely fashion. The DOE technical effort to solve problems was characterized as only just being defined in many significant areas, including the prevention of corrosion of waste canisters (NRDC PS at 18). Other participants contended that: the design and evaluation studies of penetration seals and backfill material might not be completed soon enough to meet the goal of achieving an operational repository by 1997 to 2006; the long-term effects of heat and radiation on the integrity of the seal materials are not known; tests of cement seals with epoxy resin in bedded salt deposits are insufficient to assure stability of such seals over a period of 10,000 years; and field tests of liquid permeability during a period of 3 months cannot provide confidence concerning the stability of seals during a period of 10,000 years. Participants also contended that no information had yet been provided which specified the type of backfill material most suitable for specific geological media and capable of withstanding thermal stress (CDC PS at 19-22).

Although technical problems associated with the development of waste packages and engineered barriers could delay DOE's schedule, DOE believes that the uncertainties surrounding the waste package would be resolved or bounded as a result of implementation of its program (DOE PS at II-160, CS at II-96). The DOE Waste Package Program Plan (ONWI-96) which was issued in August 1980, updated in June 1981 (NWTS-96) and updated further in DOE's April 1984 draft Mission Plan, sets forth details of DOE's program. Waste package performance criteria will be developed in the near future. Final action on the criteria will be contingent upon the final issuance of NRC's technical criteria (10 C.F.R. Part 60, Subpart E), the publication of the relevant regulatory guides on waste packages, and the ONWI-33 series of criteria documents, i.e., the reports DOE/NWTS-33(1), (2), (3), "NWTS Program Criteria for Mined Geologic Disposal of Nuclear Wastes."

Earlier, DOE had planned to complete the waste package preliminary designs for salt in September 1982, for basalt in June 1985, for tuff in June 1984, for granite in September 1984, and for argillaceous rock in December 1984, and to establish a baseline for waste form specifications by June 1983 (ONWI-96). According to DOE's April 1984 draft Mission Plan, the current reference canister material for basalt is carbon steel. Alternative materials include an iron-chromium-molybdenum alloy, copper and a copper-nickel alloy. On the basis of preliminary corrosion test results, carbon steel has also been selected as the reference canister material for salt. The titanium alloy Tricode 12 has been designated as an alternative material. Type 304L stainless steel has been identified as the reference container material for tuff; other austenitic stainless steels, Inconel and copper are alternatives. Waste-package conceptual designs have been developed for basalt, salt and tuff. (The conceptual design for tuff is based on saturated conditions; a conceptual design for the unsaturated zone will be available in late FY 84 (DOE draft Mission Plan, April 1984)).

Tests with spent fuel and borosilicate glass have been initiated under site-specific conditions for basalt, salt and tuff. Preliminary waste acceptance requirements have been developed for basalt and salt. In addition, for salt media, interim waste-acceptance requirements for borosilicate glass and draft waste acceptance requirements for spent fuel were prepared in FY 83. Preliminary requirements for tuff will be prepared in FY 84. DOE intends to submit the baseline waste form specifications developed during the conceptual design studies for acceptance by NRC. The specifications will be subjected to configuration control for application throughout the waste processing and disposal program.

According to the DOE draft Mission Plan the complete waste package performance model will be verified and validated by September 1989. Further, the program plan calls for completion of the waste package final design that takes into account the selected site environmental conditions, after completion of *in-situ* testing in FY 89 and FY 90. Packing material is included in the reference waste package only for basalt. The reference packing material for basalt is a mixture of crushed basalt and sodium-bentonite clay. Ongoing physical property testing of reference packing material is expected to be completed in FY 87 and ongoing radionuclide sorption, solubility and diffusion testing are to be completed by September 1989.

Some participants' statements are pessimistic assessments based on the fact that the DOE program has not yet reached the critical milestones — e.g., establishment of waste form specifications, completion of waste package preliminary designs, verification of a waste package performance model, and qualification of barrier materials. However, the Commission believes that these technical problems will be solved without delaying a repository schedule. DOE has put in place an extensive nuclear waste research program that addresses each of these technical problems. Research results already reported on waste form packaging and barrier materials indicate that these research efforts, although not yet completed, can reasonably be expected to provide solutions to those problems when those solutions are needed to meet the DOE schedule (DOE PS at II-129 to II-197, CS at II-93 to II-100).

The Commission's positive assessment is strengthened by provisions in the Nuclear Waste Policy Act of 1982. Title II of the Act authorizes DOE to undertake steps leading to the construction, operation and maintenance of a deep geologic test and evaluation facility and to establish a focused and integrated research, development and demonstration program. In the area of waste package design, the Act directs that DOE's Mission Plan identify a process for solidifying high-level radioactive waste or packaging spent fuel with an analysis of the data to support selection of the solidification process or packaging technique. The Act calls for a schedule for implementing such a plan and for an aggressive research and development program to provide a high-integrity disposal package at a reasonable price (§ 301(a)(8)). The Commission notes that DOE's published draft Mission Plan (April 1984) addresses these issues in detail. Congressional authorization of those programs, together with the assurance of necessary funding, provides the Commission additional confidence that the required research work will be done in a timely manner.

The Commission also notes that the programs to solve the major technical problems relating to the timely development of waste forms, waste packages, and engineered barriers can proceed in parallel. Because the waste repository must be designed as a system, the problems are interrelated; however, the relationships are such that solving one problem need not await the solution of another. DOE could proceed for a number of years on waste package development before making a decision on the form of the waste, without affecting the repository availability schedule.

B. Institutional Uncertainties

The principal institutional issues that affect the schedule for availability of a mined geologic repository include: measures for dealing with Federal-State disputes; an assured funding mechanism that will be sufficient over time to cover the period for developing a repository; an organizational capability for managing the high-level waste program, whether this be DOE or a successor organization; and a firm schedule and establishment of responsibilities which will lead to repository development in a reasonable period of time. Each of these is discussed in turn.

1. Measures for Dealing with Federal-State-Local Concerns

The President and Congress have recognized the need to involve State and local governments in the decisionmaking process and have taken steps, including enactment of the Nuclear Waste Policy Act of 1982, to establish an institutional framework to accomplish this end. DOE pointed out that Presidents Carter and Reagan have considered State involvement in site selection an important aspect of the high-level radioactive waste disposal program. President Carter, in his message to Congress, directed "the Secretary of Energy to provide financial and technical assistance to States and other jurisdictions to facilitate the full participation of State and local government in review and licensing proceedings." He committed the Federal government to work with State, tribal and local governments in the siting of high-level waste repositories. Within a framework of "consultation and concurrence," a host State would have a continuing role in Federal decisionmaking involving the siting, design and construction of a high-level waste repository (DOE CS at II-11, II-13 to II-14). President Reagan's statement of October 8, 1981, similarly instructed DOE to work closely with industry and State governments in developing methods of storing and disposing of commercial high-level waste.

Although industry groups believed that DOE had made substantial progress in cooperating with State and local authorities by encouraging their direct participation in planning and preliminary site-selection activities (UNWMG-EEI CS at V-27, V-28), States and environmental groups were skeptical that the mechanisms proposed by DOE for incorporating State and local views (e.g., consultation and concurrence) would work satisfactorily. Many States asserted a lack of confidence in DOE's claims that it would be able to gain agreement from States by persuasive measures (e.g., Ohio PS at 5; NY PS at 74; Wis PS, Kelly, at 5) and noted that information-sharing was inadequate to reduce or overcome a State's resistance to a repository (e.g., NY PS at 74; NRDC PS at 69). The States also believed that DOE had underestimated potential State and local opposition to the siting of a repository (CEC PS at 27, Ohio PS at 12) and that consultation and concurrence must include a mechanism for resolving intergovernmental disputes (Vt PS at 3). Other participants argued that many States had already imposed bans on waste disposal (NECNP PS at 32) and that DOE had presented no means for resolving State nonconcurrence (NRDC PS at 69). Still others claimed that the State's role in the site-selection process must be specifically defined (Del PS at 6); but that DOE had provided no basis for optimism that this could be done (NECNP PS at 69). Some participants suggested that local opposition to waste repositories could be overcome by providing financial compensation to nearby communities (AIChE PS at 6) but that DOE had not adequately considered compensation to host communities for socioeconomic impacts (Ohio PS at 14).

The recently enacted Nuclear Waste Policy Act of 1982 defines the roles of the States and Indian tribes in repository site selection, and thereby reduces some of the uncertainties in settling disputes between the Federal government and affected States and Indian tribes. By providing for information exchange, for financial and technical assistance, and for processes of consultation, cooperation, negotiation and binding written agreement, the Act should help to minimize the potential for more formal objections and confrontations.

Specifically, the Act requires DOE to identify the States with one or more potentially acceptable sites for a repository and to notify the governing bodies of the affected States or Indian tribes of those sites (§ 116(a)). The Act establishes detailed procedures for consultation with the States and Indian tribes regarding repository site selection (§ 117). DOE, NRC and other agencies involved in the construction, operation, or regulation of any aspect of a repository in a State must provide to the State and to any affected Indian tribe, timely and complete information regarding plans made with respect to the site characterization, development, design, licensing, construction, operation, regulation, or decommissioning of such a repository (§ 117(a)(1)). If DOE fails to provide such information requested by the State or affected Indian tribe in a timely manner, it must cease operations at the site (§ 117(a)(2)). The Act also provides that DOE must consult and cooperate (§ 117(b)) with the affected States and Indian tribes and must enter into a binding written agreement (§ 117(c)) setting forth the procedures under which information transfer, consultation and cooperation is to be conducted.

Following consultation with affected States and Indian tribes, the Secretary of Energy is to recommend to the President three sites suitable for characterization as candidates for selection as the first and second repositories (by July 1, 1985, and July 1, 1989, respectively) (§ 112(b)(B), (C)). The President must then submit to Congress his recommendation of sites qualified for construction authorization for a first and second repository (no later than March 31, 1987, and March 31, 1990, respectively) (§ 114(a)(2)(A)). Following submission by the President of a recommended site to Congress, the Governor or legislature of the State, or the Indian tribe in which such site is located, may disapprove the site designation and submit (within 60 days) a notice of disapproval to Congress (§ 116(b)(2)). The site is disapproved unless Congress passes a joint resolution within 90 days to override the State or Indian tribe disapproval (§ 115(c)). The Commission recognizes that the latter provision may create uncertainty in gaining the needed approvals of repository sites from the affected States or Indian tribes. Nevertheless, the Commission believes that, on balance, this congressional action to establish a detailed process for State and tribal involvement in the development of repositories will reduce overall uncertainties by encouraging Federal-State cooperation and by limiting the potential for formal State or Indian tribe objections that could lead to disruption of project plans and schedules. This conclusion is consistent with the views expressed by State participants in this proceeding that a mechanism for State participation, including the resolution of State objections and nonconcurrences, is necessary for State cooperation and for progress in repository development (Tr. at 117, 119, 120). Further, the Act fixes the point in time at which a State may raise formal objections. Once that time has passed, this should reduce uncertainties at later stages.

The Act stipulates that DOE will reimburse costs incurred by affected States and Indian tribes in participating in the activities identified above. The Act provides that the Secretary of Energy shall make financial grants (§§ 116, 118) to each State or affected Indian tribe notified by DOE that a potentially acceptable repository site exists within its jurisdiction. These grants are made to enable the State or affected Indian tribe to participate in the review and approval activities required by the Act (§§ 116, 117), or authorized by written agreement entered into with DOE. Further, DOE is to make financial grants (§§ 116, 118) to each State or affected Indian tribe where a candidate site for a repository is approved, to enable the State or Indian tribe to conduct the following activities: (a) review activities taken for purposes of determining impacts of such a repository, (b) develop a request for impact assistance, (c) engage in site monitoring, testing or evaluation, (d) provide information to its residents, and (e) request information. In addition, the Act specifies that financial assistance will be provided to mitigate any economic, social, public health and safety, or environmental impacts of the development of a repository. The Act also provides that State and local government units shall receive payments equal to the amount they would receive from taxing such site characterization and repository development activities in the same manner that they tax other real property and industrial activities (§ 116). By providing a tangible benefit to those localities or Indian reservations where repository sites are being investigated, this provision should address one concern frequently expressed by State and tribal organizations, and may result in a more willing acceptance of a repository site.

In sum, the Commission believes that the provisions of the Nuclear Waste Policy Act of 1982 reduce uncertainties regarding the role of affected States and Indian tribes in repository site selection and evaluation, and minimize the potential for direct confrontation between the Federal government and the States or tribal organizations with respect to the disposal of commercial high-level waste and spent fuel. By reducing these uncertainties, the Act should help minimize the potential that differences between the Federal government and States or Indian tribes will substantially disrupt or delay the repository program. Further, as discussed previously in this section, the decisionmaking process set up by the Act provides a detailed, step-by-step approach which builds in regulatory involvement. This should also provide confidence to States and Indian tribes that the program will proceed on a technically sound and acceptable basis.

2. Continuity of the Management of the Waste Program

The Commission recognizes that the waste disposal program involves activities conducted over a period of decades. Thus, there is a need for long-term stability of management and organization. The Commission's Second Prehearing Memorandum and Order of November 6, 1981, sought comments on the implications of the possible dismantling of the DOE and assignment of its functions to other Federal agencies. In response, DOE stated:

> The ability of the Federal Government to implement the waste isolation program would not be affected by the President's September 24, 1981 proposal to dismantle DOE. As demonstrated by his Nuclear Policy Statement of October 8, 1981 . . . the President is committed to the swift deployment of means of storing and disposing of commercial high-level nuclear waste. Thus, some governmental unit will continue the program aggressively if DOE is dismantled.

(DOE PHS at 8). The DOE statement was amplified by the Deputy Secretary of Energy in the oral presentations on January 11, 1982:

> [A]s far as the reorganization is concerned, the plan is not, I think, to do away with the activities of the Department of Energy. The plan, as it has been announced so far, is to in fact merge the activities, in particular, these activities into the Department of Commerce. And we do not visualize at this time any significant changes in the way in which the programs relating to waste management would be altered, either technically or from a management point of view.

(Tr. at 13).

The nuclear industry participants agreed with DOE's view on this question (Consolidated Industry Group PHS at 18; AIF PHS at 7; SE2 PHS at 6; ANS PHS at 8; UG at 2). However, State participants and intervenor groups disputed the DOE view. They saw the potential dismantlement of DOE as leading to further delay in resolution of the radioactive waste disposal problem and asserted that DOE's possible abolition made representations regarding the future success of its waste program useless (Consolidated State Group PHS at 2, 9; Minn PHS at 6-8).

The Commission does not believe that the Administration's proposal to transfer the activities of the Department of Energy to the Department of Commerce introduces substantial new uncertainties regarding the continuity of Federal management of the nuclear waste program. As the Department of Energy stated, the Administration's proposal, if adopted, would simply transfer the nuclear waste program functions from one Federal agency to another. Moreover, congressional action is needed to adopt the Administration's proposal. Yet, in the 3 years since the Administration's proposal to dismantle DOE was made, there has been no discernible action by the Congress to proceed with adoption of the proposal. Because the Congress has not taken action toward adoption of the Administration's proposal, and because the proposal, even if adopted, would consist of only a transfer of the program from one agency to another, the Commission does not believe that the Administration's proposal constitutes a significant source of management uncertainty for the nuclear waste program.

The Commission believes that residual uncertainties regarding the continuity of Federal management of the nuclear waste program have also been reduced by the Nuclear Waste Policy Act of 1982. The Act provides for the establishment of an Office of Civilian Radioactive Waste Management within the Department of Energy. This Office is to be headed by a Director appointed by the President, with Senate confirmation, who will report directly to the Secretary of Energy (§ 304).

Further, the Act raises the activities of this Office to a high level of visibility and accountability by stipulating that an annual comprehensive report of the activities and expenditures of the Office will be submitted to Congress and that an annual audit of the Office will be conducted by the Comptroller General, who will report the results to Congress. The Act also requires two additional elements that provide added assurance of continuity: a "Mission Plan" and a schedule of activities for DOE. The Mission Plan is a detailed and comprehensive report which is intended to provide "an informational basis sufficient to permit informed decisions to be made in carrying out the repository program and the research, development, and demonstration programs required under this Act." The Secretary of Energy has already submitted a draft Mission Plan to the States, the affected Indian tribes, the Commission and appropriate government agencies for their comments; after revising the plan, DOE must submit it to the appropriate congressional committees (§ 301(a) and (b)). The schedule of DOE's activities in conducting this program was discussed in § 2.2-A.1, above. Taken together, the provisions of the Nuclear Waste Policy Act establish a detailed management framework for the conduct of the repository program that should help ensure both sound management and continuity — whether the responsibility for the repository program is retained in DOE or is transferred to another Federal agency.

3. Continued Funding of the Nuclear Waste Management Program

There is general agreement among all participants that the program to develop a mined geologic repository for nuclear wastes will require more than a decade of effort at a total cost of several billion dollars. A steady source of funding will be needed to assure the timely success of the program. DOE pointed out that it would request an adequate level of funding for the National Waste Terminal Storage (NWTS) Program as stated in the Department's Position Statement (DOE CS at II-30). In addition, DOE stated that Congress' commitment to the commercial waste disposal program was demonstrated by the continuous increase in the level of funding since 1976. The funding level was increased by more than a factor of 10 between 1976 and 1980 (DOE CS at II-30). Some participants disagreed with DOE's optimism concerning the future availability of funds and pointed out that competing priorities for Federal funds could deprive DOE of the necessary resources (CDC PS at 7; Lewis PS at 9; NRDC PS at 28; Tr. at 203).

Congress passed a continuing resolution for FY 83 funding of DOE's nuclear waste program at the level of $259.4 million. This is about $10 million more than DOE's earlier FY 83 request of $249 million. Additionally, the Nuclear Waste Policy Act authorizes the Secretary of Energy to enter into contracts and collect a fee of 1 mill per kilowatt-hour of electricity generated by nuclear reactors in return for the Federal government's acceptance of title, subsequent transportation, and disposal of high-level radioactive waste or spent fuel (§ 302(a)(2)). In order to be able to use a Federal repository, the Act required the generator or owner of such waste or spent fuel to enter into a contract by June 30, 1983, or the date on which generation is commenced or title is taken, whichever occurs later (§ 302(b)(2)). The Commission must require the negotiation of such contracts as a precondition to the issuance or renewal of a license (§ 302(b)(1)(B)). The Commission notes that all such contracts have been executed. DOE testified in the January 11, 1982, hearing that it expected the funds collected under such a program would allow support of the DOE waste program at an initial level of $185 million. Under the program subsequently adopted by the Congress, these funds are to be placed into a nuclear waste fund to support DOE's repository program. The general approach prescribed by the Act is to operate DOE's nuclear waste program on a full-cost-recovery basis. In this regard, the Act provides that DOE must annually review the amount of the fees established to evaluate whether collection of the fees will provide sufficient revenues to offset the costs expected. In the event DOE determines that the revenues being collected are less than the amount needed in order to recover the costs, DOE must propose to Congress an adjustment to the fee to ensure full cost recovery. The Act also provides (§ 302(e)(5)) that, if at any time, the monies available in the Waste Fund are insufficient to support DOE's nuclear waste program, DOE will have the authority to borrow from the Treasury. The Commission believes that the long-term funding provisions of the Act should provide adequate financial support for DOE's nuclear waste program.

4. DOE's Schedule for Repository Development

The DOE reference schedule described in its April 1984 draft Mission Plan establishes the earliest date of repository availability as 1998 and delineates the logic and the period of activities that are deemed achievable under current program assumptions. While DOE acknowledges that contingency time is required in the schedule to accommodate such factors as institutional uncertainties, public hearings, or possible project reorientation, it believes that an appropriate amount of time has, in fact, been allowed in the reference schedule. Under the reference schedule, DOE expects that disposal facilities will be operational in 1998 (DOE draft Mission Plan, April 1984). DOE's updated repository development schedule specifies the critical milestones prior to commencing construction of the first repository as:

March	1985	(basalt)	Commencement of exploratory shaft
September	1985	(tuff)	work* at three sites (three different
—	—	(salt)	media: salt, basalt and tuff)**
August	1990		Submission of application for authorization to construct the first repository
August	1993		Construction authorization for the first repository

*Including borehole drilling.

**An October 1982 update of this information indicated that a pilot borehole was started in September 1982 for an exploratory shaft in tuff at the Nevada Test Site. In May 1982, DOE initiated work on surface preparation, construction of drilling pads and support buildings for the drilling operation at the BWIP basalt site. In January 1982, a borehole was begun at a point 300 feet from the BWIP planned exploratory shaft location to provide data for planning the shaft excavation. No exploratory shaft work has begun at the Paradox Basin bedded salt site. As noted in the siting discussion under the Second Commission Finding, the Nuclear Waste Policy Act of 1982 requires DOE to complete certain actions before site characterization. These include issuance of siting guidelines concurred in by NRC, preparation of environmental assessments, notification of State and affected Indian tribes where sites are located, and holding of public hearings in the vicinity of each site.

The Commission concurred in DOE's repository siting guidelines on July 3, 1984, enabling DOE to proceed to complete the other site-selection tasks. The Commission notes that DOE's draft Mission Plan (April 1984) anticipated the completion of the siting guidelines by mid-Summer 1984 and DOE revised its site-selection schedule accordingly. Final environmental assessments for five nominated sites (including salt, basalt and tuff media) are to be completed in December 1984, at which time three of the five sites will be recommended for characterization.

NRC's construction authorization (under 10 C.F.R. Part 60) would mark the end of the site-selection process.

Some participants believe that DOE cannot have a waste disposal facility available by 2007. These participants concluded that DOE's slow progress in the past suggests that DOE may be unable to solve the many problems that will arise in the future and that DOE's schedule for repository development is unduly optimistic (e.g., Minn PS at 6; Ill PS at 2; OCTLA PS at 8-9; CDC PS at 7).

One of the primary purposes of the recently enacted Nuclear Waste Policy Act of 1982 is "to establish a schedule for the siting, construction, and operation of repositories that will provide reasonable assurance that the public and the environment will be adequately protected from the hazards posed by high-level radioactive waste and such spent nuclear fuel as may be disposed of in a repository." (§ 111(b)(1)). The Commission recognizes that, if fundamental technical breakthroughs were necessary, it would not be possible for Congress to legislate their solution or specify schedules for their accomplishment. However, as discussed previously, such breakthroughs are not necessary. Rather, the remaining uncertainties are reflected in the need for step-by-step evaluation and development based on ongoing site studies and research programs. The Commission believes the Act provides means for resolution of those institutional and technical issues most likely to delay repository development, both because it provides an assured source of funding and other significant institutional arrangements, and because it provides detailed procedures for maintaining progress, coordinating activities and rectifying weaknesses. For these reasons, the Commission believes that the selection and characterization of suitable sites and the construction of repositories will be accomplished within the general time frame established by the Act, or within a few years thereafter.

The provisions of the Nuclear Waste Policy Act of 1982 that establish schedules for repository development are elaborate and allow for various contingencies. A number of steps are involved before NRC considers authorization of construction. DOE is to nominate five sites it believes suitable for site characterization for possible repository development (§ 112(b)). DOE is to recommend for site characterization three candidate sites to the President (§ 112(b)(1)(B)); the President is to recommend one of the characterized sites to the Congress (§ 114(a)(2)(A)); the affected State or Indian tribe is given an opportunity to submit a notice of disapproval to the Congress (§§ 115(b), (116)(b)(2), 118(a)); the Congress may overturn a State or Indian tribe's disapproval of the site by passing a resolution of approval (§ 115(c)); and, if Congress approves or no notice of disapproval is submitted by a State or Indian tribe, then DOE is to apply for construction authorization (§ 114(b)).

DOE's revised reference schedule (DOE draft Mission Plan, April 1984) states that the application for repository construction authorization will be submitted to the Commission in August 1990. Under the terms of the Act the Commission is expected to reach a decision within 3 years of the application date, or by August 1993 (§ 114) (under certain conditions, extension by 1 year would be permitted). If the NRC decision is favorable, the repository would be constructed and would begin operation, according to DOE's "reference schedule," in January 1998. Earlier dates can be achieved if the Presidential review time is reduced, if DOE promptly files the construction authorization application, if NRC provides a construction authorization in less than 3 years, or if DOE constructs the repository in a shorter period than provided in its estimated schedule. However, it is prudent to assume that such a contraction of the schedule will not be realized.

The Nuclear Waste Policy Act of 1982 establishes "not later than January 31, 1998" as the date when DOE is to begin disposal of high-level radioactive waste or spent fuel (§ 302(a)(5)(B)). This is consistent with the current dates of the DOE schedules discussed above and with the detailed step-by-step milestones established by the Act. The schedule established by the Act would assure the operation of the first repository well before the years 2007-09, i.e., the period of concern in the present proceeding.

Despite the delays in DOE's earlier milestones, the Commission believes that the program established by the Act is generally consistent with the schedule presented by DOE in this proceeding and that DOE's milestones are generally both realistic and achievable. Achievement of the scheduled first date of repository operation is further assured by other provisions of the Act which specify means for resolution of those institutional and technical issues most likely to delay repository completion. In addition to those provisions discussed previously, the Commission notes that the Act clarifies how the requirements of the National Environmental Policy Act are to be met (e.g., §§ 113(c), (d); 114(a), (f); 119(a); 121(c)). The Act also requires that any Federal agency determining that it cannot comply with the repository decision schedule in the Act must notify both the Secretary of Energy and Congress, explaining the reasons for its inability to meet the deadlines. The agency must also submit recommendations for mitigating the delay (§ 114(e)(2)). These provisions of the Act, as well as those that support the technical program — the provisions for research, development, and demonstration efforts regarding waste disposal (Title II of the Act), increase the prospects for having the first repository in operation not later than the first few years of the next century.

The Commission also finds reasonable assurance that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of commercial high-level radioactive waste and spent fuel generated up to that time. The Nuclear Waste Policy Act of 1982 establishes Federal responsibility and a clearly defined Federal policy for the disposal of such waste and spent fuel and creates a Nuclear Waste Fund to implement Federal policy. The Act establishes as a matter of national policy that this responsibility is a continuing one, and provides means for the Secretary of Energy to examine periodically the adequacy of resources to accomplish this end.

The Commission notes that as of September 30, 1982, the generating capacity of all commercial nuclear power plants in the U.S. with operating licenses or construction permits was 131 electrical gigawatts (GWe) and the capacity of those under construction permit review was about 5

GWe (NUREG-0871, Vol. 1, No. 4, at 2, 8). DOE, in its letter of March 27, 1981, to the Presiding Officer of this proceeding, provided an estimate of 180 GWe for the capacity of operating LWRs in the year 2000. This value is significantly lower than the value (276 GWe) presented in DOE's 1980 position statement (DOE PS at V-4) and lower than that (202 GWe) presented in the NRC's Generic Environmental Impact Statement on spent fuel handling and storage (NUREG-0575, Vol. 1, at 2-4). The validity of the latter predictions has been affected by the cancellations of a number of proposed units during the past 2 years. The DOE 1981 estimate of 180 GWe in the year 2000 appears to be a reasonable estimate of the likely installed capacity at that time. On this basis, during the 40 years of operation of each plant, using as a realistic assumption a 60% capacity factor, the electrical energy generation would be about 4300 GWe-years. Assuming 38 metric tons of heavy metal (MTHM) are discharged for each gigawatt-year (IRG Final Report at D-6; NUREG-0575, Vol. 1, at 2-4) the total discharged spent fuel from these plants would likely be about 160,000 metric tons. The capacity of each proposed repository will depend on such factors as the thermal loading limit in waste emplacement, space limitations within the host rock, nuclear power generation capacity in the region to be serviced by the repository, and economy of scale considerations (DOE PS at III-70 to III-79; IRG Final Report at D-21). In its cross-statement, DOE's estimate that three to six repositories might be needed was based on the assumption that nuclear power generation capacity grows to 250 GWe by the year 2000 and remains at that level until 2040 (DOE CS at II-53). The representative characteristics of each repository used by DOE were 2000 acres and a 40- to 100-kW/acre loading, corresponding to a repository capacity of about 70,000 to 170,000 metric tons of uranium, respectively (DOE PS at III-76). Reflecting the reduction in nuclear power projections, DOE estimated in the January 1982 hearing that the ultimate reactor capacity would be about 200 GWe (Tr. at 236). DOE then assumed a repository capacity of 100,000 metric tons and concluded that "between two and three" repositories would be needed (Tr. at 237). To accommodate the 160,000 metric tons we have assumed, two repositories, each with 100,000-metric-ton capacity, would appear to be sufficient.

Repository completion and operation at 3-year intervals would result in having adequate capacity about 3 years after initial operation of the first repository (DOE PS at III-86). As noted earlier, emplacement of spent fuel in the first repository should begin not later than the first few years of the next century. Thus, if the first repository begins to receive spent fuel in the year 2005, the second may begin operation as early as

2008, in which case all spent fuel would be emplaced by about 2026, assuming DOE's estimated receiving rates (DOE PS at III-71) and operation of each repository as completed. Because the rate of waste emplacement during the first 5 years of operation would be about 1800 metric tons per year (DOE PS at III-71), only 5400 metric tons would be emplaced in the first repository by the time the second began operation. This would satisfy the requirements of § 114(d) of the Nuclear Waste Policy Act, i.e., the prohibition of emplacement of more than 70,000 metric tons in the first licensed repository before the second repository is in operation. If the DOE estimated emplacement rates (which would increase to 6000 metric tons/year after the first 5 years) are realized, it will take about 15 years to emplace 70,000 metric tons in the first repository.

For the foregoing reasons, the Commission finds reasonable assurance that one or more mined geologic repositories for commercial high-level radioactive waste and spent fuel will be available by the years 2007-09, and that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of commercial high-level radioactive waste and spent fuel originating in such reactor and generated up to that time.

2.3 Third Commission Finding

The Commission finds reasonable assurance that high-level radioactive waste and spent fuel will be managed in a safe manner until sufficient repository capacity is available to assure the safe disposal of all high-level radioactive waste and spent fuel.

Nuclear power plants whose operating licenses expire after the years 2007-09 will be subject to NRC regulation during the entire period between their initial operation and the availability of a waste repository. The Commission has reasonable assurance that the spent fuel generated by these licensed plants will be managed by the licensees in a safe manner. Compliance with the NRC regulations and any specific license conditions that may be imposed on the licensees will assure adequate protection of the public health and safety. Regulations primarily addressing spent fuel storage include 10 C.F.R. Part 50 for storage at the reactor facility and 10 C.F.R. Part 72 for storage in independent spent fuel storage installations (ISFSI). Safety and environmental issues involving such storage are addressed in licensing reviews under both Parts 50 and 72, and continued storage operations are audited and inspected by NRC. NRC's experience in more than eighty individual evaluations of the safety of spent fuel storage shows that significant releases of radioactivity from spent fuel under licensed storage conditions are extremely remote (*see* discussion in § 2.4, below).

Some nuclear power plant operating licenses expire before the years 2007-09. For technical, economic or other reasons, other plants may choose, or be forced, to terminate operation prior to 2007-09 even though their operating licenses have not expired. For example, the existence of a safety problem for a particular plant could prevent further operation of the plant or could require plant modifications that make continued plant operation uneconomic. The licensee, upon expiration or termination of its license, may be granted (under 10 C.F.R. Part 50 or Part 72) a license to retain custody of the spent fuel for a specified term (until repository capacity is available and the spent fuel can be transferred to DOE under § 123 of the Nuclear Waste Policy Act of 1982) subject to NRC regulations and license conditions needed to assure adequate protection of the public. Alternatively, the owner of the spent fuel, as a last resort, may apply for an interim storage contract with DOE, under § 135(b) of the Act, until not later than 3 years after a repository or monitored retrievable storage facility is available for spent fuel. For the reasons discussed above, the Commission is confident that in every case the spent fuel generated by those plants will be managed safely during the period between license expiration or termination and the availability of a mined waste repository for disposal.

To assure the continuity of safe management of spent fuel, the Commission, in a separate action, is preparing an amendment to 10 C.F.R. Part 50 which would require licensees of operating nuclear power reactors to submit, no later than 5 years before expiration of the reactor operating license, written notification to the Commission, for its review and approval, of the actions which the licensee will take to manage and provide funding for the management of all irradiated fuel at the reactor site following expiration of the reactor operating license, until ultimate disposal of the spent fuel in a repository. The licensee's notification will be required to specify how the licensee will fund the financial costs of extended storage or other disposition of spent fuel. It is possible for the funding of the storage to be provided by an internal reserve fund or special assessment during that 5-year period to cover the costs of storage of the spent fuel after the expiration of the reactor operating license. The storage costs are not large relative to power generation costs. A representative figure is $1 million/year for storage of spent fuel in reactor basins beyond the operating license expiration (NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference BWR Power Station," Addendum 2, July 1983; NUREG/CR-0672, "Technology, Safety and Costs of Decommissioning a Reference PWR Power Station," Addendum 1, July 1983).

Additional assurance that the conditions necessary for safe storage will be maintained until disposal facilities are available is provided by the Commission's authority to require continued safe management of the spent fuel past the operating license expiration or termination (10 C.F.R. § 50.82). If a utility should have technical problems in continuing its commitment to maintain safe storage of its spent fuel, NRC as the cognizant regulatory agency would intervene and the utility would be required to assure safe storage. If a licensee fails financially, or otherwise must cease its operations, the cognizant State public utility commission would be likely to require an orderly transfer to another entity. The successor would take over the licensee's facilities and, provided the conditions for transfer of licenses prescribed in NRC regulations (10 C.F.R. § 50.80) were met by the succeeding entity, operation of the original licensee's facilities would be permitted to continue. Moreover, an orderly transfer to a successor organization would be mandatory to protect the substantial capital investment. Further, the Commission believes that the possibility of a need for Federal action to take over stored spent fuel from a defunct utility or from a utility that lacked technical competence to assure safe storage is remote, but the authority for such action exists (§§ 186c and 188 of the Atomic Energy Act of 1954, as amended, 42 U.S.C. §§ 2236, 2238).

Interim storage capacity may be required for plants whose operating licenses expire or are terminated before sufficient repository capacity is available. As discussed in the rationale for the fifth finding, the Nuclear Waste Policy Act of 1982 includes a number of provisions to assure the availability of interim storage capacity for spent fuel during the period before repository operation (§§ 131 through 137). Provisions are made for Federal government-supplied interim storage capacity (up to 1900 metric tons) for civilian power reactors whose owners cannot reasonably provide adequate storage capacity.

In all cases where the interim storage is at a licensee's site, safe management will be assured by compliance with NRC regulations and specific license conditions. Where DOE provides the interim storage capacity, except in the use of existing capacity at Government-owned facilities, DOE is to "comply with any applicable requirements for licensing or authorization" (§ 135(a)(4)). If existing federally owned storage facilities are used, NRC is required to determine "that such use will adequately protect the public health and safety" (§ 135(a)(1)). These provisions of the Act would assure that spent fuel will be managed in a safe manner until repository capacity is available. Facilities for reprocessing high-level waste, should any be constructed or become operational before a repository is available, would be licensed under 10 C.F.R. Part 50, and solidification and interim storage of high-level waste would be provided for at such facilities. For the foregoing reasons, the Commission finds reasonable assurance that high-level waste and spent fuel will be managed in a safe manner until sufficient repository capacity is available for its safe disposal.

2.4 Fourth Commission Finding

The Commission finds reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations.

Although the Commission has reasonable assurance that at least one mined geologic repository will be available by the years 2007-09, the Commission also realizes that for various reasons, including insufficient capacity to immediately dispose of all existing spent fuel, spent fuel may be stored in existing or new storage facilities for some periods beyond 2007-09. The Commission believes that this extended storage will not be necessary for any period longer than 30 years beyond the term of an operating license. For this reason, the Commission has addressed on a generic basis in this decision the safety and environmental impacts of extended spent fuel storage at reactor spent fuel storage basins or at either onsite or offsite spent fuel storage installations. The Commission finds that spent fuel can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of reactor operating licenses. To ensure that spent fuel which remains in storage will be managed properly until transferred to DOE for disposal, the Commission is proposing an amendment to its regulations (10 C.F.R. Part 50). The amendment will require the licensee to notify the Commission, 5 years prior to expiration of its reactor operating license, how the spent fuel will be managed until disposal.

The Commission's finding is based on the record of this proceeding which indicates that significant releases of radioactivity from spent fuel under licensed storage conditions are highly unlikely. It is also supported by the Commission's experience in conducting more than eighty individual safety evaluations of storage facilities.

The safety of prolonged spent fuel storage can be considered in terms of four major issues: (a) the long-term integrity of spent fuel under water pool storage conditions, (b) structure and component safety for extended facility operation, (c) the safety of dry storage, and (d) potential risks of accidents and acts of sabotage at spent fuel storage facilities. Each of these issues is discussed separately below, in light of the information provided by the participants in this proceeding, and NRC experience in regulating storage of spent fuel.

A. Long-Term Integrity of Spent Fuel Under Water Pool Storage Conditions

The Commission finds that the cladding which encases spent fuel is highly resistant to failure under pool storage conditions. As noted by DOE in its Position Statement, there are up to 18 years of continuous storage experience for zircaloy-clad fuel and 12 years continuous storage experience for stainless-clad fuel (DOE PS at IV-73). Corrosion studies of irradiated fuel at twenty reactor pools in the United States suggest that there is no detectable degradation of zircaloy cladding. Data from corrosion studies of spent fuel stored in Canadian pools also support this finding (A.B. Johnson, Jr., "Behavior of Spent Nuclear Fuel in Water Pool Storage" (UC-70), Battelle Pacific Northwest Laboratories, BNWL-2256 (September 1977), at 10-11, 17).

The long-term integrity of spent fuel in storage pools, which has been confirmed by observation and analysis, was cited by industry participants (e.g., Consolidated Industry Group PHS at 3-6; UNWMG-EEI PS, Doc. 4, at 8; UG at 2). No degradation has been observed in commercial power reactor fuel stored in onsite pools in the United States. Extrapolation of corrosion data suggests that only a few hundredths of a percent of clad thickness would be corroded after 100 years (A.B. Johnson, Jr., "Utility Spent Fuel Storage Experience," PNL-SA-6863, presented at the American Nuclear Society's Executive Conference on Spent Fuel Policy and its Implications, Buford, Georgia (April 2-5, 1978)). The American Nuclear Society cited a study (G. Vesterbend and T. Olsson, BNWL-TR-320, May 1978, English Translation of RB78-29), which concluded that degradation mechanisms such as general corrosion, local corrosion, stress corrosion, hydrogen embrittlement, and delayed hydrogen cracking are not expected to produce degradation to any significant extent for 50 years (ANS PS at 34).

Canadian experience, including occasional examination during 17 years of storage, has indicated no evidence of significant corrosion or other chemical degradation. Even where the uranium oxide pellets were exposed to pool water as a result of prior damage of the fuel assembly, the pellets have been inert to pool water, an observation also confirmed by laboratory studies ("Canadian Experience with Wet and Dry Storage

Concepts," presented at the American Nuclear Society's Executive Conference on Spent Fuel Policy and Its Implications, Buford, Georgia (April 2-5, 1978)). Another Canadian study concluded that "50 to 100 years under water should not significantly affect their [spent fuel bundles] integrity" (J.F. Walker, "The Long-Term Storage of Irradiated CANDU Fuel Under Water," AECL-6313, Whiteshell Nuclear Research Establishment (January 1979)). This appraisal was based on findings such as no deterioration by corrosion or mechanical damage during 16 years of storage in water, no release of fission products from the uranium dioxide matrix during 11 years of storage in water, and no fission-product-induced stress corrosion cracking anticipated during water storage at temperatures below 100°C (C.E.L. Hunt, J.C. Wood, and A.S. Bain, "Long-Term Storage of Fuel in Water," AECL-6577, Chalk River Nuclear Laboratories (June 1979)).

The ability of spent fuel to withstand extended water basin storage is also supported by metallurgical examination of Canadian zircaloy-clad fuel after 11 years of pool storage, metallurgical examination of zircaloy-clad PWR and BWR high-burnup fuel after 5 and 6 years in pool storage, and return of Canadian fuel bundles to a reactor after 10 years of pool storage. Periodic hot-cell examination of high-burnup PWR and BWR bundles over 6 years of pool storage at the WAK Fuel Reprocessing Plant in Germany has also confirmed that spent fuel maintains its integrity under pool storage conditions. Other countries having favorable experience with pool storage of zircaloy-clad spent fuel include: the United Kingdom, 13 years; Belgium, 12 years; Japan, 11 years; Norway, 11 years; West Germany, 9 years; and Sweden, 7 years (Johnson, "Utility Spent Fuel Storage Experience," *supra,* at 7). Programs of monitoring spent fuel storage are being conducted in Canada, the United Kingdom and the Federal Republic of Germany (DOE PS at IV-59 to IV-61; UNWMG-EEI PS, Doc. 4, at 23).

The only fuel failures which have occurred in spent fuel pools involved types of fuel and failure mechanisms not found at U.S. commercial reactor facilities, e.g., degradation of zircaloy-clad metallic uranium fuel from the Hanford N-Reactor as a result of cladding damage in the fuel discharge system. The system differs from the fuel discharge systems of commercial reactors. Moreover, metallic uranium fuel is not used in commercial power reactors. NRDC cited some conclusions drawn by Mr. Justice Parker regarding his lack of confidence in long-term storage of spent fuel, based on the Windscale Inquiry in Great Britain in 1978, which involved stainless-steel-clad, gas-cooled reactor fuel (NRDC PS at 92). This is not pertinent to pool storage of commercial spent fuel since the high-temperature conditions in a gas-cooled reactor which can cause sensitization of the cladding are not experienced by fuel in boiling or pressurized water reactors (Johnson, "Utility Spent Fuel Storage Experience," *supra*, at 17-18).

Some participants did not agree that there is an adequate basis for confidence in safe extended-term spent fuel storage. Although agreeing with the extent of experience cited by DOE and other participants, the Natural Resources Defense Council, for example, stressed that more experience is needed before one can be confident of safe extended storage. NRDC considered the length of storage experience cited by DOE as insufficient to establish that spent fuel can be stored safely for periods well in excess of 40 years (NRDC PS at 88-92). A similar position was taken by the State of Minnesota (Minn PHS at 8-9). NRDC referred to the problem of the long-term storage of spent fuel reported in the Windscale Inquiry Report by the Hon. Mr. Justice Parker, Vol. 1, at 29-30. However, the conclusion quoted from the report, when taken in context, refers only to irradiated fuel from AGR (advanced gas-cooled) nuclear power plants. As noted earlier, the conditions to which the fuel cladding is exposed in gas-cooled reactors differ from those in U.S. commercial light water reactors. Moreover, the cladding of AGR fuel is identified as stainless steel in the Windscale Inquiry Report. Only two commercial LWR nuclear power plants operating in the U.S. today use stainless steel clad. Most U.S. nuclear fuel is zircaloy clad, and reactor operators have not seen evidence of degradation of LWR spent fuel, either zircaloy or stainless steel clad, in storage pools (A.B. Johnson, Jr., "Spent Fuel Storage Experience," *Nuclear Technology*, Vol. 43, at 171 (Mid-April 1979)). Further, as stated earlier, cladding degradation caused by stainless steel sensitization in an AGR high-temperature environment is not pertinent to the lower-temperature environment of LWRs. Therefore, the problem of long-term storage of spent fuel reported in the Windscale Inquiry is not relevant to U.S. spent fuel.

After expiration of a reactor operating license, the fuel storage pools at the reactor site would be licensed under 10 C.F.R. Part 72. The requirements of 10 C.F.R. Part 72 provide for operation under conditions involving a careful control of pool water chemistry to minimize corrosion. The required monitoring of the pool water would provide an early warning of any problems with defective cladding, so that corrective actions may be taken. Experience indicates that, under licensed storage conditions, significant releases of radioactivity are highly unlikely. The Commission is confident that the regulations now in place will assure adequate protection of the public health and safety and the environment during the period when the spent fuel is in storage (NUREG-0575, "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel," Vol. 1, August 1979, at ES-12, 4-10 to 4-17).

Although confidence that spent fuel will maintain its integrity during storage for an additional 30 years beyond the facility's license expiration date involves an extrapolation of experience by a factor of 2 or 3 in time, the extrapolation is made for conditions in which corrosion mechanisms are well understood. Technical studies cited above support the conclusion that corrosion would have a negligible effect during several decades of extended pool storage. The Commission finds that this extrapolation is reasonable and is consistent with standard engineering practice.

B. Structure and Component Safety for Extended Facility Operation for Storage of Spent Fuel in Water Pools

Questions were raised concerning the adequacy of structural materials and components of spent fuel storage basins to function effectively during periods that are double those assumed in the base design. This concern was expressed in connection with the possible necessity for longer storage times if permanent disposal is not available by the year 2006 (Del PS at 4). The experience at the General Electric Company Morris Operation in Illinois, where a mechanical failure caused contaminated water to leak into the environment, was cited as an example of an unforeseen failure that could jeopardize the safety of spent fuel storage (NECNP PS at 65). A generic problem regarding pipe cracks in borated water systems at PWR plants was also cited as evidence of uncertainty that long-term interim storage would be safely accomplished without modification and fuel shuffling (NECNP PS at 64). The Commission notes that the latter problem was discussed in detail in the Atomic Safety and Licensing Board Notification, "Pipe Cracks in Stagnant Borated Water Systems at PWRs," dated August 14, 1979, in the ASLB consideration of a proposed licensing amendment to permit modification of a spent fuel storage pool (*Commonwealth Edison Co.* (Zion Station, Units 1 and 2), LBP-80-7, 11 NRC 245 (1980)). The Notification referred to by NECNP indicated that cracks had occurred in safety-related type-304 stainless steel piping systems which contained stagnant borated water. Apparently, the cracking was attributable to stress corrosion caused by the residual welding stresses in heat-affected zones. The NRC staff review found that such cracking was not directly related to spent fuel pool modifications, and that necessary repairs could be readily made. The staff concluded that cracks in low-pressure spent fuel cooling systems do not have safety significance.

Extensive experience with storage pool operation has demonstrated the ability of pool components to withstand the operating environment (DOE CS at II-145 to II-148). In the relatively few cases of equipment failure, pool operators have been able to repair the equipment or replace defective components promptly (UNWMG-EEI PS, Doc. 4, at 25; UG at 2). The Commission finds no reason why spent fuel storage basins would not be capable of performing their cooling and storage functions for a number of years past the design-basis period of 40 years if they are properly maintained.

As one participant pointed out, "the pool structure as well as the racks are designed to withstand extreme physical conditions set forth in NRC licensing requirements. These include seismic, hydrologic, meteorological and structural requirements" (UNWMG-EEI PS, Doc. 4, at 25; UG at 2). The design requirements are set forth in 10 C.F.R. Parts 50 and 72. The design basis siting conditions for storage pools at reactor sites are those of the reactor itself. Siting conditions are reviewed by the NRC staff, the Advisory Committee on Reactor Safeguards and the Atomic Safety and Licensing Board at the construction permit stage and then reviewed again in connection with the issuance of the facility's operating license. In issuing a power reactor operating license, the Commission is, in effect, expressing its confidence that the design basis siting conditions will not be exceeded during the 40-year license period. If pool storage facilities were used to store spent fuel after expiration of reactor operating licenses, the utilities would be able, as part of their continuing maintenance of storage facilities, to replace defective components in a timely way, if needed, so as to avoid any safety problems. Some participants (e.g., NECNP PS at 63; Minn PHS at 8-9; and Del PS at 4) do not place the same weight which the Commission does on experience at spent fuel storage facilities and on studies cited by DOE and certain others which support the argument that the structural integrity of these basins can be readily maintained (DOE CS at II-145, III-13; UNWMG-EEI PS, Doc. 4, at 19). The disagreements appear to center largely on the extent to which present experience may be relied upon as a basis for predicting the safety of spent fuel storage over a period two or three times the design period.

The degradation mechanisms involved in spent fuel pool storage are well understood. The resulting changes in fuel cladding and pool systems and components are gradual and thus provide sufficient time for the identification and development of remedial action without subjecting plant personnel or the public to significant risk. The fuel storage racks are designed to maintain their integrity for many decades; if they fail in any way, they may be replaced. There are a number of routine and radiologically safe methods for maintenance at spent fuel storage basins to ensure their continued effective performance. These include replacing racks or other components, or moving spent fuel to another storage facility. The Commission finds that the extensive operating experience with many storage pools adequately supports predictions of long-term integrity of storage basins.

The Commission concludes that the experience with spent fuel storage provides an adequate basis for confidence in the continued safe storage of spent fuel in water pools either at or away from a reactor site for at least 30 years after expiration of the plant's license.

C. *Safety of Dry Storage of Spent Fuel*

While the record of this proceeding has focussed on water pool storage, the Commission notes that dry storage of spent fuel has also been addressed to a limited extent (e.g., DOE PS at IV-12 to IV-22 and IV-63, CS at II-147, PHS at 9; UNWMG-PS, Doc. 4, at 16-17 and CS at III-6 to III-7; Tr. at 69-72). The NRC's regulation 10 C.F.R. Part 72 specifically covers dry storage of spent fuel (§ 72.2(c)), and experience with dry storage was a subject of public comment in the rulemaking (NUREG-0587, "Analysis of Comments on 10 C.F.R. Part 72," October 1980, at II-12 to II-13). NRC reports, NUREG-0575, "Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel" August 1979, and NUREG/CR-1223, "Dry Storage of Spent Nuclear Fuel, A Preliminary Survey of Existing Technology and Experience" April 1980, which have been referenced in this proceeding, examined potential environmental impacts and experience with interim dry storage of spent fuel. The GEIS (NUREG-0575, *supra*, Vol. 1, at 8-2) contained the conclusion that the use of alternative dry passive storage techniques for aged fuel, now being investigated by the Department of Energy, appears to be as feasible and environmentally acceptable as storage of spent fuel in water basins. Prior to the adoption of Part 72, dry storage of irradiated fuel had been licensed under Part 50 at the Hallam sodium graphite reactor. Dry storage is also presently licensed under Part 50 at the Ft. St. Vrain high-temperature gas reactor.

Although the number of years of experience with dry storage systems is less than that with water pool storage, the understanding of some of the material degradation processes experienced in water pool storage should be applicable to dry storage. As discussed below, dry storage involves a simpler technology than that represented by water basin storage systems.[5] Water basin storage relies upon active systems such as pumps, renewable filters, and cooling systems to maintain safe storage. Favorable water chemistry must also be maintained to retard corrosion. On the other hand, dry storage reduces reliance upon active systems and does not need water which together with impurities may corrode spent fuel cladding. With convective circulation of an inert atmosphere in a sealed dry system, there is little opportunity for corrosion.[6] For these reasons, the Commission believes that safe dry storage should be achievable without undue difficulty. New dry storage experience with light water reactor (LWR) fuel is becoming available for examination, and the evaluations discussed below suggest that the favorable results of up to almost two decades of dry storage experience with non-LWR spent fuel can also be obtained for LWR spent fuel in adequately designed dry storage installations.

A recent review of dry storage experience by Johnson, *et al.*, in "Behavior of Spent Nuclear Fuel and Storage Components in Dry Interim Storage" (*supra* note 5), provides an update of dry storage activities, particularly with respect to zircaloy-clad spent fuel. In that report (at 18-24) the experimental data base for nonzircaloy-clad spent fuel, including stainless-steel-clad fuel and the data base for zircaloy-clad fuel are discussed. Tests conducted to verify the integrity of zircaloy cladding have not indicated any degradation in dry storage (*id.* at 27). In summary, the report states (at 44-45):

> Operating information is available from fueled dry well, silo, vault, and metal cask storage facilities. Maximum operational histories are:
>
> | | **All Fuel** | **Zircaloy-Clad Fuel** |
> |---|---|---|
> | Dry wells | up to 18 yr | up to 3 to 4 yr |
> | Vaults | up to 18 yr | up to 1 yr |
> | Silos | up to 7 yr | up to 7 yr |
> | Metal casks | – | < 1 yr |
>
> All times related to 1982.
>
> Operational history with interim storage in metal casks is minimal; however, there is extensive experience with metal shipping casks. In addition, metal storage casks have been designed and tested, and cask tests with irradiated fuel are currently

[5] *See*, for example, K. Einfeld and J. Fleisch, "Fuel Storage in the Federal Republic of Germany" and R.J. Steffen and J.B. Wright, "Westinghouse Advanced Energy Systems Division," *Proceedings of the American Nuclear Society's Topical Meeting on Options for Spent Fuel Storage*, Savannah, Georgia, September 26-29, 1982; A.B. Johnson, Jr., E.R. Gilbert, and R.J. Guenther, "Behavior of Spent Nuclear Fuel and Storage System Components in Dry Interim Storage," PNL-4189, August 1982.

[6] "Fuel Storage in the Federal Republic of Germany," *supra* note 5, at 3.

under way in the Federal Republic of Germany and are planned in Switzerland and the United States. The integrity of zircaloy-clad fuel in a given demonstration test is relevant to predicting fuel behavior in other dry storage concepts under similar conditions.

Information on experience with dry cask storage in other countries is also becoming available. Einfeld and Fleisch's paper, "Fuel Storage in the Federal Republic of Germany," *supra* note 5, discussed the results of dry storage research on spent fuel in an inert atmosphere. They note on page 3 of their report:

> Several tests have been conducted to verify the integrity of LWR spent fuel cladding in dry storage. To date none of the integrity tests has indicated that the cladding is degrading during long-term storage. Even under conditions more severe than in the casks, the fuel shows no cladding failures. From the tests listed in Table II it can be concluded that dry storage under cask conditions even with starting temperatures to 400°C is not expected to cause cladding failures over the interim storage period.

Einfeld and Fleisch continue in their report (at 3-4) to comment on the successful demonstration of cask storage:

> A technical scale demonstration program with a fueled CASTOR cask is underway in the FRG since March 1982. The 16 assemblies which are subject to that program originate from the Wurgassen boiling water reactor. They resided in the core during 4 cycles of operation, burning up to about 27.8 GWD/t U.
>
> The general objectives of the demonstration with a fully instrumented cask and fuel bundles are the verification of cask design parameters, the operational experience in cask handling and the expansion of the data base on fuel performance. Fig. 2 shows a schematic drawing of the cask design and the axial thermocouple locations.
>
> The operational experiences and corresponding test data confirm the assumptions made about the cask concept and the cask loading and handling procedure. In addition, the technology data base for operating an interim storage plant could be expanded.
>
> – In-pool loading of a large storage cask and specific cask handling has been successfully demonstrated.
>
> – The passive heat transfer capabilities of the cask and fuel cladding integrity have been verified. The maximum local fuel rod temperatures for fuel with about one year decay time were within the expected range.
>
> – The total radiation shielding characteristics (< 10 mrem/h) are verified in practice (references deleted).

The authors conclude:

> The realization of the transport/storage cask concept, which is well under way in the Federal Republic of Germany, will provide sufficient interim spent fuel storage

capacity with the facilities planned or under construction. Dry interim storage is a proven technology and thus it constitutes an essential step in closing the backend of the nuclear fuel cycle.

R.J. Steffens and J.B. Wright's paper,[7] "Drywell Storage Potential," discussed drywell storage experience with pressurized water reactor spent fuel at the Nevada Test site. On page 6 of the paper, the authors note:

> Another drywell performance assessment method being employed during the demonstration storage period is that of periodically monitoring the storage canister atmosphere for fission products, specifically krypton-85 gas. Samples drawn to date have shown no detectable concentrations of this product after approximately 3 years of storage, indicating a maintenance of the fuel cladding integrity.

A third paper presented at the same Topical Meeting, by E.R. Gilbert and A.B. Johnson, Jr., "Assessment of the Light-Water Reactor Fuel Inventory for Dry Storage," focuses on dry spent fuel storage with respect to an acceptable temperature range for storage in air. They conclude on page 8 of their report:

> Dry storage demonstrations now in progress suggest that by 1986 a major fraction of the U.S. PWR spent fuel inventory that was placed in water storage before 1981 can be stored in dry storage facilities below 150 to 200°C.
>
> The LWR fuel inventory offers good prospects that the thermal characteristics of consolidated fuel will be acceptable for dry storage by proper selection of fuel.
>
> Dry storage of LWR fuel with defective cladding may be tolerable in inert cover gases or at temperatures below the threshold for significant oxidation in oxidizing cover gases. The range of acceptable storage temperatures is being investigated.

With respect to dry storage of spent fuel, the Commission notes the summary statement from "Behavior of Spent Nuclear Fuel and Storage Components in Dry Interim Storage" (PNL-4189), *supra* note 5, at xvii:

> Operational problems in vaults and dry wells have been minor after 1 to 18 yr. of operation (in 1982); and 7 yr of silo experience suggests that decades of satisfactory operation can be expected. Demonstration tests with irradiated fuel in metal storage casks are just beginning, but metal shipping casks with mild steel chambers have been used since the mid-1940s. Metal storage/shipping casks have successfully survived fire, drop, and crash tests.

[7] *Proceedings of the American Nuclear Society's Topical Meeting on Options for Spent Fuel Storage*, Savannah, Georgia (September 26-29, 1982).

Thus, with respect to the storage of spent fuel under dry conditions at storage installations located either at reactor sites or away from reactor sites, the Commission believes that current dry-storage technology is capable of providing safe storage for spent nuclear fuel. The modular character of dry storage installations enhances the ability to perform maintenance or to correct mechanical defects, if any should occur. The Commission is confident that its regulations will assure adequate protection of the public health and safety and the environment during the period when the spent fuel is in storage.

The Commission notes that § 211(2)(B) of the Nuclear Waste Policy Act authorizes the Secretary of Energy to carry out research on, and to develop facilities to demonstrate, dry storage of spent nuclear fuel. Although this provision indicates a judgment on the part of the Congress that additional research and demonstration is needed on the dry storage of spent fuel, the Commission believes the information discussed above is sufficient to reach a conclusion on the safety and environmental effects of extended dry storage. All areas of safety and environmental concern (e.g., maintenance of systems and components, prevention of material degradation, protection against accidents and sabotage) have been addressed and shown to present no more potential for adverse impact on the environment and the public health and safety than storage of spent fuel in water pools.

The technical studies cited above support the conclusion that corrosion would have a negligible effect during several decades of extended dry storage. The Commission's confidence in the safety of dry storage is based on an understanding of the material degradation processes, rather than merely on extrapolation of storage experience — together with the recognition that dry storage systems are simpler and more readily maintained. For these reasons, the Commission is confident that dry storage installations can provide continued safe storage of spent fuel at reactor sites for at least 30 years after expiration of the plant's license.

D. Potential Risks of Accidents and Acts of Sabotage at Spent Fuel Storage Facilities

The Commission finds that the risks of major accidents at spent fuel storage pools resulting in offsite consequences are remote because of the secure and stable character of the spent fuel in the storage pool environment, and the absence of reactive phenomena — "driving forces" — which may result in dispersal of radioactive material. Reactor storage pools and independent spent fuel storage installations have been designed to safely withstand accidents caused either by natural or man-made phenomena. Even remote natural risks such as earthquakes and tornados and the risks of human error such as in handling or storing spent fuel are addressed in the design and operational activities of storage facilities and in NRC's licensing reviews thereof under its regulations. Under 10 C.F.R. Parts 50 and 72, spent fuel is stored in facilities structurally designed to withstand accidents and external hazards, such as those cited above, and to preclude radiation and radioactive material emissions from spent fuel that would significantly endanger the public health and safety. In order to preclude the possibility of criticality under normal or accident conditions, the spent fuel is stored in racks designed to maintain safe geometric configurations under seismic conditions. The spent fuel itself consists of solid ceramic pellets which are encapsulated in metal-clad rods held in gridded assemblies and stored underwater in reinforced concrete structures or in sealed dry storage installations such as concrete dry wells, vaults and silos or massive metal casks. The properties of the spent fuel (which in extended storage has decayed to the point where individual fuel assemblies have a heat generation rate of several hundred watts or less) and of the benign storage environment result in spent fuel storage being an activity with very little potential for adversely affecting the environment and the public health and safety. While any system employing high technology is subject to some equipment breakdowns or accidents, water pool storage facilities have operated with few serious problems (DOE PS at IV-56 to IV-57; UNWMG-EEI PS, Doc. 4, at 26). In these cases, the events at spent fuel pools have been manageable on a timely basis. Similarly, dry storage of spent fuel, as discussed in § C, above, appears to be at least as safe as water pool storage. A discussion of risks related to spent fuel storage is provided below.

Comments from participants on the subject of accidents and their potential consequences at spent fuel storage facilities included a description of nonspecific references to numerous "accidents" in spent fuel storage facilities, a discussion of cases of leaks and inadvertent releases of contaminated storage pool water, and a suggestion that waste storage should be physically separated from reactor operation to reduce the risk of damage to the storage facility in the event of a reactor accident, and vice versa (NY PS at 102-07; OCTLA PS at 12). The State of New York, in its discussion of possible accidents at spent fuel storage pools, cited reports of an accident in the Soviet Union that is believed to have involved reprocessing plant wastes stored in tanks at a waste storage facility (NY PS at 107-08). The situation, as reconstructed from limited data, cannot be compared to the storage of ceramic fuel in metal cladding, placed in water storage pools. The issue raised, therefore, is not relevant to this proceeding. The need for continued management of pool storage facilities over an extended time period was considered by some participants as creating a potential hazard because of the increased possibility of human errors or mismanagement (NRDC PS at 89-90). The State of New York characterized the Three Mile Island reactor accident as caused by multiple technical and human failures, and postulated that such failures are possible at storage facilities, and would result in serious offsite consequences (NY PS at 107).

These observations do not appear to take account of the numerous safety analyses that have been made of water pool storage and of alternative long-term storage methods which have demonstrated storage to be both safe and environmentally acceptable. Of course, the possibility of human error cannot be completely eliminated. However, Commission regulations (e.g., 10 C.F.R. Part 55; 10 C.F.R. Part 72, Subpart I) include explicit requirements for operator training, the use of written procedures for all safety-related operations and functions in the plant, and certification or licensing of operators, with the objective of minimizing the opportunity for human error. Unlike the accident at the Three Mile Island reactor, human error at a spent fuel storage installation does not have the capability to create a major radiological hazard to the public. The absence of high temperature and pressure conditions that would provide a driving force essentially eliminates the likelihood that an operator error would lead to a major release of radioactivity (DOE CS at II-156 to II-158). In addition, features incorporated in storage facilities are designed to mitigate the consequences of accidents caused by human error or otherwise (DOE PS at IV-34).

The possibility of terrorist attacks on nuclear facilities was advanced as an argument against the acceptability of extended interim storage of spent fuel (NRDC PS at 90). The intentional sabotage of a storage pool facility is possible, and NRC continues to implement actions to further improve security at such facilities. The consequences would be limited by the realities that, except for some gaseous fission products, the radioactive content of spent fuel is in the form of solid ceramic material encapsulated in high-integrity metal cladding and stored underwater in a reinforced concrete structure. Under these conditions, the radioactive content of spent fuel is relatively invulnerable to dispersal to the environment (NUREG-0575, Vol. 1, *supra*). Similarly, dry storage of spent fuel in dry wells, vaults, silos and metal casks is also relatively invulnerable to sabotage and natural disruptive forces, because of the weight and size of the sealed, protective enclosures which may include 100-ton steel casks, large concrete-lined near-surface caissons and surface concrete silos (NUREG/CR-1223, *supra*, at IV-C.2).

E. Summary

In summary, the Commission finds that spent fuel can be stored safely at independent spent fuel storage installations or at reactor sites for at least 30 years beyond the expiration of reactor operating licenses. This finding is based on extensive experience and on many factors that are not site-specific. These factors include the substantial capability of the fuel cladding to maintain its integrity under storage conditions, a capability verified in extensive technical studies and experience; the extreme thermal and chemical stability of the fuel form, enriched uranium oxide pellets; the long-term capability of spent fuel storage facilities to dissipate spent fuel heat and retain any radioactive material leakage; and the relatively straightforward techniques and procedures for repairing spent fuel storage structures, replacing defective components or equipment, or undertaking other remedial actions to assure containment of radioactivity (Johnson, "Behavior of Spent Nuclear Fuel in Water Pool Storage" (UC-70), *supra*). These factors contribute to the assurance that spent fuel can be stored for extended periods without significant impact on the public health and safety and the environment. Moreover, any storage of spent fuel at independent spent fuel storage installations or reactor sites beyond the operating license expiration will be subject to licensing and regulatory control to assure that operation of the storage facilities does not result in significant impacts to the public health and safety.

For the reasons discussed previously (§§ 2.4-A through 2.4-D, above), the Commission also concludes, from the record of this proceeding, that storage of spent fuel either at or away from a reactor site for 30 years beyond the operating license expiration would not result in a significant impact to the environment or an adverse effect on the public health and safety. The Commission's findings are also supported by NRC's experience in more than 80 individual safety evaluations of spent fuel storage facilities conducted in recent years. The record indicates that significant releases of radioactivity from spent fuel under licensed storage conditions are highly unlikely. This is primarily attributable to the resistance of the spent fuel to corrosive mechanisms and the absence of any conditions that would result in offsite dispersal of radioactive material. The Commission concludes that the possibility of a major accident or sabotage with offsite radiological impacts at a spent fuel storage facility is extremely remote because of the characteristics of spent fuel storage. These include the inherent properties of the spent fuel itself, the benign nature of the water pool or dry storage environment, and the absence of any conditions that would provide a driving force for dispersal of radioactive material. Moreover, there are no significant additional nonradiological impacts which could adversely affect the environment if spent fuel is stored beyond the expiration of operating licenses for reactors. The nonradiological environmental impacts associated with site preparation and construction of storage facilities are, and will continue to be, considered by the NRC at the time applications are received to construct these facilities, which are licensed under NRC's regulations in either 10 C.F.R. Part 50 for reactors or 10 C.F.R. Part 72 for independent spent fuel storage facilities. The procedure to be followed in implementing the Commission's generic determination is the subject of rulemaking which the Commission has conducted.

2.5 Fifth Commission Finding

The Commission finds reasonable assurance that safe independent onsite spent fuel storage or offsite spent fuel storage will be made available if such storage capacity is needed.

The technology for independent spent fuel storage installations as discussed under the Fourth Commission Finding, is available and demonstrated. The regulations and licensing procedures are in place. Such installations can be constructed and licensed within a 5 year time interval. Before passage of the Nuclear Waste Policy Act of 1982 the Commission was concerned about who, if anyone, would take responsibility for providing such installations on a timely basis. While the industry was hoping for a government commitment, the Administration had discontinued efforts to provide those storage facilities (Tr. at 157-58). The Nuclear Waste Policy Act of 1982 establishes a national policy for providing storage facilities and thus helps to resolve this issue and assure that storage capacity will be available.

Prior to March 1981, the DOE was pursuing a program to provide temporary storage in offsite, or away-from-reactor (AFR), storage installations. The intent of the program was to provide flexibility in the national waste disposal program and an alternative for those utilities unable to expand their own storage capacities (DOE PS at I-11; DOE CS at II-66). Consequently, the participants in this proceeding assumed that, prior to the availability of a repository, the Federal government would provide for storage of spent fuel in excess of that which could be stored at reactor sites. Thus, it is not surprising that the record of this proceeding prior to the DOE policy change did not indicate any direct commitment by the utilities to provide AFR storage. On March 27, 1981, DOE placed in the record a letter to the Commission stating its decision "to discontinue its efforts to provide Federal government-owned or controlled away-from-reactor storage facilities." The primary reasons for the change in policy were cited as new and lower projections of storage requirements and lack of congressional authority to fully implement the original policy.

The record of this proceeding indicates a general commitment on the part of industry to do whatever is necessary to avoid shutting down reactors or derating them because of filled spent fuel storage pools. While industry's incentive for keeping a reactor in operation no longer applies after expiration of its operating license, utilities possessing spent fuel are required to be licensed and to maintain the fuel in safe storage until removed from the site. Industry's response to the change in DOE's policy on federally sponsored, away-from-reactor (AFR) storage was basically a commitment to do what is required of it, with a plea for a clear unequivocal Federal policy (Tr. at 157-59). The Nuclear Waste Policy Act of 1982 has now provided that policy.

The Nuclear Waste Policy Act defines public and private responsibilities for spent fuel storage and provides for a limited amount of federally supported interim storage capacity. The Act also includes provisions for monitored retrievable storage facilities and for a research, development and demonstration program for dry storage. The Commission believes that these provisions provide added assurance that safe independent onsite or offsite spent fuel storage will be available if needed.

In Subtitle B of the Act, "Interim Storage Program," Congress found that owners and operators of civilian power reactors "have the primary responsibility for providing interim storage of spent nuclear fuel from such reactors" by maximizing the use of existing storage facilities on site and by timely additions of new onsite storage capacity. The Federal government is responsible for encouraging and expediting the effective use of existing storage facilities and the addition of new storage capacity as needed. In the event that the operators cannot reasonably provide adequate storage capacity to assure the continued operation of such reactors, the Federal government will assume responsibility for providing interim storage capacity for up to 1900 metric tons of spent fuel (§ 131(a)). Such interim storage capacity is to be provided by the use of available capacity at one or more Federal facilities, the acquisition of any modular or mobile storage equipment including spent fuel storage racks, and/or the construction of new storage capacity at any reactor site (§ 135(a)(1)).

The Nuclear Waste Policy Act authorizes the Secretary of Energy to enter into contracts with generators or owners of spent fuel to provide for storage capacity in the amount provided in the Act (§ 136(a)(1)). However, such contracts may be authorized only if the NRC determines that the reactor owner or operator cannot reasonably provide adequate and timely storage capacity and is pursuing licensed alternatives to the use of Federal storage capacity (§ 135(b)).[8] Further, any spent fuel stored in the "interim storage program" is to be removed from the storage site or facility "as soon as practicable" but in no event later than 3 years following the availability of a repository or monitored retrievable storage facility (§ 135(e)). The Act establishes an "Interim Storage Fund" for use in activities related to the development of interim storage facilities, including the transportation of spent fuel and impact assistance to State and local governments (§ 136(d)).

In addition to providing for interim storage capacity, Congress found that "the long-term storage of high level radioactive waste or spent nuclear fuel in monitored retrievable storage facilities is an option for providing safe and reliable management of such waste or spent fuel." By June 1, 1985, the Secretary of Energy must complete a detailed study of the need for, and feasibility of, such a facility and submit to Congress a proposal for the construction of one or more such facilities. The Act also directs the Secretary of Energy to establish a demonstration program, in cooperation with the private sector, for the dry storage of spent nuclear fuel at reactor sites and provide consultative and technical assistance on a cost-sharing basis to assist utilities lacking interim storage capacity to obtain the construction, authorization and appropriate license from the NRC. Such assistance may include the establishment of a research and development program for the dry storage of no more than 300 metric tons of spent fuel at federally owned facilities (§ 218(a), (b), and (c)).

The Commission's confidence that independent onsite and/or offsite storage capacity for spent fuel will be available as needed is further supported by the strong likelihood that only a portion of the total spent fuel generated will require storage outside of reactor storage basins (DOE PS at V-3 to V-13). Estimates of the amount of spent fuel requiring storage away from reactors have declined significantly over the duration of this proceeding (DOE March 27, 1981, letter from O. Brown, II, DOE Office of General Counsel, to M. Miller, NRC, Presiding Officer in this proceeding).

DOE reported that cumulative spent fuel discharges, previously estimated as 100,000 metric tons of uranium (MTU), dropped to 72,000 MTU through the year 2000. Projected requirements for additional spent fuel storage capacity begin in 1986 (instead of 1981) and increase to 9500 MTU per year by 1997. Earlier projections indicated a need for

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[8] Accordingly, the Commission has published proposed "Criteria and Procedures for Determining the Adequacy of Available Spent Nuclear Fuel Storage Capacity," 10 C.F.R. Part 53 (48 Fed. Reg. 19,382 (1983)).

16,000 MTU per year for additional storage capacity in 1997.[9] DOE pointed out that additional storage requirements could be satisfied in a number of ways, including: (a) use of private existing AFR storage facilities; (b) construction of new water basins at reactor facilities or away-from-reactor facilities by private industry or the utilities; (c) transshipment of spent fuel between reactors operated by different utilities; (d) disassembly of spent fuel and storage of spent fuel rods in canisters; and (e) dry storage at reactor sites.

Subsequently, DOE published new estimates for additional spent fuel storage capacity ("Spent Fuel Storage Requirements," DOE/RL-82-1, June 1982). These estimates show a maximum required away-from-reactor (AFR) storage capacity of 8610 metric tons uranium of spent fuel in the year 1997. This is a decline from DOE's previously published planning-base case. The information in Table 1, below, is excerpted from DOE/RL-83-1 and provides a range of projections of additional storage capacity needs. The first column is a projection of storage capacity needed over and above the currently existing and planned storage capacity. The second column provides projected values of additional storage capacity needed if maximum re-racking is conducted at existing or planned reactor basin storage pools. The storage capacity needs shown in the second column are somewhat smaller than in the first column. A further decrease in additional needed storage capacity is shown in the third column, which takes into account the possibility of transshipment of fuel from one reactor basin to another basin owned by the same utility. The projected values of needed storage capacity in the first and third columns provide a range of upper- and lower-bound values, respectively. The most likely outcome expected by DOE corresponds to the values in the second column. This was formerly known as the planning-base and is now termed the reference case. All projections shown in the table assume the maintenance of a full-core reserve. The magnitude of need for additional spent fuel storage capacity projected by DOE has continued to decline, even though DOE has not assumed the use of newly developed technology, such as fuel rod consolidation.

The cumulative amount of spent fuel to be disposed of in the year 2000 is expected to be 58,000 metric tons of uranium (Spent Fuel Storage Requirements (Update of DOE/RL-82-1), DOE/RL-83-1, published January 1983). The additional required storage capacity of 13,000 metric tons of uranium projected in the second column for the year 2000 is less than 25% of the total quantity of spent fuel projected to be in storage. It

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[9] DOE's planning-base studies assume maximum basin re-racking at reactors and the maintenance of full-core reserve in reactor basins.

Table 1: Additional Cumulative Spent Fuel Storage Requirements, Over and Above Current and Planned Storage at Reactor Storage Basins (Metric Tons of Uranium)*

Year	No change in current or planned storage capacity	Use maximum re-racking of current and planned storage capacity	Maximum re-racking plus transshipment
1982	0	0	0
1983	0	0	0
1984	13	13	0
1985	13	13	0
1986	110	110	3
1988	550	490	90
1990	1,500	1,360	310
1995	5,610	5,060	3,000
2000	14,760	13,090	10,370

*Spent Fuel Storage Requirements (Update of DOE/RL-82-1) DOE/RL-83-1, published January 1983.

is expected that additional storage will be provided at the reactor site, with some smaller portion to be moved off site.

In response to the Commission's Second Prehearing Memorandum and Order (November 6, 1981) the participants commented on the significance to the proceeding of issues resulting from the DOE policy change on spent fuel storage. The utilities generally limited their written responses to a restatement of the safety of interim storage and an affirmation of the technical and practical feasibility of the alternatives to Federal AFR storage facilities. An implied commitment by industry to implement AFR storage if necessary using one of the several feasible spent fuel storage alternatives is evident from the responses of the utilities, the nuclear industry, and associated groups (i.e., Tr. at 159).

Based upon the foregoing, the Commission has, then, reasonable assurance that safe independent onsite or offsite spent fuel storage will be available if needed. The technology is demonstrated and the licensing procedures are in place. The Nuclear Waste Policy Act establishes a national policy on interim storage of spent fuel and provides for contingency Federal storage capacity to augment that provided by industry. Further, the amount of fuel which may have to be stored in independent spent fuel storage facilities is less than was originally thought.

REFERENCE NOTATION

The following abbreviations have been used for the reference citations in the Appendix:

PS	Position Statement
CS	Cross-Statement
PHS	Prehearing Statement
Tr.	Transcript* of January 11, 1982 public meeting with the Commissioners

Participants have been identified by the following citations.

Citation	Participant
AIChE	American Institute of Chemical Engineers
ANS	American Nuclear Society
AEG	Association of Engineering Geologists
AIF	Atomic Industrial Forum, Inc.
Bech	Bechtel National, Inc.
CDC	California Department of Conservation
CEC	California Energy Commission
CPC	Consumers Power Company
Del	State of Delaware
DOE	U.S. Department of Energy
ECNP	Environmental Coalition on Nuclear Power
GE	General Electric Company
Ill	State of Illinois (PS includes Roy affidavit)
Lewis	Marvin I. Lewis
Lochstet	Dr. William A. Lochstet
Minn	State of Minnesota
MAD	Mississippians Against Disposal
NECNP	New England Coalition on Nuclear Pollution
NfE	Neighbors for the Environment (PS includes papers by Dornsife, Rae, and Strahl)
NRDC	Natural Resources Defense Council, Inc.
NY	State of New York

*The Commission considers this transcript to be part of the administrative record in this rulemaking. However, the transcript has not been reviewed for accuracy by the Commission or the participants, and therefore is only an informal record of the matters discussed.

Citation	Participant
OCTLA	Ocean County and Township of Lower Alloway Creek
Ohio	State of Ohio
SC	State of South Carolina
SE2	Scientists and Engineers for Secure Energy, Connecticut Chapter
SHL	Safe Haven, Ltd.
SMP	Sensible Maine Power, Inc.
TVA	Tennessee Valley Authority
UNWMG-EEI	Utility Nuclear Waste Management Group-Edison Electric Institute
USGS	United States Geological Survey
Vt	State of Vermont
Wis	State of Wisconsin (PS includes comments by Deese, Mudrey, Kelly, and Leverance)
UG	The Utilities Group (Niagara Mohawk Power Corp., Omaha Public Power District, Power Authority of the State of New York, and Public Service Company of Indiana, Inc.)

Atomic Safety and Licensing Appeal Boards Issuances

ATOMIC SAFETY AND LICENSING APPEAL PANEL

Alan S. Rosenthal, Chairman
Dr. John H. Buck
Dr. W. Reed Johnson
Thomas S. Moore
Christine N. Kohl
Gary J. Edles
Dr. Reginald L. Gotchy
Howard A. Wilber

APPEAL BOARDS

Cite as 20 NRC 375 (1984) **ALAB-779**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

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

In the Matter of **Docket No. 50-322-OL**

LONG ISLAND LIGHTING COMPANY
(Shoreham Nuclear Power Station, Unit 1)

August 3, 1984

The Appeal Board explains, for the benefit of the parties and the Commission, its agreement with the determination of the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel not to refer to the Appeal Board his denial of intervenor's motion calling for his disqualification from participation in any matters concerning the Shoreham facility.

RULES OF PRACTICE: REFERRAL OF RULING (MOTION FOR DISQUALIFICATION)

The Commission's regulation at 10 C.F.R. § 2.704(c) provides for referral to the Commission or Appeal Board of only those disqualification motions addressed to the presiding officer or a designated member of a licensing board.

MEMORANDUM

On June 22, 1984, intervenors Suffolk County and the State of New York filed a motion calling upon B. Paul Cotter, Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, to disqualify himself from participating in any matters concerning the Long Island Lighting Company's (LILCO) Shoreham Nuclear Power Station. This motion is one of three filed by the intervenors seeking disqualification of, respectively, the presiding Licensing Board in the low-power phase of the *Shoreham* operating license proceeding, NRC Chairman Palladino, and Judge Cotter. Administrative Judges Marshall E. Miller, Glenn O. Bright, and Elizabeth B. Johnson, who constitute the low-power Licensing Board, declined to step down. As required by 10 C.F.R. § 2.704(c) their decision was referred to us. We affirmed. *See* ALAB-777, 20 NRC 21 (1984). The motion to disqualify Chairman Palladino is pending before him.

Judge Cotter denied the motion for his disqualification in a memorandum and order issued on August 1, 1984. LBP-84-29A, 20 NRC 385. In a footnote in his decision, he observed that 10 C.F.R. § 2.704(c) provides for referral "to the Commission or the Atomic Safety and Licensing Appeal Board, as appropriate" of only those disqualification motions addressed to the "presiding officer or a designated member of an atomic safety and licensing board . . . ." Thus, he did not refer the motion to us.

We agree with Judge Cotter's disposition insofar as referral to this Board is concerned. To begin with, the express terms of the regulation apply only where "*the presiding officer* does not grant the motion or *the board member* does not disqualify himself . . ." (emphasis added). Judge Cotter is neither the "presiding officer" nor a "member" of a licensing board assigned to hear this case. Moreover, as best we can tell from the administrative history of this regulation, there was no intent to include within its scope anyone other than members of individual licensing boards.[1] Finally, it appears that Judge Cotter came into contact with the *Shoreham* litigation only in his administrative capacity as Chairman of the Atomic Safety and Licensing Board Panel. His functioning in that role here is better supervised by the Commission rather than an appeal board.

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[1] When the Commission revised section 2.704 in 1975, it explained: "Section 2.704 currently contains provisions pertaining to the disqualification of a 'presiding officer' on his own motion or that of a party. Clarifying language has been added to reflect current understanding and practice that these provisions apply to all members of a licensing board. In addition, this Section is revised to reflect that a motion to disqualify a Board member shall be referred to the Commission, or the Atomic Safety and Licensing Appeal Board, as appropriate." 40 Fed. Reg. 51,995-96 (1975).

We have stated our intention not to review Judge Cotter's decision for the information of the parties and the Commission. In the circumstances, we express no view whatsoever with respect to the merits of the motion for disqualification.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING APPEAL BOARD

Administrative Judges:

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

In the Matter of **Docket No. 50-322-OL-3**
(Emergency Planning)

LONG ISLAND LIGHTING COMPANY
(Shoreham Nuclear Power Station, Unit 1) **August 15, 1984**

The Appeal Board denies as interlocutory a party's appeal of a Licensing Board order denying that party's request for discovery. Treating the appeal as a motion for directed certification of the order, the Appeal Board denies the motion.

RULES OF PRACTICE: INTERLOCUTORY APPEALS

Section 2.730(f) of 10 C.F.R. generally prohibits interlocutory appeals. The single exception to that prohibition is found in 10 C.F.R. 2.714a, which allows an appeal from certain orders entered on petitions for leave to intervene in an adjudicatory proceeding.

RULES OF PRACTICE: INTERLOCUTORY APPEALS (DISCOVERY ORDERS)

An order granting discovery against a non-party to a proceeding has all of the attributes of finality insofar as that non-party is concerned and, thus, is appealable as a matter of right. *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-116, 6 AEC 258, 258 (1973). *See also Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-773, 19 NRC 1333 (1984).

RULES OF PRACTICE: INTERLOCUTORY APPEALS (DISCOVERY ORDERS)

An order that denies discovery by quashing a subpoena addressed to a non-party is wholly interlocutory in character and, accordingly, is not immediately appealable. *Zion, supra,* 6 AEC at 258; 10 C.F.R. 2.730(f).

RULES OF PRACTICE: INTERLOCUTORY APPEALS (DISCRETIONARY REVIEW)

A Licensing Board ruling normally will qualify for discretionary interlocutory review only if it 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. *Public Service Co. of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977). Discovery rulings rarely meet those tests. *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-634, 13 NRC 96, 99 (1981). *See also Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), ALAB-608, 12 NRC 168, 170 (1980).

APPEARANCES

Lawrence Coe Lanpher, Karla J. Letsche, Michael S. Miller, and **Christopher M. McMurray,** Washington, D.C., and **Martin Bradley Ashare,** Hauppauge, New York, for the intervenor, Suffolk County, New York.

Richard J. Zahnleuter, Albany, New York, for the intervenor, State of New York.

Donald P. Irwin and **Lee B. Zeugin,** Richmond, Virginia, for the applicant, Long Island Lighting Company.

Stewart M. Glass, New York, New York, for the Federal Emergency Management Agency.

Bernard M. Bordenick for the Nuclear Regulatory Commission staff.

MEMORANDUM AND ORDER

On July 26, 1984, intervenor Suffolk County filed a notice of appeal (together with a supporting brief) from a July 10, 1984 oral order of the Licensing Board in the emergency planning phase of this operating license proceeding. That order denied the County's motion seeking, *inter alia*, to compel the Federal Emergency Management Agency (FEMA) to produce certain documents.

In an unpublished July 27 order, we directed the County to show cause why the appeal should not be summarily dismissed in light of the general prohibition in 10 C.F.R. 2.730(f) against interlocutory appeals.[1] By way of response, the County conceded that the Licensing Board's oral order was interlocutory in character but nonetheless maintained that we should review it in the exercise of our discretion.[2] In this circumstance, we elected to treat the appeal as, in effect, a motion for directed certification of the oral order[3] and, accordingly, called for the views of the other parties to the controversy respecting whether the criteria for granting such relief were met.[4]

For the reasons that follow, we dismiss the appeal and deny directed certification.[5]

A. In our *Zion* decision more than a decade ago, we took note of the distinction, insofar as appealability is concerned, between an order "*granting* discovery against a non-party to the proceeding" and an order

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[1] The single exception to that prohibition is found in 10 C.F.R. 2.714a, which allows an appeal from certain orders entered on petitions for leave to intervene in an adjudicatory proceeding.

[2] Memorandum to Show Cause Why Suffolk County's July 26 Appeal Should Not Be Dismissed (August 1, 1984) at 2-8.

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

[4] August 2, 1984 order (unpublished). In memoranda filed on August 10, 1984, (1) the State of New York supported Suffolk County; and (2) FEMA, the applicant Long Island Lighting Company and the NRC staff each took the position that interlocutory appellate review of the Licensing Board discovery order was not warranted.

[5] Our unpublished August 2 order did not either (1) specifically dismiss the appeal; or (2) detail the basis for our conclusion that the appeal would not lie and thus the County's papers should be treated as seeking discretionary appellate review. We therefore deal with these matters in this opinion.

that "*denies* discovery by quashing a subpoena addressed to the non-party."[6] The former, we observed, "has all of the attributes of finality insofar as that non-party is concerned" and, thus, is appealable as a matter of right.[7] On the other hand, an order denying discovery "is wholly interlocutory in character" and, accordingly, is not so appealable given the provisions of 10 C.F.R. 2.730(f).[8]

Precisely the same distinction is drawn in federal judicial practice.[9] And it explains why, in ALAB-773,[10] we recently entertained the appeal of FEMA from a Licensing Board order directing it to produce documents sought by the County. Because FEMA is a non-party in this proceeding, that production order had "all of the attributes of finality." In contrast, the Licensing Board order now challenged by the County — *denying* a discovery request directed to FEMA — has none of the attributes of finality but, rather, "is wholly interlocutory in character."[11]

B. A Licensing Board ruling normally will qualify for discretionary interlocutory review only if it "either (1) threaten[s] the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated by a later appeal or (2) affect[s] the basic structure of the proceeding in a pervasive or unusual manner."[12] We have observed that "[d]iscovery rulings rarely meet those tests."[13] Indeed, insofar as our research has disclosed, no prior endeavor to obtain directed certification of the *denial* of a discovery request has been successful.

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[6] *Commonwealth Edison Co.* (Zion Station, Units 1 and 2), ALAB-116, 6 AEC 258, 258 (1973) (emphasis in original).

[7] *Ibid.* As noted in *Zion* (at n.3), that consideration was at the root of our acceptance of an appeal from a Licensing Board order directing non-parties to comply with subpoenas issued at the behest of one of the parties to an antitrust proceeding. *See Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-122, 6 AEC 322 (1973).

[8] *Zion, supra,* 6 AEC at 258.

[9] *Compare EEOC v. Neches Butane Products Co.*, 704 F.2d 144, 148 (5th Cir. 1983) (discovery orders generally not appealable apart from a final decision in the case) with *Branch v. Phillips Petroleum Co.*, 638 F.2d 873 (5th Cir. 1981) (non-party government entity claiming privilege may appeal immediately from an order granting discovery against it).

[10] 19 NRC 1333 (1984).

[11] In these circumstances, we need not decide whether, had the July 10 oral ruling been an appealable order, the appeal nonetheless would have been subject to dismissal as untimely. Inasmuch as the notice of appeal was not filed until July 26, the answer to this question would have hinged in turn upon whether the 10-day appeal period prescribed in 10 C.F.R. 2.762(a) was applicable and, if not, what other provision of the Rules of Practice might be taken as setting a time limit.

[12] *Public Service Co. of Indiana* (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-405, 5 NRC 1190, 1192 (1977).

[13] *Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-634, 13 NRC 96, 99 (1981). *See also Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), ALAB-608, 12 NRC 168, 170 (1980) ("As a general matter, discovery rulings of licensing boards are not promising candidates for the exercise of our discretionary authority to review interlocutory orders.").

We see no reason to reach a different result here. Plainly, should it turn out that the discovery ruling in question contributes materially to an unfavorable outcome on the emergency planning issues, Suffolk County will be free to mount its challenge to the ruling on an appeal from that outcome. Equally plainly, there is no room for a serious claim that the ruling has affected the basic structure of the proceeding at all — let alone in a pervasive or unusual manner. To the contrary, the situation at bar cannot be differentiated from that in any other case in which a party endeavored unsuccessfully to acquire certain information to assist its preparation for trial. Even if the party might have been entitled to obtain the sought information by way of discovery, it scarcely follows that the proceeding was significantly altered in structure simply because the request was not enforced by the trial tribunal.

We need add only that the County's cause is not advanced by its reliance[14] on the following direction in the Commission's 1981 *Statement of Policy on Conduct of Licensing Proceedings:*

> 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.[15]

We have previously determined that "the Policy Statement does not, either explicitly or by necessary implication, call for a marked relaxation of the [existing interlocutory review] standard. Rather, in terms, it simply exhorts the licensing boards to put before us legal or policy questions that, in their judgment, are 'significant' and require prompt appellate resolution."[16] In this instance, the Licensing Board apparently did not regard its July 10 oral order as involving questions of that stripe. Nor do we. The legal issue at the root of this controversy was considered and decided in ALAB-773, *supra*. All that is currently in question is whether the Licensing Board correctly applied the standard established in that decision to the particular factual situation before it. That hardly is the kind of inquiry that the Commission's Policy Statement had in mind.

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[14] Suffolk County's August 1 Memorandum, note 2 *supra*, at 2.

[15] CLI-81-8, 13 NRC 452, 456.

[16] *Virginia Electric and Power Co.* (North Anna Power Station, Units 1 and 2), ALAB-741, 18 NRC 371, 375 (1983).

Appeal *dismissed;* directed certification *denied.*
It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker
Secretary to the
Appeal Board

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE
CHIEF ADMINISTRATIVE JUDGE B. PAUL COTTER, JR.

In the Matter of | **Docket No. 50-322-OL-4 (ASLBP No. 84-503-01 Misc.)**

SUFFOLK COUNTY AND STATE OF NEW YORK MOTION FOR DISQUALIFICATION OF CHIEF ADMINISTRATIVE JUDGE COTTER
(Shoreham Nuclear Power Station, Unit 1)

August 1, 1984

The Chief Administrative Judge of the Atomic Safety and Licensing Board Panel denies Intervenor Suffolk County's motion for recusal on the grounds that he has no adjudicatory responsibilities in connection with the *Shoreham* proceeding, and consequently no adjudicatory responsibility from which to recuse himself.

RULES OF PRACTICE: RECUSAL

The rules governing motions for recusal and their resolution are generally the same for the administrative judiciary as for the judicial branch itself, and the Commission has followed that practice.

RULES OF PRACTICE: RECUSAL

The Chief Administrative Judge of the Atomic Safety and Licensing Board Panel has no authority to decide any issue pending in the *Shoreham* proceeding, and consequently no adjudicatory responsibility from which to recuse himself.

ATOMIC SAFETY AND LICENSING BOARD PANEL: CHIEF ADMINISTRATIVE JUDGE; AUTHORITY

The Chief Administrative Judge of the Atomic Safety and Licensing Board Panel has no authority to refuse to perform the administrative responsibilities of his position.

MEMORANDUM AND ORDER

On June 22, 1984, the captioned County and State moved that the undersigned "disqualify himself from participating in any matters concerning the Long Island Lighting Company's ('LILCO') Shoreham Nuclear Power Station ('Shoreham')." Movants allege that a series of events during the 2 weeks ending March 30, 1984 (the date I appointed an Atomic Safety and Licensing Board to consider a motion filed by the Long Island Lighting Company), established grounds for concluding that I had "*in some measure* adjudged the facts as well as the law of [this] case in advance of hearing it" (emphasis in original), *citing Cinderella Career and Finishing Schools, Inc. v. FTC*, 425 F.2d 583, 591 (D.C. Cir. 1970), *quoting with approval from Gilligan, Will & Co. v. SEC*, 267 F.2d 461 (2d Cir. 1959). The NRC Staff filed a response on July 12, 1984.

The motion is anomalous and is devoid of basis or apparent precedent. Motions for disqualification or recusal are normally directed to a presiding judicial official who has responsibility for deciding a contested issue or issues. *See Withrow v. Larkin*, 421 U.S. 35 (1975). The rules governing such motions and their resolution are generally the same for the administrative judiciary as for the judicial branch itself, and this Commission has followed that practice. *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), CLI-82-9, 15 NRC 1363, 1366 (1982). In the instant case, I have no adjudicatory responsibilities in connection with the *Shoreham* proceeding. I am not a member of the Atomic Safety and Licensing Board hearing the case nor do I serve as an alternate member, a special master, a special assistant, or in any other quasi-adjudicatory position in connection with the case. *See* 10 C.F.R. §§ 2.704, 2.721 and 2.722 (1984). Consequently, I have no authority to decide any issue pending in the *Shoreham* proceeding and no adjudicatory responsibility from which to recuse myself.

To the extent the motion may be intended to address my role as the principal administrative officer of the Atomic Safety and Licensing

Board Panel, it is equally without foundation. I did appoint the members of three licensing boards which are hearing various aspects of the *Shoreham* proceeding, and, because of conflicts in workload, have had to reconstitute at least one of those Boards. *See* notices published at 47 Fed. Reg. 6510 (1982) (reconstitution); 48 Fed. Reg. 22,235-36 (1983) (emergency planning board); and 49 Fed. Reg. 13,611-12 (1984) (low-power board). Those appointment actions were taken pursuant to administrative responsibilities imposed upon me as Chief Administrative Judge of the Atomic Safety and Licensing Board Panel by the Atomic Energy Act and the Commission. 42 U.S.C. § 2011 (1982), as amended, 10 C.F.R. §§ 2.704, 2.721 (1984). I do not have the authority myself to refuse to perform such duties. *See Boyle v. United States*, 515 F.2d 1397, 1402 (Ct. Cl. 1975) and *Nagel v. Department of Health and Human Services*, 707 F.2d 1384, 1387 (Fed. Cir. 1983). Even if I did, I would not take any such action on the basis of the instant motion. The motion consists of a collection of unfounded accusations, unsupported allegations, distortions of events, hearsay, and omissions of significant facts (for example, the omission of the complete February 22, 1984, ruling of the Shoreham Licensing Board) concocted in an effort to create an appearance of impropriety or bias that does not exist. It does not warrant further discussion and will be dismissed.*

Nevertheless, the aggregate effect of the accusations and omissions is to inject a spurious dispute into the *Shoreham* proceeding and to impugn my own integrity. The latter result has broader effect because it has the potential to cast a shadow over other proceedings conducted by Atomic Safety and Licensing Boards that I have appointed in the past and will appoint in the future. Consequently, to remove those potentially harmful effects, attached (not published) to this memorandum and incorporated herein by reference as if set forth at length is my statement concerning the events resulting in the appointment of a board to hear LILCO's Supplemental Motion for Low-Power Operating License filed March 20, 1984.

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*Section 2.704(c) of 10 C.F.R. Part 2 provides that the denial of a motion to disqualify "shall be referred to the Commission or the Atomic Safety and Licensing Appeal Board, as appropriate, which will determine the sufficiency of the grounds alleged." By its terms, § 2.704(c) applies to a presiding officer or a member of a licensing board and therefore does not appear, on its face, applicable to the instant decision.

Order

For all the foregoing reasons, it is, this 1st day of August 1984,
ORDERED

That the Suffolk County and State of New York Motion for Disqualification of Chief Administrative Judge Cotter is denied.

B. Paul Cotter, Jr.
ADMINISTRATIVE JUDGE

[The Attachment has been omitted from this publication but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

Cite as 20 NRC 389 (1984) **LBP-84-29B**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

James L. Kelley, Chairman
Dr. James H. Carpenter
Glenn O. Bright

In the Matter of

Docket Nos. 50-400
50-401
(ASLBP No. 82-472-03-OL)

CAROLINA POWER & LIGHT COMPANY and NORTH CAROLINA EASTERN MUNICIPAL POWER AGENCY
(Shearon Harris Nuclear Power Plant, Units 1 and 2)

August 3, 1984

In this Memorandum and Order, the Licensing Board completes its rulings on the admissibility of the over 100 emergency planning contentions submitted by various intervenors.

EMERGENCY PLANNING: DECONTAMINATION

Emergency plans are not called upon by regulation or guidance to give an account of materials available for evacuee decontamination. NUREG-0654 focuses on providing for decontamination of emergency workers, who would be likely to face greater contamination dangers than evacuees would. *See* the evaluation criteria under § II.K in NUREG-0654. However, the plans must show that the responsibility for evacuee decontamination has been assigned to organizations which will be adequately trained to carry out the task.

EMERGENCY PLANNING: DECONTAMINATION

Any large decontamination of evacuees or vehicles at the border of the plume EPZ would very likely impede prompt evacuation of the most threatened part of the population around the plant. The desire to avoid purported safety measures that would impede evacuation is reflected in evaluation criterion II.J.10.h of NUREG-0654. It calls for siting the host areas, and thus the principal decontamination centers, "at least 5 miles, and preferably 10 miles, *beyond* the boundaries of the plume [EPZ]."

EMERGENCY PLANNING: REENTRY AND RECOVERY

The emphasis in evaluation criteria II.M.1 and II.M.3-4 in NUREG-0654 is on planning for the decision to reenter, not on measures to be executed during reentry and recovery. Presumably, the thought behind this emphasis is that the decision to reenter is equivalent to a decision to relax protective measures (evaluation criterion M.1 in NUREG-0654, § II) and is therefore to be made with a degree of care which requires some advance thought. However, since reentry and recovery would not take place under the same time pressures protective actions would, planning for measures to be executed during reentry and recovery needn't be more than general.

EMERGENCY PLANNING: IMPLEMENTING PROCEDURES

A finding that there is reasonable assurance that the plans can be implemented is, under 10 C.F.R. § 50.47(a)(2), to be based largely on the plans, not the myriad details of the implementing procedures. Implementability is a characteristic of good *plans*, for even the best implementing procedures cannot rescue an ill-conceived plan. Thus it is to the adequacy of planning that all of the Commission's planning standards and evaluation criteria are directed, not the mechanical details of implementing procedures. An intervenor looking to introduce such procedures into litigation would have to point to some plan provision drafted in such a way that a board would have to look at the implementing procedures under it to determine whether there was reasonable assurance it could be implemented. *Accord Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1106-07 (1983).

EMERGENCY PLANNING: EMERGENCY PLANNING ZONE SUB-AREAS

Sub-areas of the plume EPZ need not be perfectly regular, concentric rings, or parts thereof, any more than the EPZs themselves should be exactly 10 or 50 miles in radius. "The boundaries of the sub-areas shall be based upon the same factors as the EPZ, namely demography, topography, land characteristics, access routes, and local jurisdictions." NUREG-0654, Appendix 3, at 4-4.

EMERGENCY PLANNING: EMERGENCY PLANNING ZONE BOUNDARIES

State and local planning officials are not obliged to supply a written justification of their boundary-making until they are faced with an admitted contention on the subject. In particular, in the absence of such an admitted contention, officials need not justify in writing the exclusion from the plume EPZ of areas just inside the 10-mile limit. Section 50.47(c)(2) of 10 C.F.R. says that the plume EPZ shall be "about" 10 miles in diameter, not "at least."

EMERGENCY PLANNING: EVACUATION ROUTES

Evacuation routes are not simply routes out of the plume EPZ; they are routes to public shelters. Thus in order to reach the nearest shelter, some routes may have to carry traffic *toward* the plant before they carry it away.

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INDEX OF MR. EDDLEMAN'S CONTENTIONS

*Admitted, or admitted in part.
**Conditionally rejected.

FINAL SET OF RULINGS ON ADMISSIBILITY OF OFFSITE EMERGENCY PLANNING CONTENTIONS, RULING ON PETITION FOR WAIVER OF NEED-FOR-POWER RULE, AND NOTICE OF UPCOMING TELEPHONE CONFERENCE CALL

EVACUATION OF SPECIAL POPULATIONS: Eddleman Contentions 139, 140, 88, 235, 236(A), 236(B), 204, and 230

These contentions, for the most part, allege inadequate planning for the evacuation of certain populations: recreational, mobility-impaired, and school. We reject all of these contentions except 236(A) and 230, which we consider in connection with similar contentions filed by Dr. Richard Wilson.

Contentions 139, 140, and 88, which all deal with the recreation population, were first submitted before the offsite plans were available, and are now resubmitted without change. They therefore sometimes allege inaccurately, or about the onsite plan, or even the FES. In discussing these three contentions individually, we focus on their principal thrusts.

Furthermore, we shall construe allegations apparently directed at the onsite plans to be directed now to the offsite plans.

Contentions 139 and 140 both allege that the emergency response plans (ERPs) do not provide for prompt enough evacuation of the recreation population. These two contentions do not claim that particular plan provisions cause unnecessary delays in evacuation. As its sole basis, Contention 139 asserts that given the average wind speed around Harris, 7 mph, only about 1 hour and 25 minutes would be available to evacuate everyone in the plume EPZ. Contention 139 also asserts that since the effects of a severe accident at Harris could extend beyond the plume EPZ, the ERPs should "take into account" the recreation population within 20 miles of the plant. By "tak[ing] into account" we assume the contention means "evacuate."

As we said in our June 14, 1984, Order (unpublished), the NRC rules set no time limit on evacuation. *Id.* at 22-23. In particular, the NRC does not, and, in the nature of things, probably could not, require that if — in the situation Mr. Wells Eddleman treats as if it were the only one possible — evacuation were to begin precisely when a plume was released, evacuation could always be a step ahead of the plume. What the NRC rules do call for is that evacuation time estimates be part of the plans, to add to the information which would enable emergency response officials to choose wisely between sheltering and evacuation, both when evacuation is feasible before plume passage, and when it is not.

As were six contentions we rejected in our June 14, 1984, Order at 6, Contention 139's implied call for evacuation of the recreation population within a 20-mile area is an impermissible attack on the Commission's regulation on the size of the plume EPZ, 10 C.F.R. § 50.47(c)(2), which sets the radius of the plume EPZ at "about 10 miles."

Contention 88, besides repeating Contentions 139 and 140, asserts that the FES should have considered the costs of transportation and other emergency response adequate to assure the health and safety of the recreation population in the plume EPZ. As an attack on the FES, this contention comes too late. Even if Contention 88 is construed to be now directed at the ERPs, it is still to be rejected. Although funding "must be discussed between the individual nuclear utilities and the involved State and local governments . . ." (NUREG-0654, FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980, § I.G at 25), neither NRC regulation nor guidance suggests that the ERPs — which are supposed to make clear what is to be done in an emergency, how, and by whom (NUREG-0654, at 29) — should also set out costs.

Contentions 235, 236(A), 236(B), and 204 all concern evacuation transportation for the mobility-impaired. Contention 236(A) and one aspect of Contention 235 overlap and are encompassed by Wilson 7 and so will be considered later with Wilson 7. Contention 235 is the most general of this group of four contentions. It alleges that the State and local ERPs "fail to assess the resources necessary or available" to protect the mobility-impaired. As its principal basis, the contention cites the guidance in evaluation criterion J.10.d in NUREG-0654, which says that State and local ERPs for the plume EPZ "shall include: . . . d. Means for protecting those persons whose mobility may be impaired due to such factors as institutional or other confinement."

Mr. Eddleman apparently interprets the word "means" in J.10.d to mean "assessment of necessary and available resources." Assuming he is right, it would appear to us that in relation to some protective actions planned for the mobility-impaired, no assessment is needed, and that in relation to the remaining protective actions, Contention 235's call for assessment repeats other contentions which we have either admitted or deferred. Contention 235 cites as lacking assessment § IV.E.6 of each county plan and § IV.E.4.b of the State plan. The cited county sections list four protective measures which are part of sheltering: closing windows and doors, turning off air conditioners, "relocat[ing] to the best protection factors (PF)" in buildings, and distribution of KI. We see no need for the plans to assess the resources necessary and available for closing windows and turning off air conditioners, and we have already admitted contentions which allege that the PFs should be determined in advance of the emergency preparedness exercises, and that the county ERPs should include the quantities of KI stored for emergency use. *See* our June 14, 1984, Order at 18, 21-22. The cited State section lists the organizations which are to provide evacuation transportation for nonambulatory patients. Contention 235's concern with the adequacy of the resources of these organizations echoes the concerns behind Contention 236(A) and Wilson 7, and so we consider the three together later.

Contention 236(B) alleges that contrary to 10 C.F.R. § 50.47(b)(10) and evaluation criterion J.10.d in NUREG-0654, § II, the State and local ERPs do not show that "self-transport capability exists for all facilities for" the mobility-impaired and prisoners in the plume EPZ. We are not sure what Contention 236(B) intends. Certainly, the bases it cites do not support a claim that these facilities should have their own evacuation transportation resources. Perhaps Contention 236(B) intends to say that the lack of assessment alleged by Contention 236(A) might be justified if the plans were to show that these facilities could evacuate without any transportation resources the emergency response organizations named in the plans might have. If this is 236(B)'s intention, 236(B) is simply repeating the call for an assessment of resources for evacuation transportation. Thus, according to how Contention 236(B) is read, it is either redundant or lacking in basis.

Contention 204 alleges that the plans do not provide radiation-protected evacuation for people who require life support while being evacuated. As basis, the contention cites § III.C.3.a(3) of the State ERP, at 13, and alleges that this section points out the lack of radiation protection on National Guard helicopters. In fact, that section says nothing about radiation-protected evacuation. Rather, it reports that National Guard helicopters carry no life-support equipment. No NRC regulations or guidance call for radiation-protected evacuation.

Contention 230, the last of the group dealing with transportation for special populations, alleges principally that the ERPs fail to demonstrate adequacy of the resources available to evacuate the schools. Contention 230 is very similar to parts of Contention 222 and Wilson 7. We consider later these three contentions together.

MONITORING AND DECONTAMINATION OF EVACUEES: Eddleman Contentions 240 and 241

Contention 240, which we admit in part, alleges that procedures in the ERPs for monitoring evacuees for radioactive contamination are inadequate because, although the ERPs assign local governments the responsibility for monitoring at evacuation shelters, the ERPs do not show that the local governments have the "capabilities" for decontaminating evacuees, nor are the locations for evacuee decontamination and availability of materials for evacuee decontamination clear in the plans.

Since the contention distinguishes between "capabilities" and "materials," we construe the allegation that the plans do not show that local governments have the capabilities for evacuee decontamination to mean that the plans do not show that the responsibility for this task has been assigned to organizations which will be adequately trained to carry out the task.

Each of the County ERPs is very clear about where monitoring and decontamination of evacuees would take place. *See* Figure 6 in the ERPs for Chatham and Lee Counties, Figure 5 in the Harnett ERP, and Figure 7 in the Wake ERP. The ERPs do not give, and are not called upon by regulation or guidance to give, an accounting of materials available for evacuee decontamination. Indeed, neither regulations nor guidance even mention evacuee decontamination. Rather, NUREG-0654 focuses on providing for decontamination of emergency workers, who would be likely to face greater contamination dangers than evacuees would. *See* the evaluation criteria under § II.K in NUREG-0654.

However, the ERPs do not clearly show that local governments have the "capabilities" for evacuee decontamination. The Applicants cite sections which purport to assign responsibility for evacuee decontamination, others which the Applicants claim provide backup for the groups assigned the primary responsibility, and still other passages which provide for training the organizations assigned the primary responsibility. *See* Applicants' Answer at 75. However, one county plan does not clearly assign the primary responsibility, and no county plan clearly assigns the backup responsibility. Item (2) in Figure 6 of the Chatham plan says that decontamination of evacuees will be done by "Radiological Response Teams." Chatham ERP at 32. But we are unable to determine from the plan what unit of Chatham County government is responsible for establishing, training, and directing these teams.[1]

As for backup for evacuee decontamination, the Applicants, citing §§ IV.G.6 and 7 of the State ERP and § IV.E.12 of the county ERPs, claim it will be provided by the North Carolina Radiation Protection Section (RPS). But the cited section in the county plans speaks explicitly only of management of the shelters, and registration, feeding, and monitoring of evacuees; and it is not clear that the first of the cited State sections, IV.G.6, is speaking about more than decontamination of emergency workers. Annex H, the Plan Cross-Reference, which relates plan sections to the evaluation criteria of NUREG-0654, relates that section only to evaluation criterion K, which deals only with emergency

[1] Neither is it clear who is responsible for monitoring at the shelters in Chatham County. Item (2) in Figure 6 in the Chatham plan, at 32, assigns the monitoring to the County Department of Emergency Management, but § IV.E.12 of the same plan, at 31, assigns the monitoring to the Siler City Fire Department.

workers.[2,3] The other of the cited State sections, IV.G.7, speaks of State assistance only for monitoring.

Therefore Contention 240 is admitted, but only on the following questions: (1) What agency of Chatham County government is responsible for the decontamination of evacuees at the Chatham County Shelters? and (2) Which emergency response organizations are assigned the responsibility of providing support for the decontamination of evacuees? Perhaps all that is needed to answer these questions is authoritative clarification of the relevant sections of the ERPs.

Contention 241 alleges that the plans' use of schools as shelters in which decontamination would be done is unwise, that the schools would be left contaminated after a radiological emergency and the children using them later thus endangered. The contention offers monitoring as an alternative to decontamination in shelters, and by implication, decontamination of the evacuees "after they leave the EPZ before they continue to a host area," to prevent the spread of contamination and panic.

We reject 241. Part of it is without basis, and the rest does not address plan provisions which appear to satisfy these concerns as far as NRC rules require and good sense allows.

First, there is no asserted basis for the not very credible allegation that schools used as shelters would be left contaminated. Second and last, the ERPs do, in fact, provide for monitoring of evacuees and vehicles at traffic control points (*see* §§ III.C.2.j and III.D.1.c of the State ERP), and for some decontamination before evacuees proceed to shelters (*see* §§ IV.E.5.a-f of the State ERP), but they subordinate decontamination to the greater need to evacuate the plume EPZ quickly (*see id.*, §§ IV.E.5.a-c).

We are not aware of any NRC regulation or guidance which calls for monitoring and decontamination of all evacuees before they get beyond the plume EPZ. It would seem that any large-scale decontamination effort on the border of the plume EPZ would very likely impede prompt evacuation of the most threatened part of the population around the plant. The desire to avoid purported safety measures that would impede evacuation is reflected in evaluation criterion J.10.h, which calls for siting the host areas, and thus the principal decontamination centers, "at

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[2] But then, the page references in Annex H are not always complete, or accurate. *See, e.g.*, the page references for evaluation criterion J.12, at H-5.

[3] The Applicants also claim that a representative from the Shearon Harris Plant Environmental Radiation Control Unit, or from State Emergency Response Team (SERT), "will be dispatched to the scene to supervise the decontamination." Applicants' Answer at 75. The Applicants cite §§ IV.F.6 and 7 of the county plans. These sections, however, are together nearly identical to § IV.G.6 discussed above, and thus share its lack of clarity. Again, Annex H relates them only to evaluation criterion K, on control of doses to emergency workers.

least 5 miles, and preferably 10 miles, *beyond* the boundaries of the plume [EPZ]." (Emphasis in original.)

REENTRY AND RECOVERY: Eddleman Contentions 210, 100, and 100B

Contention 210 makes the general allegations that § IV.H of the State ERP fails to contain the general plan for recovery and decontamination which is required by 10 C.F.R. § 50.47(b)(13), and fails to comply with evaluation criteria M.1, M.3, and M.4[4] in NUREG-0654, which deal with both recovery and reentry.

Contentions 100 and 100B are more specific. They allege that the ERPs do not provide means of decontaminating farmland and homes, nor adequate provisions for decontamination of food and homes. Contention 100 makes this allegation with respect to contamination from "Class IX" accidents, 100B with respect to contamination from "Class X."

We reject all three of these contentions. They do not take account of all the provisions for reentry and recovery in the ERPs, nor do they show why the provisions they do take account of — only those in § IV.H of the State plan — do not conform to the cited evaluation criteria.

The emphasis in the cited criteria is on planning for the decision to reenter, not on what the contentions appear to be most concerned about, namely measures to be executed during reentry and recovery. The only evaluation criterion which says anything about those measures says only that "each organization, as appropriate, shall develop general plans and procedures for reentry and recovery . . . ." Evaluation criterion M.1 in NUREG-0654, § II. Thus the criterion is no more specific about measures to be executed during reentry and recovery than the planning standard it quotes, 10 C.F.R. § 50.47(b)(13).

Presumably, the thought behind this emphasis is that this decision to reenter is equivalent to a decision to relax protective measures (evaluation criterion M.1 in NUREG-0654, § II) and is therefore to be made with a degree of care which requires some advance thought. However, since reentry and recovery would not take place under the same time pressures protective actions would, planning for measures to be executed during reentry and recovery needn't be more than general.

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[4] The contention cites M.1, M.2, and M.3, but we take it M.1, M.3, and M.4 are intended, for M.2 applies only to the licensee's ERP, while M.4 does apply to the State ERP.

The various plans appear to conform to the guidance of the evaulation criteria in NUREG-0654, § II.M, particularly to the emphasis in those criteria on the decision to reenter. The second part of criterion M.1 calls on the plans to "describe the means by which decisions to relax protective measures . . . are reached." Sections IV.H.1-5 of the State ERP, and IV.G.1-3 of the county ERPs, appear to do just that. Criterion M.3 calls on the State plan to "specify means for informing . . . response organizations that a recovery . . . is to be initiated, and of any changes in the organizational structure . . . ." Sections IV.G.3-5 and IV.G.6.d-e appear to do just that. Criterion M.4 says that the State plan should "establish a method for periodically estimating total population exposure." This estimating, the crucial basis for the decision to reenter, appears to be provided for in §§ IV.H.1-3 of the State plan. Though the contentions cite the quoted criteria against the plans, they do not argue why the plans do not meet the criteria.

The ERPs appear to show conformance with that part of the criteria which the contentions are most concerned about, namely, the first part of M.1, that "each organization . . . shall develop general plans and procedures for reentry and recovery . . . ." Section IV.H.6 of the State plan briefly discusses responsibilities for public information, traffic control, assistance for evacuees in preparing to return to evacuated areas, and the monitoring of reentry and recovery operations. Section IV.G.4 of the county plans, which the contentions do not mention, lists several recovery operations, including medical services, continuous and long-term monitoring of people and property, security of property, and, of particular concern to the contention, "decontamination of people, animals, property, food and water." Section IV.G.4.a in the county plans. Many parts of § III in all the ERPs assign particular reentry and recovery responsibilities. In relation to decontamination, *see, e.g.*, in the State ERP, §§ III.C.3.f (operation of portable showers, decontamination of roads and structures), III.D.1.q (assessment of radiological damage to land and livestock), and III.D.3.c (management of waste from decontamination); in the county ERPs, Chatham § III.E.3.b (earth moving and washdowns). The contentions do not address these and similar passages.

MEDICAL CARE:
Eddleman Contentions 57-C-7, 56, 57-C-8, and 63

These four contentions overlap a great deal. To give a clearer sense of the whole of what they seek, we focus here on Contention 57-C-7, viewing the others as elaborations of it, and overlooking their redundancies.

Contention 57-C-7 has three main parts. The first alleges that there will not be enough hospitals to treat "radiation victims." Contention 56[5] elaborates on this by alleging that there are no plans to use hospitals which are more than 30 miles from SHNPP.

The second part of 57-C-7 alleges, correctly, that the State ERP does not contain the plans the hospitals have for treating radiation victims. Contention 57-C-8 elaborates by alleging that in order to judge whether the evaluation criteria in NUREG-0654, § II.L, have been satisfied, the State ERP should include all the procedures and reference materials mentioned in § V.B.2 of the State ERP: maps locating hospitals, addresses and phone numbers of hospital administrators, reports evaluating the capacities and needs of the hospitals, their plans for treating radiation victims, and the procedures for choosing hospitals and determining their needs.

The third and last part of 57-C-7 alleges that the State ERP does not provide "training or protection" for emergency workers transporting radiation victims to hospitals. The contention cites the State ERP at 85, with the comment, "handwaving." Contention 63 and part of 56 allege that the ERPs fail "to establish care for radiation victims on a mobile basis." Contention 63 alleges that to establish such care, the ERPs should provide for equipping mobile units, for staffing them and training the staff, and for assuring that adequate staff would be continuously available during a radiological emergency. Contention 63 cites as legal basis the footnote to the evaluation criteria in NUREG-0654, § II.L.

We reject all of these contentions except the first part of 57-C-7. The rejected contentions or parts of contentions either call for more than regulations and guidance call for or permit, or do not address the plans. We discuss the admitted portion of 57-C-7 after we discuss the other contentions and the rest of 57-C-7.

In relation to the second part of 57-C-7, neither NRC regulations nor guidance even suggest that any ERP should contain either the plans hospitals have for treating radiation victims or the procedures and reference materials — maps, phone numbers, reports, plans — mentioned in § V.B.2 of the State ERP: "Applicable supporting and reference documents and tables may be incorporated by reference .... The plans should be kept concise as possible. The average plan should consist of perhaps hundreds of pages, not thousands." NUREG-0654, at 29. Neither do we see why the information referred to in § V.B.2 must be in the

[5] When filed over 2 years ago, 56 was aimed at the onsite plan. It is now resubmitted, unaltered, but we construe its resubmission to mean that it is now intended as a contention about the offsite plans.

plans before it can be determined whether the plans conform to the evaluation criteria in § II.L in NUREG-0654. We would think that that determination could be made on the basis of information now in plans.

In relation to the third part of 57-C-7, the contention's citation to the State ERP at 85 apparently refers to Items e-g on that page, which discuss the training of personnel with medical duties. Citations to sections which provide for training are not much support for a contention which says the plans don't provide for training. The contention calls these passages "handwaving," but that word can hardly specify deficiencies in such a way as to make them the subject of admissible contentions. Further, 57-C-7's allegation that the State plan doesn't provide protection for personnel transporting radiation victims doesn't address the plans' many provisions for control of radiological exposure of emergency workers. *See, e.g.*, § G of the State ERP. Finally, no NRC regulation or guidance requires the ERPs to provide for the mobile equivalent of what hospitals can provide for radiation treatment. The footnote which Contention 63 cites is not to the contrary. It says only that plans and services developed under statutes and public health guidance which predate NUREG-0654 "should be compatible" with the response plans for Harris. Contention 63 cites no passages from either the guidance or the statutes cited in the footnote which require the sort of mobile care Contentions 63 and 56 allege should be provided.

We admit the first part of Contention 57-C-7, though in altered form. As we noted in our discussion of CHANGE's Contention 33 (at Tr. 868-69), we are barred by the Commission's decision in *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-83-10, 17 NRC 528 (1983) from considering in litigation, as 57-C-7 would have us do, whether medical services available in the region of Harris are in quantity adequate to deal with the number of people who, in a radiation accident at Harris, might be either contaminated and otherwise injured ("contaminated injured" in the language of NUREG-0654, § II.L) or simply seriously injured by radiation alone. The Commission accepted the thesis in *San Onofre* that there are likely to be so few contaminated injured that no arrangements beyond those already made under NUREG-0654, §§ II.L.1 and 3, and 10 C.F.R. Part 50, Appendix E, § IV.E.6, need be made, and that those seriously injured by radiation alone are so unlikely to need emergency treatment that treatment for them can be arranged *ad hoc*, going beyond local services if necessary. *San Onofre, supra*, 17 NRC at 535-36.[6]

[6] Here, then, is another reason why the mobile version of such treatment, called for by Contentions 63 and 56, is not required.

Therefore, we cannot admit the first part of 57-C-7 in the form in which it is presented. However, there is within that part of 57-C-7 something like a "lesser-included" contention, namely, that the ERPs should at least show what medical services are available for those seriously injured by radiation alone.

We admit this lesser-included contention, and we do so on the basis of the same case the Applicants cite in opposing all the contentions on medical care. Although *San Onofre* bars us from deciding whether medical facilities are quantitatively adequate, it requires that "emergency plans should include a listing of those local and regional medical facilities which have the capabilities to provide appropriate diagnosis and treatment for radiation exposure." *San Onofre, supra,* 17 NRC at 536. Here the Commission is speaking only of "individuals who have been subjected to dangerous levels of radiation and who need medical treatment for that reason." *Id.* at 535.

The ERPs for Harris do have lists of hospitals which "will support the plant and the surrounding communities in the event of a radiological emergency." Section V.B.3 of the State ERP. However, neither the State ERP nor the county ones make clear whether these hospitals are prepared to treat severe radiation exposure *per se.* Section V.B.2 of the State ERP speaks only of "victims of radiological accidents," or "contaminated patients," or "radiation accident victims." The county ERPs are no less ambiguous. *See, e.g.,* the Chatham ERP, § V.B.3.

Other aspects of the plans may indicate that the listed hospitals are prepared only for "contaminated injured" patients. For example, Annex H, the Plan Cross-Reference, refers to the pages among which these lists appear as intended to conform to the guidance of NUREG-0654, § II.L, but the only talk about lists in that guidance deals only with "contaminated injured." Also, the "Radiation Accident Hospital Evaluation Check Sheet" which the State ERP sets out (at 67) does not appear capable of unambiguously spotting those hospitals which are capable of treating severe radiation exposure *per se.*

Perhaps the main thing required to resolve 57-C-7 as admitted is — as with Contention 240 — authoritative clarification of the ERPs. However, even if the lists in the ERPs are of institutions which can treat radiation exposure, the lists may be incomplete: Section V.B.3 says that the RPS maintains lists of hospitals at greater distances which will provide backup, but *San Onofre* says the *plans* should include lists of local *and* regional hospitals with the necessary capabilities. *San Onofre, supra,* 17 NRC at 536.

We note last that we do not admit that part of Contention 56 which calls for plans to use medical facilities which are further than 30 miles from the Harris plant. Half of the hospitals listed in the State ERP are just that.

EXPERIENCE AND TRAINING:
Eddleman Contentions 212, 124, and 243

Contention 212 alleges that the planners have not been properly trained and cites as factual bases the planning deficiencies alleged in Mr. Eddleman's other contentions. We reject this contention. The number of Mr. Eddleman's admitted contentions appears to be too small to provide an adequate basis for 212. More fundamentally, however, this contention is premature. Unless and until it has been shown that Mr. Eddleman's emergency planning contentions have merit, there would be no practical reason to consider this contention. This contention could be reasserted when and if the developed evidentiary record provides a basis for it.

Contention 124 alleges that the Applicants and the counties which overlap the plume EPZ lack the experience and technical ability necessary to plan for a radiological emergency and to implement protective measures in the event of such an emergency. We reject the contention. It offers not the slightest indication of what levels of experience and technical ability are practically or legally necessary, or of how the Applicants and the counties fall short of these levels.

NRC regulations and guidelines set out standards and criteria for plans and preparedness, not for an applicant's or a county's experience. Of course, some regulations and guidelines do call for certain levels of technical ability, in communications, for example; but shortcomings in such abilities must be alleged with specificity.

Contention 243 alleges that since not all emergency response personnel have been trained yet, the ERPs do not meet the planning standard in 10 C.F.R. § 50.47(b)(15), which says that "training is provided to those who may be called on to assist in an emergency." We reject this contention also. The only deadline for completion of training is the natural one implied by whatever date is set for the emergency preparedness exercises. What the NRC looks for in relation to training is commitment, as evidenced by adequate planning, and results, as evidenced by preparedness exercises, but not the mere completion of training by some particular date before the exercises.

EMERGENCY PREPAREDNESS EXERCISES:
Eddleman Contentions 81 and 208

Contention 81 alleges that the ERPs have not been tested — "or otherwise formally evaluated" — will not be tested "before the plant operates," and should be.

Contention 208 adds that the ERPs "have not been tested under adverse weather conditions, e.g., snow, ice, fog, tornadoes or severe winds, or evacuation at the times most people are asleep (e.g., 1 am to 6 am)."

We reject both of these contentions. They do not address relevant provisions in the ERPs, and they implicitly attack the regulations. For one thing, the ERPs are being "formally evaluated" by FEMA and the NRC Staff, and in this proceeding. But more, as the regulations make clear, a full-scale exercise of the ERPs will be conducted before the plant operates at more than 5% of rated power. *See* § IV.F.1.b of Appendix E in 10 C.F.R. Part 50. But neither regulations nor guidance set out any deadline for the tests other than operation above 5% of rated power. Thus, that the ERPs for the Harris Plant have not been tested yet raises no litigable issue.

Moreover, as the ERPs make clear, some of the annual exercises will be conducted in adverse weather, though no explicit mention is made of conducting them during tornadoes; some exercises will be conducted between midnight and 6 a.m.; and some will even be unannounced. *See* §§ VII.A.2-4 of the county ERPs. However, NRC regulations prudently rule out mandatory evacuation of the plume EPZ, an area of well over 300 square miles.

In our rulings on Contentions 81 and 208, we have taken the contentions at face value, as being about the planning for the exercises, not their results. However, the contentions, especially 208, may be attempting to reserve a right to file contentions on the results. Under the Commission's view of 10 C.F.R. § 50.47(a)(2), results of the exercises are not necessarily litigable in these hearings, but § 50.47(a)(2) was declared invalid by the D.C. Court of Appeals in *Union of Concerned Scientists v. NRC*, 735 F.2d 1437 (D.C. Cir. 1984). The regulation is still in effect while the Commission's petition for rehearing is before the Court, but if the Court's May 25 ruling becomes law, the Intervenors will have a chance to file contentions on the results of the exercises.

PUBLIC EDUCATION AND INFORMATION: Eddleman Contentions 227, 228, and 229

These three contentions have largely to do with the emergency preparedness brochures mentioned in § IV.D.2.a of the State ERP. We defer ruling on 227 and reject the other two contentions.

Contention 227 alleges that the brochure is not available yet and that the brochure therefore does not contain the information called for in §§ II.G.1.a-d of NUREG-0654. The brochure is now available. Its adequacy, the second issue 227 raises, is litigable, and has been litigated, most recently in *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-753, 18 NRC 1321, 1331 (1983), *aff'g the detailed findings of* LBP-83-27, 17 NRC 949 (1983). Therefore, as we did with CHANGE 2 (at Tr. 967), we defer ruling on 227. In accordance with the 30-day rule in this proceeding, and the discussion in the telephone conference of July 12, 1984 (Tr. 2203), Mr. Eddleman and the other Intervenors have until August 10, 1984, to file revisions of their contentions on the brochure, specifying the respects in which the brochure is inadequate, and why.

Contention 228 alleges that the Applicants must demonstrate that the information called for by §§ II.G.1.a-d of NUREG-0654, and slated for the brochure, will be made available periodically to the public. We reject this contention. It merely paraphrases planning standard (b)(7) of 10 C.F.R. § 50.47 and evaluation criterion II.G.1 of NUREG-0654. The contention doesn't address any provision of the ERPs and thus could not, and does not, allege any deficiencies in the ERPs. In fact the State plan provides means for making the relevant information "available to the public on a continuous basis." Section IV.D.2 of the State ERP. Among the means is annual dissemination of emergency preparedness brochures. *Id.*

Neither the syntax nor the intent of Contention 229 is easy to construe, but the contention appears to allege that the planning standard on public education and information (subsection (b)(7) of 10 C.F.R. § 50.47) and the evaluation criteria under that standard (§ II.G of NUREG-0654) cannot be met unless the ERPs provide means to verify that the public has received and understood the education made available to it. We reject 229. The planning standard and evaluation criteria the contention cites do not call for any program of verification. Rather, their emphasis is on making the information readily available. To this end, the cited standard and criteria call for a variety of means of disseminating information and a high degree of involvement in the disseminating by State and local response organizations. The contention cites, but hardly addresses, the ERP provisions which are meant to conform to the cited standard and criteria. Thus the contention provides no basis for thinking that the provisions might fall significantly short of assuring that the public will be adequately educated. Such variety and involvement as the ERPs provide for appear to have such a high probability of successfully informing the public that a program of verification would be only marginally useful at best.

INGESTION EPZ:
Eddleman Contention 206

This contention alleges that the ERPs do not provide for sheltering milk animals and placing them on stored feed during a site emergency or a general emergency, contrary to the guidelines in Appendix 1 of NUREG-0654 at 1-12, 1-16. We reject this contention for not addressing the relevant provisions of the plans. The ERPs provide both for placing cattle on stored feed (*see* § IV.F.5.b of the State ERP) and for the timing of such action (*see* §§ IV.E.2.b, IV.E.4 and IV.F.4 of the State ERP). These provisions appear to conform to evaluation criterion II.J.9 of NUREG-0654 (except that they cite a revision of the FDA guidance cited by the criterion). Although the criterion and the plan provisions meant to conform to it are not presented in the graded emergency level format of the pages the contention cites from Appendix 1, and therefore do not say what to do during a site emergency or a general emergency, it would appear that the criterion and conforming provisions, by relying on FDA recommendations, implicitly provide for the actions the Appendix is explicit about. The contention says nothing to the contrary. We note that NUREG-0654 nowhere speaks of sheltering animals.

SIGNATURES AND MEMORANDUM OF UNDERSTANDING:
Eddleman Contentions 57-C-18 and 200

These two contentions allege that the ERPs are incomplete because they do not contain the Memorandum of Understanding between the State and the Applicants (57-C-18) and because the signature pages (at iii-iv) are not filled out (200). The contentions conclude that therefore there is no assurance the plans can be implemented.

We reject both of these contentions. They proffer no bases for thinking that the final form of the plans will not contain the Memorandum and signatures. To the contrary, the intent of the planners to include these items is clearly shown by the inclusion in the ERPs of pages marked as being reserved for these items. Moreover, the existence in the plans of letters of agreement between the county emergency management agencies and the Applicants (*see* Attachment 1, at 1-3, in each county ERP), and between Carolina Power and Light Company and the Radiation Protection Section of the State's Department of Human Resources (Attachment 1, at 1-29, of the State ERP), indicate that there are no significant obstacles in the way of drafting the Memorandum and acquiring the signatures.

IMPLEMENTING PROCEDURES: Eddleman Contention 213-a

This contention alleges that since the ERPs do not contain implementing procedures, they do not contain sufficient information about how they will be implemented, and thus violate the requirement in 10 C.F.R. § 50.47(a)(2) that there be reasonable assurance they can be implemented.

We reject this contention as it stands, but there is within it, as there was within Contention 57-C-7, something like a lesser-included contention, which we admit. First, NRC regulations and guidance consider the implementing procedures to be separate from the plans. Section V of Appendix E to 10 C.F.R. Part 50 sets out requirements applicable to a separate submission of the implementing procedures for the onsite plans. Evaluation criterion II.P.7 of NUREG-0654 calls for the titles of the offsite implementing procedures, not the procedures themselves, to be listed in an appendix to each offsite plan. As we've noted before, NUREG-0654 says that the average plan "should consist of perhaps hundreds of pages, not thousands." NUREG-0654, Appendix 1, at 1-29.

Second, a finding that there is reasonable assurance that the plans can be implemented is, under the regulation the contention cites, 10 C.F.R. § 50.47(a)(2), to be based largely on the plans, not the myriad details of the implementing procedures: § 50.47(a)(2) says that the NRC will base its finding on FEMA findings, and that "a FEMA finding will primarily be based on a review of the plans." Implementability is a characteristic of good *plans*, for even the best implementing procedures cannot rescue an ill-conceived plan. Thus it is to the adequacy of planning that all of the Commission's planning standards and evaluation criteria are directed, and it is the adequacy of planning that we're after in this proceeding. The mechanical details implementing procedures largely consist of are almost never suitable for litigation. Contention 213-a points to no plan provision drafted in such a way that we would have to look at the implementing procedures under it to determine whether there was reasonable assurance it could be implemented.

Last, however, 213-a is admissible in one respect: stated so that it does not, in effect, attack the regulations, 213-a says that the plans should incorporate the implementing procedures to whatever extent called for by regulations or guidance. There are bases for admitting 213-a phrased this way: as we noted above, evaluation criterion II.P.7 calls for each plan to have an appendix which lists implementing procedures by title. None of the offsite plans for Harris have such an appendix. Annex H, the Plan Cross-Reference, cites certain page numbers in each plan as containing material tailored to criterion P.7, but all the citations are to sections entitled "Concept" or "Concept of Operations."

Judging from the Foreword to the ERPs (at vii), we imagine that the Applicants' argument against admitting 213-a as we've just construed it would be that criterion P.7, being guidance, does not set out a requirement, and that the goal of P.7 is met by the present form of the ERPs, namely, five parts consisting of — in the words of the Foreword — detailed "State procedures" and "county procedures," "additional detail" in several annexes, and "the existence of emergency procedures at the State and local levels." Foreword to the ERPs at vii. Thus "separate implementing procedures are not deemed necessary" (*id.*), and, the argument might conclude, *a fortiori*, that an appendix listing unnecessary procedures by title is not necessary.

However, it does not appear that the ERPs are — or, given their length, could be — detailed enough to be implementing procedures, though they are, of course, in a more generic sense, "procedures." Moreover, though Annexes C-G are quite detailed, they deal only with notification. Last, if the emergency procedures the Foreword says already exist at the State and local levels have, in fact, the character of implementing procedures, then criterion P.7 calls for a list of them in appendices to the plans. Presumably the goal of P.7 is to assure not only that the implementing procedures are prepared in advance of plant operation above 5% of rated power, but also to assure coordination between the plans and the implementing procedures. Thus P.7 also calls for the appendices to list for each procedure the plan section it implements.

In sum, 213-a is admitted in the following form: either each offsite ERP should contain an appendix which conforms to evaluation criterion II.P.7 of NUREG-0654, or it should be demonstrated that such an appendix is unnecessary because its functions are performed in some other way by the present form of the plans.

PLAN MAINTENANCE; IDENTIFICATION OF LOCATIONS OF CERTAIN PERSONS AND INSTITUTIONS:
Eddleman Contentions 99 and 209

Contention 99, originally filed May 14, 1982, and now resubmitted unchanged, is confusingly drafted. Given its opening lines and the regulations it cites, one could reasonably conclude, as did the Applicants and the Staff, that 99 means to allege that the plans, both onsite and offsite, do not contain provisions for keeping the plans up to date, especially for keeping up to date information such as the locations of day-care centers, schools, disabled persons, emergency personnel, and the like. But one could also reasonably conclude that 99 means to say primarily that the listed categories of information should be in the plans, and secondarily that the information be up to date. This latter reading of 99 is suggested by Contention 209, which alleges that, "with a handful of exceptions," none of which 209 states, the information asked for in 99 still isn't in the plans.

We reject both contentions. In relation to the onsite plan they are filed too late, and in relation to the offsite plans they are without bases: they do not address the plan provisions on updating, § VII.F of the State ERP and §§ VII.D of the county ERPs; and the regulations 99 cites, 10 C.F.R. § 50.54(t) and § IV.G of Appendix E to 10 C.F.R. Part 50, apply only to the onsite plan. We note that the plan provisions on updating appear to conform to the applicable planning standard, 10 C.F.R. § 50.47(b)(16). Moreover, though 209 says that some of the information requested in 99 is still not in the plans, it does not say what information is not. It is therefore lacking in specificity.

SITE-SPECIFIC PLANNING:
Eddleman Contention 242

This contention alleges that occasional references in some of the ERPs to North Carolina nuclear power plants other than Harris, and North Carolina counties other than those which overlap the Harris plume EPZ, indicate that the site-specific planning required by various NRC regulations has been compromised — that "the SHNPP plan is a copy of the McGuire plan," and that officials around SHNPP "have not seen the plan yet or they surely would have caught these errors." The contention cites two such references, one in § IV.D.1 of the Chatham plan, at 26, and the other in § VI.D.1 of the same plan, at 42.

We reject this contention. A serious contention alleging failure to tailor plans to the particularities of the Harris site would have to show, for example, that the ERPs for Harris did not adequately take into account particularities of the Harris site, such as the organization of county governments around the plant, or the capacity of the road system around the plant. We might be concerned if one of the county plans simply copied a list of shelters or county agencies from the McGuire plan. But, as it is, all the contention suggests is that, in an attempt either to keep the plans for different North Carolina plants as parallel as possible, or simply to save time and effort, certain names have been repeated by mistake. Indeed, it would be surprising if the drafts-people of a new plan did not at least consult previously approved plans for other plants in the area.

ONSITE EMERGENCY PLANNING: Eddleman Contentions 151, 157, 103, and 137

These four contentions cover various aspects of the Applicants' onsite ERP. Contentions 151 and 157 were submitted on May 2, 1983, in response to the filing of the onsite plan on March 29, 1983. On November 1, 1983, we deferred ruling on these two contentions until the parties had had the opportunity to comment on certain documents we asked the Applicants to file in connection with the deferred contentions. *See* our Memorandum and Order, November 1, 1983 (unpublished), slip op. at 4, 6; and Tr. 778. The Applicants filed the documents in February 1984; and on April 3, 1984, Mr. Eddleman filed amendments to the deferred contentions. We now rule on them.

In its original form, 151 alleged that the onsite plan did not conform to 10 C.F.R. Part 50, Appendix E, § IV.E.4, which requires the onsite plans to make and describe "arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies onsite." On February 1, 1984, the Applicants served on the Board and the parties a letter of agreement between Carolina Power and Light and three physicians for services in a radiation emergency. Thus, the onsite plan now conforms to the regulation Contention 151 cites.

Nonetheless, Mr. Eddleman submitted an "amended" 151. It is, however, simply a new contention. It alleges that "it is not clear" either that the three physicians will be adequately trained, or that they are bound "to stay in the area near Harris" and, more generally, "bound by their agreements in the future." We reject amended 151. It offers no reason to think that the physicians' training might be inadequate, or that the agreement with them is not binding. We note that the agreement commits Carolina Power and Light to bear the costs of training the physicians. Last we cannot imagine that such a letter of agreement could bind the three signers to remain in the area of the Harris site for the life of the plant. In time, the duties of one or more of them will probably have to be assigned to others. These reassignments are provided for in §§ 5.1.1 and 5.1.2 of the onsite plan, which name the officers responsible for negotiating and maintaining letters of agreement.[7]

Contention 157 alleges that the onsite plan does not comply with NUREG-0737, Supplement 1, § 8.2.1.k, which requires that the design of the Technical Support Center (TSC) take "into account good human factors engineering principles." The principal basis of the contention originally was simply that the onsite plan gave no analyses of any human factors engineering in the TSC.

On February 17, 1984, the Applicants filed with the Board and the parties an eight-page document entitled "Summary of Design Standards and Criteria for the TSC Encompassing Human Factors Engineering," to which is attached a "furnishings plan" precise to the level of waste bins and coatracks. Despite the discussions in this document of such human factors topics as layout, noise control, instrument displays, and protective systems, Mr. Eddleman chooses to ignore the document in his "amendments" to 157. In them he does little more than assert that a TSC must be able to function in a real emergency. A contention which pays no attention to the principal document on its subject, a document drawn up for the sake of this proceeding, must be rejected.

Contentions 103 and 137 were first submitted in 1982 on May 14 and June 6, respectively. We deferred ruling on them because the onsite plan had not yet been filed. *See* our Memorandum and Order, September 22, 1982, LBP-82-119A, 16 NRC 2069, at 2105, 2109. Now, although we had ordered that new contentions on the onsite plan had to be filed, or old ones resubmitted or amended, within 30 days of receipt of the plan (*see id.* at 2073), 103 and 137 have been resubmitted, unchanged, a year after the onsite plan became available. Mr. Eddleman does not explain why contentions as tardy as these should be admitted. The lateness of 137 is accentuated by its allegation that the "Applicants' site emergency plan is inadequate because it does not exist." We reject 137.

Contention 103 alleges that the onsite counting laboratory is not shielded from radiation well enough to assure that analyses of primary coolant can be done quickly enough for a timely declaration of a level of

[7] The Applicants claim that the letter of agreement is signed by the physicians in their capacity as officers of the corporation named in the letterhead, and that therefore the agreement would survive even if one of the signers left the area permanently. Applicants' Answer at 98. However, the only support for the Applicants' claim is the letterhead.

emergency. Not only is this contention a year late, it proffers no factual basis for its claim. We therefore reject it.[8]

MAPS:
Eddleman Contentions 211, 250, 251, 252, 253, and 254

Up to now, we have been considering contentions Mr. Eddleman filed or resubmitted in April of 1984. Five of the six contentions we're about to rule on, 250-254, were filed on May 10, shortly after the prehearing conference, with our leave.

Contention 211 was filed before the other five. It alleges that the offsite plans do not include the operations and ingestion pathway maps called for by evaluation criteria II.J.10.a and b of NUREG-0654. During the prehearing conference, the Applicants claimed that the operations map was already in Annex H of the onsite plan and merely had to be moved to the offsite plan (Tr. 1000-01), and that since the map had been available since the onsite plan had been filed, any contention on the map was late-filed. Tr. 904, 905, 1107. Nonetheless, without deciding the timeliness issue, we gave leave to certain intervenors, including Mr. Eddleman, to file contentions on the map as soon as possible. Tr. 906, 1106-07. Below we briefly consider the timeliness issue but move on to consider all six contentions on the merits, rejecting all of them, but two only conditionally.

The Applicants' argument that these contentions are inadmissibly late-filed is principally that the map or maps which will be included in the offsite ERPs are already in the onsite plan in a form in which State and local government agencies have concurred and thus have been available to the Intervenors since late March of last year.[9] However, even the Applicants were at one point mistaken about whether the maps were available yet. Before the prehearing conference last May, the Applicants argued in response to Contention 24 that the "Operations Map" was under development and was expected to be completed by September. Applicants' Response at 90. It wasn't until the prehearing conference

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[8] The Staff argues that the contention "shows Mr. Eddleman's fundamental misunderstanding of the NRC's emergency planning structure. Emergency action levels are determined without taking a sample of reactor core water . . . ." Staff's Response at 66. However, it would appear that emergency action levels can be determined by such a sample, though not necessarily. *See* the onsite plan, Figure 4.1-1, Basic Module 2.

[9] Mr. Eddleman in one place speaks as if the Applicants made a mistake to put the maps in the onsite plan. *See* his May 10, 1984, Response at 1. However, the very evaluation criteria on which Mr. Eddleman relies in these contentions, namely II.J.10.a and b of NUREG-0654, call for these maps to be in the onsite plans as well as the offsite.

that the Applicants began to argue that the same map was already available and could be found in Annex H of the onsite plan. Tr. 1000-01.

There is something to be said on both sides of the "lateness" question, which is a close one. In any event, we need not decide the lateness question, for all six of the "map" contentions are rejectable on the merits, and some are not vulnerable to attack on grounds of lateness. We discuss first those we reject unconditionally.

Contention 252 alleges that it is "just unfathomable" why the parts of plume EPZ sub-areas B and C which jut into sub-area A, which includes the Harris site, are not included in sub-area A, and that they should be, "to assure protection of any persons in those areas in an accident." The contention is without bases. The contention suggests that people in sub-area A would receive greater protective actions than those in other sub-areas, and that sub-areas should be arranged as concentric rings or parts of such rings. However, there is no less planning for sub-areas B and C than for sub-area A. For each sub-area, the aim of planning is the same: that adequate protective measures be taken in an emergency. Thus, although it is conceivable that sub-area A would be evacuated and sub-areas B and C would not, there is no indication that if the greatest dose-savings for people in sub-areas B and C could be achieved by a given protective measure, that measure would not be taken, whether or not the same measure were taken in sub-area A.

Moreover, NRC guidance does not suggest that the sub-areas are to be concentric rings, or parts thereof, any more than that the EPZs themselves should be exactly 10 or 50 miles in radius. "The boundaries of the sub-areas shall be based upon the same factors as the EPZ, namely demography, topography, land characteristics, access routes, and local jurisdictions." NUREG-0654, Appendix 4, at 4-4. As we noted at Tr. 982, State and local planning officials are not obliged to supply a written justification of their boundary-making until they are faced with an admitted contention on the subject.

Contention 254 is analogous to 252. It alleges that the areas within 10 miles of the Harris site but not in the plume EPZ have been excluded from the plume EPZ without justification. The contention points to two such areas but does not try to justify including them in the plume EPZ. The contention is without bases. The regulation on the size of the plume EPZ says that it shall be "about" 10 miles in diameter, not "at least." Again, the burden rests initially on an intervenor to argue why a given area should be in the plume EPZ. Only then are planning officials required to justify the exclusion. Contention 254 does not meet this initial burden. We note, however, that the Applicants have nonetheless offered justifications for the two exclusions the contention notes. *See* Applicants' May 29, 1984, Response to Eddleman Map Contentions at 20 n.8. Besides noting the political and geographical boundaries which delineate the plume EPZ in the two areas the contention points to, the Applicants claim that the excluded areas are "essentially unpopulated." *Id.*

Contention 253 alleges that the plans are deficient in routing some of the evacuees in sub-areas E, F, and G toward Raleigh, because the prevailing winds at Harris are in that direction. The contention also alleges that evacuees should not be routed along the stretch of NC-55 which is outside sub-area G but roughly parallel to G's eastern boundary, for evacuees on this route would be exposed for 3.1 miles to plumes in prevailing winds.

We reject this contention as being without basis, but not on grounds of the Applicants' argument, which, we think, is unsound. The Applicants have argued before, and now argue again, that people will not be directed to evacuate at the same time radioactivity is being released. Applicants' May 29, 1984, Response to Eddleman Map Contentions at 18. For support, the Applicants cite § IV.A.4 of the State ERP: evacuation would be the chosen protective action only if evacuation could be "completed prior to significant release and arrival of radioactive material in the affected area." However, the word "significant" in this passage is important. The passage does not rule out evacuation during any release. The point of protective measures is dose savings, and under some possible scenarios greater doses would be saved by evacuating for 1 or 2 hours than by sheltering for several.[10]

We do agree with the opinion expressed in a case cited by the Applicants: "With significant shifts in wind direction always a possibility during the course of any evacuation, it would seem impractical and possibly imprudent to preselect evacuation routes based on potential wind direction." *Metropolitan Edison Co.* (Three Mile Island Nuclear Station, Unit 1), LBP-81-59, 14 NRC 1211, 1588 (1981).

However, our principal reason for rejecting 253 is that it fails to address the evacuation routes in their full context. They are not simply routes out of the plume EPZ, they are routes to public shelters. Many evacuees from sub-areas E, F, and G are routed to Raleigh because it contains the public shelter most accessible to them. Moreover, other sub-areas are assigned to shelters more accessible to them than Raleigh is. Thus, if no one from E, F, and G evacuates to Raleigh, probably no one will. Thus, to assert that no evacuees from sub-areas E, F, and G

[10] Hence the importance of advance calculation of sheltering factors, the subject of admitted Contention 57-C-10.

should be routed toward Raleigh is virtually to assert that no public shelter should be located in Raleigh, even though it is a major city, well outside the plume EPZ, and accessible from E, F, and G by highways which become four-lane not far from the boundary of the plume EPZ. Only if Mr. Eddleman had shown that such an argument was admissible could Contention 253, which implies it, have a basis.

Similarly, the contention's complaint about traffic on NC-55 views that traffic out of context also. It is easy to find many more examples of the same sort of routing. To take the most striking example, traffic on NC-751 in the eastern part of sub-area N is routed from the boundary of the plume EPZ back *in* toward the plant, for what appears to be 2.4 miles. The apparent explanation is that in order to reach their shelters in Siler City and Raleigh, evacuees on NC-751 must head south to US-64. Similarly for evacuees on the stretch of NC-55 which parallels the eastern boundary of sub-area G: once the evacuees who head southeast out of G reach NC-55, they must turn north to reach US-401, the fastest route to their shelter in Raleigh. Besides, the stretch of NC-55 the contention is concerned about is *outside* the plume EPZ.

The remaining map Contentions, 211, 250, and 251, at first appear to be about the map itself rather than the planning the map embodies. Contention 250 alleges that the map doesn't comply with evaluation criterion II.J.10.a of NUREG-0654 because it does not show the location of relocation centers and shelter areas, and is "virtually illegible." The contention might have added that the map does not show the location of preselected sampling and monitoring points either, though these too are called for by the same criterion. Contention 251 argues analogously about evaluation criterion II.J.10.b of NUREG-0654, that the map doesn't show population by evacuation areas, though the criterion calls for such a showing. The contention might also have said that the map does not show population by 22½° sectors, though this too is called for by II.J.10.b. Contention 211 contains virtually the same allegations, but since it was filed before the prehearing conference, it bases the allegations not on the map but on the absence of any map in the plans. Contention 211 is thus superseded by Contentions 250 and 251, and therefore requires no further consideration.

Though 250 and 251 are phrased as contentions about the map, they are actually about the offsite ERPs, as becomes clear when they are stated thus: If this map is the only one which will be in the map annex of the offsite ERPs, Annex I, then the plans will not conform to criteria II.J.10.a-b. Thus, 250 and 251, being about the offsite ERPs, are not vulnerable to attack on lateness grounds. Even the allegation of illegibility, which, more than any other of the allegations in 250 and 251, appears to be about the map, is about the plans, for only at the prehearing conference did it become known that the operations map in Annex I of the offsite plan was to be a copy of the arguably hard-to-read map in Annex H of the onsite plan. As we show below, the Applicants' response to these contentions is not altogether clear and in its present form invites unnecessary litigation. We try to avoid this litigation by asking the Applicants for another filing.

Until the prehearing conference last May, it appeared that the Applicants were committed to putting into Annex I of the offsite plans maps which included all the information called for in the criteria which Contentions 250 and 251 cite — §§ II.J.10.a and b of NUREG-0654. Annex I contains a page which says that operations and ingestion pathway maps will be available later, the implication being that they will appear in Annex I. The Applicants' April 28, 1984, Answer to Contention 211 appeared to affirm that such maps would be in the plans, for, among other things, the Answer said that "a commitment has been made that the provisions of NUREG-0654 [referring to §§ II.J.10.a and b] will be met," and the Answer quoted those provisions. *See* Applicants' Answer at 89-90. Had the Applicants at that point simply said that all that remained to do was to make legible copies of certain maps in the onsite plan and place the copies in the offsite plan, there would have been no, or little, occasion for 250 and 251, for as the Applicants pointed out then (and again in their response to 250 and 251), all the information called for by §§ II.J.10.a and b is in maps in the onsite plan. One could have wondered only whether they intended to include the ingestion pathway map promised by Annex I. They argued that §§ II.J.10.a and b did not, on their faces, call for such a map, but they did not say they would not follow through on the promise in Annex I to include an ingestion pathway map.

Now, however, the Applicants could be read to be arguing that the map in Annex H of the onsite plan, which contains only some of the information called for by §§ II.J.10.a and b, is all that must, or will, appear in Annex I: In their response to 251 and 252, they argue that all that remains to be done is to put a copy of the Annex H map into Annex I. They also argue that Contention 251, by not calling for population by 22½° sectors, "apparently concedes" that such information is not expected to be in the offsite plans. We suppose also that the Applicants would still argue that §§ II.J.10.a and b do not call for any map of the ingestion pathway to be in the offsite plans.

We do not understand why the Applicants have apparently backed away from their earlier commitment to follow §§ II.J.10.a and b. We do not find persuasive their arguments that certain map information needn't be in the offsite plans. Contention 251 does not concede that population by sector need not be in the offsite plans. Indeed, 251 quotes the criterion which says such information should be in the offsite plans. Also, we do not agree that § II.J.10.a does not call for at least one ingestion pathway map. It calls for showing the locations of relocation centers and shelter areas, and, as the Applicants themselves point out, that information cannot be placed on a map of the plume EPZ. Applicants' Response to Eddleman Map Contentions at 12 n.5. The Applicants, in their response to 250 and 251, resist Mr. Eddleman's insistence that the information be not merely available but *in* the plans. However, his insistence is arguably in accord with the distinction in § II.J of NUREG-0654 between maps which are to be in the plans (*see* §§ II.J.10.a and b), and those to which the plans need only refer (*see* § II.J.11).

Litigation over what maps are and are not to be in the offsite plans — a purely mechanical question — can and should be avoided: We reject Contentions 250 and 251 on the condition that the Applicants reaffirm in writing their April 28 commitment (at 89-90 in their Answer) to include in Annex I of the offsite plan all the map information called for by §§ II.J.10.a and b, in legible form prior to fuel loading of the facility.

CCNC'S REMAINING CONTENTIONS

At the prehearing conference, we admitted parts of CCNC Contentions 2, 5 and 8. CCNC's remaining nine contentions are rejected for the reasons assigned below.

Contention 1

The contention is drafted in a rather confusing manner, but its thrust appears to be that, under the ERPs, evacuation decisions will be too long delayed. The contention misconceives the plans and their relationship to the Applicants' Emergency Classification System. Under that system, an evacuation recommendation need not await a full-scale emergency. Furthermore, evacuation *decisions* are to be made by the local officials, based on EPA protective action guidelines.

Contention 3

Appendix G to the ERPs reflects considerable planning for an emergency at Jordan Lake. Little, if any, more advance planning could be done. It may well take more time to evacuate Jordan Lake on a summer weekend than other parts of the EPZ. NRC regulations impose no time limit on evacuation. Local officials would have discretion, in such circumstances, to order the lake evacuated first.

Contention 4

The ERPs in fact contain a much greater communications capability than is alleged in this contention, as described in the Applicants' response.

Contention 6

CCNC may participate as a Joint Intervenor under EPJ 3.

Contention 7

The contention ignores the primary means of notification, sirens, as described in the plan sections cited in the Applicants' Response.

Contention 9

This contention challenges the adequacy of medical services. It is barred by the Commission's decision in *San Onofre, supra.*

Contention 10

This contention, like Contention 7 above, ignores the siren notification system.

Contention 11

This very broadly drafted contention lacks the requisite specificity and does not give adequate notice to the opposing parties.

Contention 12

This contention contains two basic allegations — that the EPZ is not sufficiently "rationalized" and that there should be evacuation planning for areas outside of the EPZ. Both impermissibly attack the EPZ rule, 10 C.F.R. § 50.47(c)(2). Local officials must actually consider the factors listed in the rule in drawing the EPZ boundary. However, nothing requires them to "rationalize" their work in writing. Evacuation planning is not required outside the 10-mile EPZ.

EMERGENCY PLANNING JOINT (EPJ) CONTENTIONS

At the prehearing conference, we admitted EPJ Contentions 1 (snow and ice) and 2 (evacuating people without cars). We also indicated that we would draft and admit several additional EPJ contentions in certain areas. These additional EPJ contentions are set forth below, coupled with a listing of the Intervenors who will be deemed co-sponsors of the contention and a tentative designation of a lead intervenor, at least for discovery purposes.[11] If the parties wish to designate another intervenor as the lead, they should notify the Board and parties to that effect by August 10, 1984. The contentions leading to an individual Intervenor's designation under the EPJ contention are now superseded.

EPJ 3

The number of volunteer workers — such as members of volunteer police, rescue, and fire departments — who would respond to an alert is extremely questionable; plans should be based on a response rate of no greater than 50% in organizations in which no attention has been given to composition which would avoid conflict between organizational and family responsibilities.

Similarly, present planning assumes that teachers will leave their cars and families in the area and supervise students on the bus and in the shelters. This is an unreasonable and unrealistic demand on teachers.

Co-sponsors: Dr. Wilson — 7f, 8g, 12(8)
CHANGE — 13
CCNC — 6

Lead Intervenor: CCNC (Conservation Council of North Carolina)

EPJ 4 — Evacuation of Schools

Section E.4.d of State Procedures (at 47) is deficient because —

(a) Fifty percent of school bus drivers are high school juniors and seniors (as young as 16½ years). They should not be expected to perform as emergency personnel without explicit and specific authorization from their parents. Even with such authorization they should not be trusted to perform in emergency situations.

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[11] We have not designated Mr. Eddleman as a lead intervenor during discovery because of his commitments in the safety hearing. We do not mean to preclude some lead role for him at the hearing stage.

(b) Adult bus drivers have minimal education and are paid very low wages. They cannot be trusted to put their jobs above family obligations or to perform adequately in emergency situations.

(c) In normal operation, each bus makes two runs each day. Thus, two round trips to the shelter sites would be required. (This factor was not considered in traffic control plans or evacuation time estimates). Students who do not normally ride buses will be an extra burden, requiring even more round trips.

(d) Most parents would demand to pick up their children at school. The chaos at every school in the area would require all local law enforcement officers and several county officers to contain. This factor is not mentioned in the plan.

Co-sponsors: Dr. Wilson — 8
Mr. Eddleman — 219 (last paragraph), 222 (last two sentences), 230
CHANGE — 26, 29.

Lead Intervenor: CHANGE (Chapel Hill Anti-Nuclear Group Effort)

EPJ-5 — Transportation for the Nonambulatory

Section E.4.b of State Procedures (at 47) is deficient because there is no listing or mechanism of identifying homebound nonambulatory people. Most ambulances and rescue squad vehicles are not adequately equipped to meet State standards for transporting hospitalized patients. A sufficient number of vehicles equipped adequately to transport the nonambulatory from hospitals and homes will not be available.

Co-sponsors: Dr. Wilson — 7
Mr. Eddleman — 262, 263(A)

Lead Intervenor: Dr. Wilson

For ease of reference, we include below the texts of the joint contentions admitted during the prehearing conference.

EPJ-1 — Evacuation in Snow and Ice

Insufficient consideration has been given in the offsite emergency plans to the effects of severe snow and ice conditions on evacuation times and/or capabilities to clear evacuation routes.

Section IV.E.8 of the State plan (at 50) is deficient because the State does not have enough snowplows in this area to effectively clear the roads of snow or ice in a reasonable amount of time.

Co-sponsors: CHANGE — 3, 32(1)
Dr. Wilson — 14, 12(7)
CCNC — 5

Lead Intervenor: CCNC

EPJ-2 — Transportation for People Without Cars

Section IV.E.4.e of the State plan (at 47) is deficient because it provides no estimate of the number of people without transportation, (Applicants' estimate of 240 families in evacuation time study (at 3-2) seems far too low), no suggestion as to how people without transportation would get to pickup points, and no criteria for determining when and where they would be "established as required."

Co-sponsors: Dr. Wilson — 9
CHANGE — 28

Lead Intervenor: CHANGE

RADIATION MONITORING CONTENTIONS

Applicants' Response to CHANGE 7 states that the contention "misreads the availability of State teams, ascribes a role to those teams which is not theirs alone, ignores the means available to relocate the teams, mistakenly assumes that field monitoring teams should not be required to relocate and ignores CP&L's considerable assessment capability early in an accident." The Board agrees that there is no asserted basis for this contention and admission is denied.

CHANGE 11 is redundant to CHANGE 7 and this contention has the same deficiencies. Admission is denied since no basis in terms of roles of the RPS monitoring teams in the overall emergency response is asserted.

The lack of focus and clear bases for both parts of Wilson Contention 2 were brought out at the prehearing conference (May 1, 1984), at Tr. 876-84. Admission is denied because of those deficiencies.

WILSON CONTENTIONS 6 AND 12

These are the only individual contentions on emergency planning that are still pending. Contention 6 alleges that § IV.E.4.a of the State ERP (at 47) is deficient because it calls for the use of commercial buses, and yet there are no commercial buses in the plume EPZ and no arrangements to use commercial buses from outside the EPZ. As came out at the prehearing conference, the word "commercial" has been removed from the cited section. Tr. 987. Thus, there is no need to consider this contention. *Cf.* our rejection of CHANGE 28. *Id.* Tr. 835-39.

Contention 12, which has many subparts, focuses on the evacuation time estimates. For the reasons given at Tr. 990-93, we are not treating contentions on the estimates as late-filed. As to the subparts of this contention, (b)(7) and (b)(8) have already in effect been admitted as parts of one or another of the joint contentions. Subpart (b)(4) is either a cross-reference to Wilson 8 (which is superseded by EPJ-4, or, by speaking only of "school problems," too vague to be litigated.

We admit subparts (b)(2) and (b)(3). We ourselves do not see the grounds for assuming that families with more than one car would evacuate in only the best of their cars. We would also like to know how it was estimated that only 240 families in Wake County exclusive of Raleigh are without cars.

We reject the remaining subparts of Contention 12. Briefly, 12(a) gives us no basis for doubting the State's letter of review and concurrence, found at the end of the Evacuation Time Estimates. Contention 12(b)(1) refers to the *backup* system of notification, but gives us no reason to think that the 15-minute notification assumption is unrealistic when made about the primary notification system, the siren system described in an annex of the plans. It's not clear what sort of validation of NETVAC Contention 12(b)(5) would call for other than full-scale evacuation of the plume EPZ. Moreover, though (b)(5) says there is no reason to accept the model's predictions, (b)(5) does not address the many reasons proffered by § 2 of the Estimates. Subpart (b)(6) does not address the plans. Sections V.5.b-e of the State ERP clearly subordinate decontamination to the need to evacuate quickly. As to 12(b)(9), we know of no requirement that the Estimates discuss alternatives to NETVAC, and (b)(9) doesn't point to any defect in NETVAC. Last, 12(b)(10) alleges that there is no justification given for the plotted points in Figures 7-1 to 7-3 of the Estimates, but we would assume that the points were determined by the NETVAC simulation.

DISCOVERY ON CONTENTIONS ADMITTED BY THIS MEMORANDUM AND ORDER

Discovery on the contentions we now admit is open. In the telephone conference of July 2, 1984, we established an earlier tentative schedule for discovery and summary disposition motions, on the assumption that these rulings would issue about July 20. These rulings are issuing about 2 weeks late, and we are adjusting the schedule to compensate for that, as follows:

Discovery Opens	August 2, 1984
Last day for filing discovery requests	October 8, 1984
Last day to respond to requests	October 31, 1984
Last day to file summary disposition motions	December 21, 1984

We are adopting the foregoing schedule on a tentative basis. Any party who wishes to request changes should file a proposed change and a brief statement of the reason for it by August 13, 1984. Bear in mind that, as the Board stated in the telephone conference (Tr. 2200-01), there will be no tolling of the times for discovery on emergency planning because of the safety hearings.

PETITION FOR WAIVER OF NEED-FOR-POWER RULE

On June 30, 1983, Mr. Eddleman filed a "Petition Under 10 C.F.R. § 2.758 Re Alternatives and Need for Power Rule." Responses in opposition were subsequently received from the Staff (August 26, 1983) and Applicants (August 31, 1983). Certain additional documents were received thereafter. The Board has concluded that Mr. Eddleman's petition must be denied. The formal order of denial, accompanied by a statement of our reasons, will be included in our Partial Initial Decision on environmental issues. We are announcing our basic conclusion on the petition at this point in order to facilitate planning by the parties for the coming months.

UPCOMING TELEPHONE CONFERENCE CALL

The Board is scheduling a telephone conference call for Friday morning, August 10, 1984, at 11:00 A.M. *This may be the only notice you will receive of the call.* (1) We expect to rule on the Applicants' motion for reconsideration with respect to Joint Contention IV; the obligation to file testimony on that contention by August 9, 1984, is suspended pending that ruling. (2) We will discuss the Applicants' motion of July 27, 1984, concerning *ex parte* extension requests. The other parties need not respond in writing to that motion; they can be heard on the telephone. (3) We will also discuss the status of Mr. Eddleman's diesel generator contentions and possible next steps in that regard, in the context of the scheduling information provided to the Board and parties by Mr. O'Neill's letter of July 31, 1984. (4) We ask the parties to look ahead to August 20, 1984, for any other matters requiring telephone discussion because the Board will be unavailable during the week of August 13.[12]

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Glenn O. Bright (by JLK)
ADMINISTRATIVE JUDGE

James H. Carpenter
ADMINISTRATIVE JUDGE

James L. Kelley, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
August 3, 1984

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[12] The Board expresses its appreciation to its Law Clerk, Steven Crockett, for his able assistance in the preparation of this Memorandum and Order.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Lawrence Brenner, Chairman
Dr. George A. Ferguson
Dr. Peter A. Morris

In the Matter of **Docket No. 50-322-OL**

LONG ISLAND LIGHTING COMPANY
(Shoreham Nuclear Power Station, Unit 1) **August 13, 1984**

The Licensing Board denies a petition, pursuant to 10 C.F.R. § 2.758(b), for exception to the regulations eliminating the financial qualifications review of electric utilities in operating license proceedings. In the alternative, the Board denies admission of an untimely financial qualifications contention. The Board also denies certification of the issue to the Commission.

RULES OF PRACTICE: WAIVER OF REGULATION

A petition for waiver or exception to the Commission's regulations, pursuant to 10 C.F.R. § 2.758(b), should only be granted in "unusual and compelling circumstances." *Northern States Power Co.* (Monticello Nuclear Generating Plant, Unit 1), CLI-72-81, 5 AEC 25, 26 (1972).

FINANCIAL QUALIFICATIONS: WAIVER OF REGULATION

In order to show that the rule precluding consideration of a utility's financial qualifications in an operating license proceeding should be waived in a particular proceeding, the party petitioning for waiver must show that the electric utility cannot recover its costs through the ratemaking process. Proposals to disallow a portion of a utility's costs are not a sufficient basis for the waiver of regulations because the outcome of such proposals is speculative.

FINANCIAL QUALIFICATIONS: WAIVER OF REGULATION

Absent evidence that a State rate commission is systematically denying a utility recovery of its costs, disallowance of construction-related costs is not an appropriate basis for waiving the financial qualifications regulations in an operating license proceeding.

FINANCIAL QUALIFICATIONS: WAIVER OF REGULATION

A party seeking waiver of the financial qualifications regulations must make a *prima facie* showing that the utility has been denied recovery of costs for safe plant operation.

CONTENTIONS: LATE-FILED

Good cause for the late filing of a contention, which is based on a recently issued document, does not exist when the information contained in that document was publicly available at an earlier date.

CONTENTIONS: LATE-FILED

With regards to the standards for the admission of a late-filed contention, a party cannot assist in the development of a sound record unless the contention presents a significant, triable issue.

MEMORANDUM AND ORDER
DENYING SUFFOLK COUNTY AND THE STATE OF NEW YORK MOTION FOR EXCEPTION FROM REGULATIONS PRECLUDING FINANCIAL QUALIFICATIONS CONTENTION AND MOTION FOR CERTIFICATION TO THE COMMISSION

I. BACKGROUND

At the July 5, 1984 prehearing conference, this Board established a schedule for hearings on the only issue still pending before us — the reliability of the emergency diesel engines.[1] Discovery has already been completed in this proceeding. The hearing will commence September 5. The other Shoreham Licensing Boards are even further along procedurally. The Board chaired by Judge Miller began hearings on the issue of emergency power sufficient for low-power testing on July 30. Those hearings were completed on August 7 with the exception of possible hearings on one sub-issue. The Board chaired by Judge Laurenson, which is hearing offsite emergency planning issues, is expected to complete its hearings in August.

On July 3, Intervenors Suffolk County and the State of New York[2] filed the following financial qualifications contention, pursuant to 10 C.F.R. § 2.714 of the Commission's regulations.

> (a) that Long Island Lighting Company ("LILCO") is not financially qualified to engage in the activities authorized or to be authorized by the operating license (including a "low power" license) which LILCO is seeking for the Shoreham Nuclear Power Plant ("Shoreham"), in accordance with the Commission's regulations; (b) that LILCO has failed to demonstrate that it possesses the financial qualifications to carry out, in accordance with the Commission's regulations, the operation of the Shoreham plant; and (c) that LILCO has failed to demonstrate that it possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for Shoreham plus the costs of permanently shutting the facility down and maintaining it in a safe condition.

Since Commission regulations preclude a financial qualifications review of an electric utility in an operating license proceeding,[3] Intervenors

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[1] This schedule is also set forth in the Board's confirmatory order of July 17, 1984 (unpublished), slip op. at 6.

[2] New York is participating as a governmental party pursuant to 10 C.F.R. § 2.715(c). For ease of reference we will refer to the County and the State as "Intervenors" proposing the financial qualifications contention.

[3] Section 2.104(c)(4) of 10 C.F.R. states that "the issue of financial qualifications shall not be considered by the presiding officer in an operating license hearing if the applicant is an electric utility." *See*
(Continued)

have petitioned that an exception be made to those regulations, pursuant to 10 C.F.R. § 2.758(b).

Section 2.758(b) permits exception to a regulation when application of the regulation to a particular proceeding would not serve the purpose for which the regulation was adopted. Intervenors assert that the application of the financial qualifications regulations to this proceeding would serve "no purpose" and that "LILCO's impending financial collapse" undermines the basic presumption behind these regulations: "the assumption that a public utility has the financial strength to engage in the activities for which it seeks a license from the Commission."[4] In support of this assertion Intervenors have filed the affidavit of Michael Dirmeier. Intervenors also request that this Board certify the issue to the Commission, pursuant to 10 C.F.R. §§ 2.718 and 2.730, if it should deny the petition for exception.

Both LILCO and the NRC Staff oppose admission of Intervenors' contention. Both assert that it is inexcusably late and that Intervenors have not shown that the balance of factors for admitting a late contention weigh in Intervenors' favor. LILCO further believes that the petition for exception should be denied because Intervenors have failed to make a *prima facie* showing that the rules would not, under special circumstances in this proceeding, serve the purpose for which they were intended. LILCO also opposes certification of the issue to the Commission.

For the reasons stated herein, this Board finds that Intervenors have not made a *prima facie* showing that application of the financial qualifications regulations to this proceeding would not serve their purpose. In addition, we find that Intervenors' motion is inexcusably late and that the balance of factors do not weigh in favor of admission of the contention, even if an exception were permitted. We further find it unnecessary to certify the issue to the Commission, and deny Intervenors' motion to that effect.

II. JURISDICTION

Intervenors have filed their petition before this Board and the Licensing Board chaired by Judge Miller. The Miller Board was established on

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also 10 C.F.R. Part 2, Appendix A, § VIII, and 10 C.F.R. §§ 50.33(f), 50.40(b) and 50.57(a). These regulations remain in effect for operating license applications until the Commission finalizes the new rule eliminating the financial qualifications review. Financial Qualifications Statement of Policy, 49 Fed. Reg. 24,111 (June 12, 1984).

[4] Memorandum in Support of Motion of Suffolk County and the State of New York for Leave to File a Contention on LILCO's Financial Qualifications to Operate Shoreham, for an Exception from Commission Rules, and for Certification to the Commission [hereinafter Intervenors' Memorandum] at 23.

March 30, 1984, solely to hear and decide LILCO's "Supplemental Motion for Low Power Operating License," dated March 20, 1984. *See* Notice, 49 Fed. Reg. 13,611 (April 5, 1984). The subject of that motion is LILCO's proposal to provide backup emergency electrical power sufficient to support low power operation without the need for the emergency diesel generators (EDGs). The issue of the reliability of the Shoreham EDGs is pending for litigation before this Board. The question of whether the Commission's rule precluding the consideration of financial qualifications as a prerequisite to issuance of an operating license should be waived in the case of Shoreham does not arise out of LILCO's supplemental motion for low power.

The Miller Board was not granted jurisdiction to hear all issues that could affect the decision of whether a low power license should be authorized. Rather, as just described, it was established only to hear and decide issues relating to the acceptability of LILCO's proposal to provide emergency electrical power without reliance on the EDGs.[5] This Board possesses residual licensing board jurisdiction over operating license issues not otherwise delegated to either the Miller Board, or, in the case of emergency planning issues, to the Board chaired by Judge Laurenson. Accordingly, we have jurisdiction to rule on the County's petition, filed under 10 C.F.R. § 2.758, for an exception to or waiver of the Commission's rule precluding litigation of the financial qualifications of LILCO to operate Shoreham. The Miller Board agrees that this Board is the proper one to rule on the County's petition for an exception.

III. PETITION FOR EXCEPTION TO FINANCIAL QUALIFICATIONS REGULATIONS

Section 2.758(b) of the Commission's regulations permits a regulation to

> be waived or an exception made for the particular proceeding. The sole ground for petition for waiver or exception shall be that special circumstances with respect to the subject matter of the particular proceeding are such that application of the rule or regulation (or provision thereof) would not serve the purposes for which the rule or regulation was adopted.

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[5] Whether any questions involving LILCO's financial situation are relevant to consideration of LILCO's proposal for emergency electrical power from sources other than the EDGs, because the proposal involves a request for waiver of a General Design Criterion, is not a matter before us. That issue is properly before the Miller Board, and has been pursued before that Board by separate pleadings from the parties.

An affidavit which specifies the specific aspect of the proceeding as to which application of the rule would not serve its purpose must be submitted with the petition. *Id.* Special circumstances justifying the waiver or exception should be stated with particularity. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 NRC 2069, 2073 (1982). If a licensing board finds that a petitioner has made a *prima facie* showing that the regulation should be waived or an exception granted, the question is then directly certified to the Commission. 10 C.F.R. § 2.758(d). The petition for waiver or exception should be granted only in "unusual and compelling circumstances." *Northern States Power Co.* (Monticello Nuclear Generating Plant, Unit 1), CLI-72-81, 5 AEC 25, 26 (1972).

Intervenors assert that LILCO's current financial difficulties constitute "special circumstances" warranting waiver of the financial qualifications regulations in this proceeding. Intervenors' Memorandum at 23. As proof of LILCO's "dire financial straits," Intervenors point to (1) LILCO's cash shortage (Dirmeier Affidavit at 8); (2) the fact that "[n]either Moody's, Standard & Poor's Corporation, nor Duff & Phelps considers any of the Company's securities to be of investment grade" (*id.* at 9); (3) the institution of a prudency investigation by the New York Public Service Commission (PSC) and the associated $1.8 billion proposed disallowance of Shoreham-related construction costs[6] (*id.* at 10); and (4) the possible acceleration of $500 million in outstanding debts related to the Nine Mile Point default (*id.* at 13). From these circumstances Intervenors conclude that "it cannot be determined that LILCO is financially qualified to operate Shoreham at any power level." *Id.* at 2.

The Commission originally proposed to eliminate the review of financial qualifications in operating license and construction permit proceedings for electric utilities in 1981. Financial Qualifications; Domestic Licensing of Production and Utilization Facilities, 46 Fed. Reg. 41,786 (August 18, 1981). This proposal was premised on the conclusions that a financial review did little to identify health and safety problems and that the regulated status of electric utilities generally assured recovery of reasonable costs. *Id.* The final rule eliminating this review was adopted in March of 1982. Elimination of Review of Financial Qualifications of Electric Utilities in Licensing Hearings for Nuclear Power Plants, 47 Fed. Reg. 13,750 (March 31, 1982).

[6] On February 10, 1984, the Staff of the New York Public Service Commission filed testimony recommending that only $2.296 billion, of an estimated overall cost for Shoreham of $4.1 billion, be included in the rate base when Shoreham becomes operational. Dirmeier Affidavit at 10.

On February 7, 1984, the Court of Appeals for the District of Columbia Circuit remanded the rule to the Commission. *New England Coalition on Nuclear Pollution v. NRC*, 727 F.2d 1127 (D.C. Cir. 1984). While the Court did not vacate the rule, it found that the rule was not adequately supported by its stated basis. In response to the Court's concerns, the Commission proposed a new rule which would eliminate the financial qualifications review only at the operating license stage. Elimination of Review of Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants, 49 Fed. Reg. 13,044 (April 2, 1984). In its June 12, 1984 Policy Statement, the Commission stated that the rules eliminating review of financial qualifications in operating license proceedings would remain in effect until the new rule was promulgated. Financial Qualifications Statement of Policy, 49 Fed. Reg. 24,111 (June 12, 1984).

The purpose of the financial qualifications regulations, applicable to electric utilities, is to eliminate Staff review of the issue in operating license proceedings on a case-by-case basis. Elimination of Review of Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants, 49 Fed. Reg. 13,044, 13,045, col. 2 (April 2, 1984). The Commission clearly stated that the basis for this exemption was that a utility's regulated status ensured that it recovered reasonable costs of operation, assuming prudent management. Costs to operate a nuclear power plant in conformance with NRC regulations are presumed to be reasonable and thus recoverable through the ratemaking process. *Id.*

The Commission's presumptions were not made in a vacuum. They rest on the line of Supreme Court cases, such as *FPC v. Hope Natural Gas Co.*, 320 U.S. 591 (1944), which allow a regulated electric utility to recover reasonable costs. 49 Fed. Reg. at 13,045, col. 2. Practical experience also supported the Commission's presumption.

> Under the financial qualifications reviews at the operating license stage conducted under the original rule, the Commission has found in every case that the state and local public utility commissions could be counted on to provide all reasonable operating costs to licensees, including costs of compliance with NRC requirements associated with safe plant operation. As a result, electric utilities applying for operating licenses have invariably been found financially qualified.

Id., col. 3.

We find that Intervenors have failed to make a *prima facie* showing that such circumstances exist in this case which would undermine the Commission's assumptions in promulgating the financial qualifications regulations. Admittedly, the Dirmeier Affidavit cites with particularity facts which reflect darkly on LILCO's financial picture. While the facts on which Intervenors rely — the Nine Mile Point default, problems in obtaining external financing and the institution of prudency proceedings — may support the contention, they are not dispositive of the petition for exception.

In order to show that the regulations should be waived, Intervenors would have to show that LILCO cannot recover its operating costs through rate regulation. Intervenors have indicated that the New York Public Service Commission has *instituted* a prudency investigation and that its Staff has *proposed* to deny $1.8 billion in Shoreham-related construction costs. Yet this proceeding has not been concluded and thus its outcome remains wholly speculative. The Commission has already expressed disfavor with speculating on the outcome of ongoing proceedings to determine the application of specific regulations to a proceeding. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), CLI-84-9, 19 NRC 1323 (1984); *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), CLI-83-17, 17 NRC 1032 (1983).[7]

Nor does this situation present issues of considerable safety significance for which a reasonable assurance now of the future outcome of the rate proceeding would be desirable. Intervenors do not allege that any particular safety problems result from LILCO's "dire" financial situation; and apparently none exist. In fact, their only fear is that "the citizens of the State and County could be faced with an irradiated plant whose owner cannot afford to operate, shut it down, or clean it up safely." Intervenors' Memorandum at 33. Although possible, it is not probable that this fear will be realized. It is unlikely that LILCO would not be found financially qualified to operate Shoreham if and when it satisfies all applicable NRC prerequisites to operation. In addition, the New York

[7] In the cited 1983 decision, the Commission disagreed with the recommendation of the Licensing Board, which included two of the members of this Board, not to permit low power testing unless and until there could be reasonable assurance that the emergency planning prerequisites for full power operation could be satisfied. Even if the Commission had agreed with the Board, the circumstances giving rise to that Board recommendation in the context of emergency planning do not apply to the subject of financial qualifications. In the former situation, the potential bar to eventual operation ran with the Shoreham facility regardless of the entity operating it. In the present context of financial qualifications, there is no basis to speculate even if Intervenors' most dire financial forecasts are realized, that the plant could not be operated in accordance with all safety requirements by either a restructured LILCO or by some other entity. This would be subject to an NRC assessment of any significant change in the entity proposing to operate the Shoreham plant (e.g., LILCO in some form of bankruptcy or a different utility operator) if and when such a proposed change is necessitated by the outcome of the State rate proceedings or other circumstances. Indeed, based on the PSC's general position (*see* note 8, below), it is more speculative to assume that *no* entity would be permitted the rate relief to cover the costs of *operation* of Shoreham than it is to assume that there would be a variety of financial arrangements which would permit some qualified entity to do so. For example, an entity not saddled with LILCO's present terms of debt service on construction funds could need a lesser degree of rate relief than LILCO would to cover its costs.

State PSC is unlikely to deny LILCO reasonable operating costs, if and when Shoreham commences commercial operation, since it does not generally do so.[8]

Nor would every denial of rate relief constitute sufficient basis for waiving the financial qualifications regulations. "When [the] NRC changed its rules, it could not have contemplated that any utility covered thereby would never have financial difficulties or that a State would never deny a utility some of the return it was seeking." *Houston Lighting and Power Co.* (South Texas Project, Units 1 and 2), LBP-83-37, 18 NRC 52, 59 (1984). To form the basis of a waiver of the regulations, the result of a State rate proceeding would have to meet the "unusual and compelling circumstances" standard. *Monticello, supra,* 5 AEC at 26. Denial of rate relief in and of itself is not unusual, unless it signals a systematic denial of costs. Whether it is compelling depends largely on its impact on LILCO, which at this point remains speculative.

Absent evidence of a systematic denial of costs, it would be inappropriate for this Board to explore financial qualifications based on the denial of construction-related costs. This is an operating license proceeding, and although Intervenors were free to request that this Board examine specific safety-related problems which have allegedly resulted from lack of funds for construction, it is inappropriate for this Board to hear those financial qualifications issues related to construction in the abstract. We discussed these precepts over 2 years ago. *See Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), LBP-82-41, 15 NRC 1295, 1305 (1982) (Construction Permit Extension).

Intervenors maintain, however, that by adding up LILCO's debts and assets, it is clear that LILCO does not have sufficient funds to operate Shoreham. However, this ignores the fact that LILCO may recover its costs of operating Shoreham through the ratemaking process, and that these funds should be used to operate Shoreham safely, in conformance with NRC regulations. To say that the funds would not be used for this purpose, requires the presumption that they will be reapportioned from the safety area to other areas. There is no basis for this Board to make that assumption at this time. In addition, while the New York Public Service Commission does not specifically conduct audits to ensure that revenues are not reallocated, it does monitor plant performance and orders special audits if problems arise. Thus, it indirectly assures "that

[8] Attachment 1 to the Eaker Affidavit, filed in support of LILCO's July 16, 1984 Reply to Intervenors' motion, attests to this fact. In response to a National Association of Regulatory Utility Commissioners' questionnaire, the New York Public Service Commission stated that it "makes allowances for all the necessary and prudently incurred operating costs, including NRC safety requirements."

monies to be spent on nuclear plant operation are not spent elsewhere." Attachment 1, to Eaker Affidavit at 4.

The bulk of the allegations in the Dirmeier Affidavit appear to be directed more toward proving the contention than toward supporting the petition for exception. We do not dispute that the Nine Mile Point default and LILCO's low bond rating are evidence of LILCO's overall weak financial position. Yet, this Board is not permitted to hear those issues until Intervenors have made a *prima facie* showing that the financial qualifications regulations should be waived. What Intervenors have overlooked is that the Commission exempted electric utilities because of their *regulated* status which generally guarantees recovery of reasonable costs and insulates a utility, at least to some extent, from traditional economic forces. It cannot be presumed that the Commission issued these regulations on the assumption that the financial picture of utilities would always be rosy. It did presume that utilities could obtain sufficient funds to operate a plant safely through rate relief. Intervenors have not made a *prima facie* showing that this presumption does not apply in this case.

It is not clear that Intervenors are required to raise a safety issue to support a petition for waiver of the financial qualification regulations. Admittedly, the major emphasis of NRC regulation of nuclear power plants has been on health and safety issues and not financial issues in the abstract. Yet, in its recent Policy Statement, the Commission specifically stated that the lack of demonstrable connection between financial qualifications and safety was not the rationale behind the new rule. Financial Qualifications Statement of Policy, 49 Fed. Reg. 24,111, col. 2 (June 12, 1984).[9] However, challenges to this rule may be limited to cases where the petitioner makes a *prima facie* showing, not that rate relief has been denied but that the local utility has been denied "costs of compliance with NRC requirements associated with *safe* plant operation." Elimination of Review of Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants, 49 Fed. Reg. 13,044 at 13,045, col. 3 (April 5, 1984) (emphasis added).

Because Intervenors have failed to make a *prima facie* showing that application of the financial qualifications regulations to this proceeding would not serve the purpose for which these regulations were adopted,

[9] LILCO cites to the Commission's decision in *Maine Yankee Atomic Power Co.* (Maine Yankee Atomic Power Station), CLI-83-21, 18 NRC 157 (1983) as the basis for its conclusion that Intervenors need to raise a safety issue to support their petition for waiver. This decision was issued prior to the 1984 Policy Statement.

we must deny their petition for waiver or exception. Specifically, Intervenors have not shown that LILCO cannot obtain sufficient funds to operate Shoreham safely through the ratemaking process. We are aware however, that LILCO is experiencing financial difficulties, and it may be appropriate for the Commission to have the Staff determine if these difficulties have led to any safety problems to date, and to continue to monitor more closely than it normally would, LILCO's operational readiness (staffing, resources, etc.) if and after any operating license is issued. *Cf. Maine Yankee, supra* note 9, where the Commission directed the Staff to review the situation to determine if any safety problems arose as a result of financial difficulties.

IV. STANDARDS FOR DETERMINING ADMISSION FOR LATE-FILED CONTENTIONS

Intervenors' motion to file a contention is untimely. Hearings before this Board, on issues for which the record has already been reopened, are scheduled to commence on September 5. Hearings before the Miller Board have already been concluded except for possible hearings on one sub-issue.

However, a contention may be admitted if the balance of the following factors weighs in an intervenor's favor.

> 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.

10 C.F.R. § 2.714(a)(1). We find that only the fact that no other party will litigate this contention weighs in Intervenors' favor. Thus, on balance, these factors weigh heavily against admission of the contention.

A. Good Cause

New information in a previously unavailable document has generally constituted a valid basis for the late filing of contentions and evidence of good cause. However, good cause does not exist when information which forms the factual basis of the contention is publicly available elsewhere. *Duke Power Co.* (Catawba Nuclear Station, Units 1 and 2), CLI-83-19, 17 NRC 1041 (1983). Despite the fact that Intervenors cite frequently to LILCO's Position Paper on Shoreham,[10] which was submitted on May 31, 1984, all information crucial to the contention was publicly available elsewhere well before that date. Other details which may be newer, add little, if anything, to the factual basis of the contention.

Intervenors premise their contention primarily on the conclusion that LILCO cannot raise the funds necessary to cover expected expenditures for 1984, and that the financial uncertainties caused by the prudency investigation and the Nine Mile Point default exacerbate those difficulties. Drawing largely on LILCO's Securities and Exchange Commission Form 10-K, dated March 30, 1984, Intervenors attempt to show that, even after accounting for funds saved through austerity programs and by omitting common stock dividends, LILCO will have a cash shortfall of approximately $80 million. Dirmeier Affidavit at 8. They maintain that LILCO cannot obtain these needed funds through external capital markets because "[a]ll of LILCO's existing lines of credit have been drawn down" (*id.* at 9) and none of its securities are considered investment grade (*id.* at 10). To further support their contention, Intervenors point to the institution of the prudency investigation, where the Staff of the New York Public Service Commission proposes to disallow $1.8 billion in Shoreham-related construction costs, and to the Nine Mile Point default, where the acceleration of approximately $500 million debt is forestalled only by successive 30-day agreements. Intervenors contend that these events place LILCO on the brink of financial collapse.

Most of the information referred to in the Dirmeier Affidavit was derived directly from LILCO's Form 10-K. However, Intervenors maintain that the May 31 Position Paper adds some crucial pieces of information — particularly not only the fact "that LILCO was teetering on the brink of bankruptcy but also that the Company requires the *affirmative action of third parties* (over whom LILCO has no control or influence) to stave off disaster: a billion-dollar bailout and concessions in the prudency proceeding." Intervenors' Memorandum at 29. In addition, "the Position Paper reveals, again for the first time, that additional austerity measures would not suffice to avert bankruptcy." Dirmeier Affidavit at 16.

The Board finds no particularly startling factual averments in these statements which could not have been discovered by reviewing publicly available documents at an early date. At a minimum, this information

[10] The purpose of the Position Paper submitted to Governor Cuomo was to outline a plan for rate phase-in of Shoreham costs and to ensure that LILCO and its ratepayers achieved some stability. Position Paper — Shoreham Nuclear Power Station, Exhibit D to Dirmeier Affidavit at 2-3.

was contained in LILCO's Form 10-K which was available by the beginning of April. However while the Form 10-K does provide specific numbers, LILCO's general financial difficulties were well known before even this document became available.

LILCO's cash shortage and the possibility of bankruptcy cannot be considered new information. The cash shortage problem was discussed in LILCO's Form 8-K, dated December 22, 1983 (Attachment 4 to Eaker Affidavit) and in testimony before the New York Public Service Commission in January and February of 1984 (Attachment 11 to Eaker Affidavit). That testimony indicated that the Company might run out of cash in the Fall of 1984. Intervenors were parties to the proceeding in which this testimony was taken. Additionally, LILCO acknowledged that austerity measures, announced on March 6, would not solve these problems. (Eaker Affidavit, Attachment 10 (N.Y. Times, Mar. 7, 1984, at B2).) The possibility of bankruptcy also cannot be considered new information. It was well publicized in late 1983 both in newspaper headlines[11] and in articles reporting on Shoreham.[12] It is impossible to believe that Intervenors, who are so integrally involved in both this proceeding and the New York Public Service Commission rate proceeding, could have missed this information.

LILCO's difficulties in obtaining external financing have also been well known for some time. As Intervenors themselves note, Moody's began lowering its ratings of LILCO's securities in December of 1983. Dirmeier Affidavit at 9. LILCO's Form 8-Ks, filed in December 1983 and January 1984, also note the Company's external financing difficulties. Attachments 4, 6 and 7 to Eaker Affidavit. In addition, LILCO's witness in the New York Public Service Commission proceeding indicated in January and February 1984 that if LILCO missed paying dividends, it would have difficulty in obtaining external financing. Attachment 11 to Eaker Affidavit. LILCO announced suspension of common stock dividends on March 6. Attachment 10 to Eaker Affidavit.

The two events on which Intervenors rely most heavily, the prudency investigation and the Nine Mile Point default, also cannot be considered recent for the purposes of this motion. The Staff of the New York Public Service Commission filed testimony, in State proceedings in which Intervenors are parties, proposing the disallowance of $1.8 billion in Shoreham-related costs on February 10, 1984. The default on payments for Nine Mile Point construction occurred on February 9. Since it was

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[11] *See* Eaker Affidavit, Attachment 12 (*Reports of Bankruptcy Option Send LILCO's Stock Plunging*, Newsday, Nov. 20, 1983, and *LILCO's Dire Option: Bankruptcy*, Newsday, Dec. 2, 1983).

[12] *See* Attachment 12 to Eaker Affidavit, N.Y. Times articles, Oct. 17, 1983, and Nov. 22, 1983.

extremely well publicized, it is impossible to believe that Intervenors were not aware of the default at an early date. Yet, even if they were not, the information was disclosed in LILCO's February 21, 1984 Form 8-K. Attachment 9 to Eaker Affidavit.

Intervenors could also have made the assertion that LILCO's financial picture was dependent on the actions of LILCO's lenders and the outcome of the prudency investigation at an earlier date. Their assertions as to the importance of these events are based primarily on the fact that LILCO has limited cash and its problems in obtaining outside financing. Yet, as indicated previously, these problems were known in late 1983, prior to the occurrence of the Nine Mile Point default and the prudency investigation. Even if Intervenors were not capable of gauging their effect on LILCO, LILCO's Form 10-K makes it explicit as Intervenors themselves note. The effect of these events "as stated by Price Waterhouse [is that] LILCO 'cannot give any assurance of its ability to meet its capital and operating requirements.'" Intervenors' Memorandum at 8, *quoting* Form 10-K.

Governor Cuomo's rejection of the plan outlined in the Position Paper also adds little, if anything, to the factual premise of the contention. As indicated above, LILCO's financial picture was well known prior to this event. Intervenors do not assert that the Governor ever intended to approve this plan, or any plan, such that LILCO's financial picture would have been substantially brighter prior to the rejection.

Although Intervenors cite quite frequently to the May 31, 1984 Position Paper, this is not sufficient to support the assertion that good cause exists for the late filing. The facts upon which Intervenors rely to support their contention, including the Nine Mile Point default, the prudency investigation, cash flow problems, and external financing difficulties were publicly available no later than mid-February 1984. For these reasons, this Board cannot find that Intervenors have shown good cause for waiting until July 3 to file their contention.

B. Other Means of Protecting the Party's Interest

Intervenors contend that "[t]here is no evidence that LILCO's financial qualifications to operate the Shoreham plant will be reviewed, evaluated or even considered by the NRC, unless the proposed contention is admitted." Intervenors' Memorandum at 30. This Board does not dispute this statement. However, the NRC is not the only entity which can ensure that LILCO has the financial qualifications to operate the plant safely. Only the New York Public Service Commission has the authority to allow rates sufficient to cover the costs of operation. If it fails to allow the sufficient rates, it may then be appropriate for the NRC to review the issue. At this point, however, the Intervenors are free to raise their concerns with the New York Public Service Commission. Thus, we cannot say that there is no other means of protecting Intervenors' interest in LILCO's financial qualifications.

C. Assistance in Developing a Sound Record

We do not dispute the fact that Suffolk County has engaged expert consultants to evaluate LILCO's financial condition. This is clear from the Dirmeier Affidavit. However, the fact that the County has engaged these experts is not wholly dispositive on the issue of whether Intervenors can assist in developing a sound record.

This Board has stated that it does not believe that the standard for reopening the record adds anything to the standards for accepting late-filed contentions, when such contentions are not related to previously litigated issues. This is because a test for significance and triability is implicit in determining whether an untimely contention will be admitted. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), LBP-83-30, 17 NRC 1132, 1143 (1983). In particular, "the extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record is only meaningful when the proposed participation is on a significant, triable issue." *Id.*

At this time, we do not find that Intervenors have presented a significant, triable issue which would assist this Board in developing a sound record. No health and safety concerns have been advanced nor does it appear that any are implicated. Intervenors have not shown that the PSC will not allow LILCO sufficient funds to operate Shoreham safely. In fact, Intervenors' only fear is of an irradiated plant whose owner cannot afford to operate it safely. As stated previously, once the plant is constructed in conformance with all applicable regulations, it is unlikely that LILCO will be unable to recover the cost of safe operation through the rate proceeding. Even if Intervenors' financial forecasts are correct, there would be no reason why the plant could not be operated, even if by some other entity, provided that all safety standards are met.

For these reasons we find that Intervenors' contention does not present the significant triable issue necessary for them to assist in developing a sound record.

D. Extent to Which Petitioner's Interest Will Be Protected by Other Parties

The Board agrees that no other party is likely to protect Intervenors' interest in litigating the financial qualifications issue. However, this factor is far outweighed by the other considerations.

E. Extent to Which Participation Will Broaden Issues or Delay the Proceeding

Intervenors cannot seriously expect this Board to believe that admission of this totally new contention "is not likely to have a material impact on the length of these proceedings." Intervenors' Memorandum at 32. Hearings before this Board are scheduled to commence on September 5, only 2 months after Intervenors filed this contention. The Miller Board is even further along procedurally. The hearings before that Board commenced within a month of the filing of the contention and have already, except possibly for one sub-issue, been completed. In order to hear this contention, we would have to authorize a new round of discovery. New testimony would have to be prepared and filed, in advance of the hearing, so as to address this new issue. Under these conditions it is impossible to see how the expected length of the proceedings could *not* be substantially increased.

Admittedly, this Board has stated that "the extent to which the petitioner's participation will broaden the issues or delay the proceeding is properly balanced against the significance of the issue." *Shoreham*, LBP-83-30, *supra*, 17 NRC at 1143. However, as stated previously, the financial qualifications issue is not nearly as significant as Intervenors would have us believe. *See* p. 440, *supra*.

On balance, even if we were to find a *prima facie* basis for granting the petition for exception, we could not admit the contention because it is inexcusably late. The only factor of the balancing test which weighs in Intervenors' favor is the fact that no other party will litigate the financial qualifications issue. This is not sufficient to overcome the unreasonable delay which the contention would impose on these proceedings; the fact that Intervenors have failed to show good cause for filing so late; the existence of an alternative forum, the State rate proceeding, in which Intervenors may protect their interests through direct participation; and the lack of any safety significance at this time.

IV. CONCLUSION

For the reasons stated, we find that Intervenors have not made a *prima facie* showing that special circumstances exist so that application of the financial qualifications regulations to this proceeding would not serve the purpose for which they were intended. Thus, we deny Intervenors' petition, pursuant to § 2.758(b), for exception to those regulations. In addition, we find that Intervenors' contention is inexcusably late and that the balance of factors for determining admission of a late-filed contention weighs heavily against Intervenors.

The Board further finds no reason to certify this issue to the Commission, pursuant to 10 C.F.R. §§ 2.718(i) and 2.730(f). To do so would be contrary to the normal course charted by § 2.758(d). This issue does not require a prompt decision from the Commission to prevent delay or expense; nor does a prompt decision appear necessary to prevent "detriment to the public interest." As we previously stated, Intervenors' contention has no apparent health and safety significance at this time. In any event, the Commission (and the Appeal Board) will be cognizant of this ruling and may direct certification on their own initiative if they believe it appropriate to do so. Intervenors may also petition the Appeal Board or the Commission to consider this issue on directed certification. However, we decline to seek certification, because we do not find it necessary in these circumstances.

IT IS SO ORDERED.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Lawrence Brenner, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
August 13, 1984

Cite as 20 NRC 443 (1984) LBP-84-30A

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Peter B. Bloch, Chairman
Dr. Kenneth A. McCollom
Dr. Walter H. Jordan

In the Matter of	**Docket Nos. 50-445**
	50-446
	(Application for Operating License)
TEXAS UTILITIES ELECTRIC COMPANY, ***et al.*** **(Comanche Peak Steam Electric Station, Units 1 and 2)**	**August 24, 1984**

In this Memorandum, the Licensing Board concludes that a request for a license for fuel loading and precritical testing may be considered pursuant to 10 C.F.R. § 50.57(c) because the activities for which a license is sought fall within the activities for which a low-power license may be granted.

LOW-POWER LICENSE: FUEL LOAD AND PRECRITICAL TESTING

A licensing board may authorize the issuance of a license for fuel load and precritical testing provided that it makes the findings required by § 50.57(a) with respect to the contested activity sought to be authorized. However, the pendency of a broad quality assurance contention requires that the motion be accompanied by evidence concerning the status of those systems required to assure that criticality will not occur during the proposed activities.

TECHNICAL ISSUES DISCUSSED

Fuel load
Precritical testing
Boron equipment
Neutron monitoring equipment
Fuel-handling equipment
Reactor protection systems
Quality control
K_{eff}.

MEMORANDUM
(Request for Evidence Relevant to Fuel Loading)

On August 7, 1984, Texas Utilities Electric Company, *et al.* (Applicants) filed a Motion for Authorization to Issue a License to Load Fuel and Conduct Certain Precritical Testing. Under this limited license, Applicants would implement safety precautions so that the core never would go critical and appreciable quantities of decay products (and decay heat) would not be generated.

The Staff of the Nuclear Regulatory Commission and Citizens for Sound Energy (CASE) have responded to the Motion. CASE opposes the motion.

The Motion is governed by 10 C.F.R. § 50.57(c), covering a license for low-power testing. Since the activities involved in fuel loading are included within the activities that may be licensed under this section, we conclude that we can authorize fuel loading and precritical testing under this section. However, the section requires us to make the findings listed in § 50.57(a) *with respect to the contested activity sought to be authorized.* The contested activities involve at least the following plant systems: (a) boron addition and monitoring equipment, (b) neutron monitoring equipment sufficient to detect significant increases in K_{eff} above 0.95, (c) fuel-handling equipment, and (d) reactor protection systems. Each of the components of these systems is relevant, including mechanical, electrical and instrumentation systems.

Because of the broad quality control contention pending in this proceeding, we must have evidence concerning the adequacy of quality control for the contested systems. In particular, we require evidence concerning the current status of QA/QC oversight of these systems, including evidence that documentation is adequate to assure that unsatisfactory or nonconforming conditions have been corrected and evidence concerning whether or not there are allegations known to the Applicants or Staff about the intimidation of QA/QC personnel who were working on these systems.

We also require evidence: (1) that appropriate QA/QC procedures have been completed for all phases of the activities for which a license is sought, (2) concerning the maximum K_{eff} to be permitted during precritical testing and the K_{eff} that analysis suggests may be achieved during precritical testing if all control rods were inadvertently removed while the boron concentration was 2000 ppm, and (3) that nonborated water will never be injected into the core, substantially diluting the boron below 2000 ppm.

This decision is issued with the unanimous approval of the Licensing Board in our companion docket 50-445 and 50-446. Hon. Herbert Grossman, who serves on the Licensing Board in the companion case involving intimidation, has reviewed this decision and has no objection to its issuance.

ORDER

For all the foregoing reasons and based on consideration of the entire record in this matter, it is, this 24th day of August 1984,

ORDERED

That Texas Utilities Electric Co., *et al.*, shall supply the evidence requested in this order to facilitate further consideration of its Motion for Authorization to Issue a License to Load Fuel and Conduct Certain Precritical Testing.

FOR THE ATOMIC SAFETY AND
LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland

Cite as 20 NRC 446 (1984) **LBP-84-31**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Lawrence Brenner, Chairman
Dr. Richard F. Cole
Dr. Peter A. Morris

In the Matter of **Docket Nos. 50-352-OL**
50-353-OL
(ASLBP No. 81-465-07-OL)

PHILADELPHIA ELECTRIC COMPANY
(Limerick Generating Station,
Units 1 and 2) **August 29, 1984**

The Licensing Board issues a Second Partial Initial Decision finding in favor of the Applicant on all controverted issues prerequisite to authorizing a low power operating license. Offsite emergency planning issues are still pending for litigation.

EMERGENCY PLANS: IMPLEMENTING PROCEDURES

The whole body of implementing procedures need not be ready for challenge in a hearing. *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076 (1983). However, this does not mean that a Board cannot examine implementing procedures which are available and arguably necessary to determine whether certain provisions in the emergency plan meet NRC planning standards. Examination of such implementing procedures is with the adequacy of the *plans* foremost in mind, since the proper object of litigation is the adequacy of the plan.

NEPA: RECIRCULATION OF FES

Since findings of the licensing tribunal are deemed to amend the FES, amendment and recirculation of the FES are not normally required, unless the differences between the decision and the FES are truly substantial. *Niagara Mohawk Power Corp.* (Nine Mile Point Nuclear Station, Unit 2), ALAB-264, 1 NRC 347, 371-72 (1975); *Allied-General Nuclear Services* (Barnwell Nuclear Fuel Plant Separations Facility), ALAB-296, 2 NRC 671, 680 (1975). *See also Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-78-1, 7 NRC 1, 29 n.43 (1978).

TECHNICAL ISSUES DISCUSSED

Onsite emergency planning
Environmental analysis of severe accidents
Quality control of welding
Environmental qualification of electrical equipment
Effect on plant structures of postulated petroleum and natural gas pipeline accidents
Cooling tower plumes; aircraft carburetor icing.

APPEARANCES

Troy B. Conner, Jr., Esq., Mark J. Wetterhahn, Esq., Robert M. Rader, Esq., and **Nils N. Nichols, Esq.,** of Conner & Wetterhahn, P.C., Washington, D.C., for Philadelphia Electric Company.

Benjamin H. Vogler, Esq., Ann P. Hodgdon, Esq., Nathene A. Wright, Esq., and **Michael N. Wilcove, Esq.,** Office of Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C., for the NRC Staff.

Frank R. Romano, Ambler, Pennsylvania, *pro se,* and for the Air and Water Pollution Patrol.

Robert L. Anthony, Moylan, Pennsylvania, *pro se,* and for Friends of the Earth.

Herbert Smolen, Esq., and **Martha W. Bush, Esq.,** Law Department, for the City of Philadelphia.

Charles W. Elliott, Esq., of Brose and Postwistilo, Easton, Pennsylvania, and **Maureen Mulligan** and **Phyllis Zitzer,** Pottstown, Pennsylvania, for Limerick Ecology Action.

Zori G. Ferkin, Esq., Harrisburg, Pennsylvania, for the Commonwealth of Pennsylvania.

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II.E. FINDINGS OF FACT *(Continued)*

SECOND PARTIAL INITIAL DECISION

I. INTRODUCTION

This is the Second Partial Initial Decision (P.I.D.) issued by this Atomic Safety and Licensing Board in this proceeding. The first "Partial Initial Decision (on Supplementary Cooling Water System Contentions)," was issued on March 8, 1983, and resolved the captioned issues in favor of the Applicant (Philadelphia Electric Company or PECo), subject to certain conditions. LBP-83-11, 17 NRC 413 (1983), *appeal pending.*

This second P.I.D. decides all other issues in controversy in favor of the Applicant which are prerequisite for authorization of the low power operating licenses requested by the Applicant for testing and operation up to 5% of rated power, pursuant to 10 C.F.R. § 50.57(c), as limited by 10 C.F.R. § 50.47(d). These issues are listed in the Table of Contents of the P.I.D. Offsite emergency planning issues, which must be resolved in favor of the Applicant as a prerequisite for authorization of operating licenses for power levels in excess of 5% of rated power, are pending for litigation in this proceeding. When and if the low power operating licenses authorized by this P.I.D. are issued is determined by the NRC Staff, based on its review of the many other NRC requirements not in controversy before us, and the certification of completion, in turn, of each of the two reactor units comprising the Limerick Generating Station.

The Limerick Generating Station, Units 1 and 2, is located in Limerick Township of Montgomery County, Pennsylvania. It is on the east bank of the Schuylkill River, approximately 4 miles downriver from Pottstown. Licenses are sought to operate two boiling water nuclear reactors, each with a rated core power level of 3293 megawatts thermal and a net electrical output of 1055 megawatts electric. Final Safety Analysis Report (FSAR) at 1.1-1.

In addition to the Applicant and the NRC Staff (Staff), the parties participating in one or more issues decided in this P.I.D. are: Intervenors Limerick Ecology Action (LEA), Friends of the Earth in the Delaware Valley and Mr. Robert L. Anthony (as a joint party and referred to as FOE), and the Air and Water Pollution Patrol and Mr. Frank R. Romano (as a joint party and referred to as AWPP). The City of Philadelphia and the Commonwealth of Pennsylvania also participated in the hearing as interested governments pursuant to 10 C.F.R. § 2.715(c). The City also litigated some of its own issues. Each party filed proposed findings of fact on issues of interest to them.

There were approximately 40 days of evidentiary hearings held on the issues decided in this P.I.D., between December 12, 1983, and June 20, 1984, in Philadelphia, Pennsylvania.

The Board's Findings of Fact follow in numbered paragraphs, keyed to the lettered subsections, in § II. The Conclusions of Law and the Order (including procedures for appeal) follow in §§ III and IV, respectively.

II. FINDINGS OF FACT

A. AWPP Contention V-4: Aircraft Carburetor Icing

1. Summary

A-1. This Air and Water Pollution Patrol (AWPP) contention arises under the National Environmental Policy Act (NEPA), and alleges that there will be increased icing in airplane carburetors due to emissions from the two Limerick large, natural draft cooling towers. The contention states:

> Neither the Applicant nor the Staff have adequately considered the potential for and the impact of carburetor icing on aircraft flying into the airspace that may be affected by emissions from the Limerick cooling towers.

A-2. We conclude that this contention lacks merit. The Applicant, supported by the Staff, has demonstrated that there will be no hazards to aircraft due to carburetor icing caused by the Limerick cooling tower plumes. Carburetor icing is a well-recognized hazard to carburetor-equipped aircraft. It is caused by water vapor freezing in the carburetor (in which the temperature can drop markedly due to the expansion of the airflow through the throttling valve). If permitted to accumulate, the ice can cause degrading engine performance to the point of failure.

A-3. The proof before us has clearly demonstrated that beyond the short distance from the cooling towers of about a quarter of a mile, the temperature and humidity differences between the plume and the ambient air are insignificant. The plumes would not present a potential carburetor icing hazard different from the naturally occurring atmosphere, because an airplane could not remain in such a small region of the plume for more than a few seconds — too short a time for carburetor icing to present a hazard. Furthermore, in the alternative, and contrary to the evidence, even if conditions in the entire plume (up to about 10 miles long) were significantly different from the surrounding air, it would be highly unlikely that an airplane would, or even could, remain in the plume long enough for sufficient carburetor ice to accumulate to cause engine failure. The plume behavior would not result in "socked in" conditions in the local airport traffic pattern so as to cause airplanes to remain in the plume for long time periods.

A-4. In any event, the above considerations are unrealistically conservative. They do not take into account the fact that normal pilot procedure is to use the required carburetor heat system to prevent ice accumulation. If carburetor ice begins to accumulate, whether caused by a plume or ambient air, there is ample timely notice to the pilot due to symptoms of the degraded engine performance, and gauges, that ice is accumulating and therefore carburetor heat should be applied to melt the ice. Pilots must face normal variations in temperature and humidity conditions over relatively small changes in airspace location of greater magnitude than variations which would be presented by cooling tower plumes.

A-5. The Applicant's witness panel included two meteorologists, Messrs. Maynard E. Smith and David E. Seymour, with impressive credentials and experience in studying cooling tower plumes (including from aircraft). Mr. Seymour is also an experienced pilot and flight instructor with a commercial license. *See* professional qualifications, ff. Tr. 6234. Likewise, the Staff presented an excellently qualified witness panel consisting of an experienced meteorologist, Mr. Earl H. Markee, and an FAA official, Mr. Bernard A. Geier, who serves as manager of the General Aviation and Commercial Division of the Flight Operations office. Mr. Geier has been a certified pilot for over 40 years, and has been a flight instructor. The Staff's panel also included a Staff nuclear engineer, Mr. Harry E.P. Krug, because of his expertise as an instrument-rated commercial pilot. *See* professional qualifications, ff. Tr. 6883. As might be expected from their qualifications, these witnesses, both in the written direct testimony and under extensive questioning at the hearing, displayed thorough knowledge and understanding and strong, thoughtful support for their conclusions. Indeed, they tried valiantly in response to sometimes confusing, repetitive questions, to explain their analyses and bases so that AWPP's lay cross-examiner, Mr. Romano, would understand the situation.

A-6. In contrast to Applicant's and Staff's witness, AWPP's representative (who also testified on behalf of AWPP), displayed insufficient knowledge and expertise to be relied upon. He is a chemist with science degrees. However, he had no knowledge of the meteorology involved in plume behavior. He has been a licensed pilot of small planes with 10 years of flying experience, much of it in the local Limerick area. However, although he is rightfully concerned, as a pilot of a small airplane, with carburetor icing, his premises of the behavior and effect of plumes were proved incorrect, as was his unlikely postulation that inexperienced, imprudent pilots might not use carburetor heat to prevent, or if necessary, remove an accumulation of carburetor ice. Romano (qualifications and testimony), ff. Tr. 6725.

A-7. The evidentiary hearing sessions on this contention were held on January 11-13 and 17-18, 1984.

2. *Behavior of Cooling Tower Plumes*

A-8. In our unpublished Memorandum and Order of November 8, 1983, we denied Applicant's motion for summary disposition of this contention. In doing so, we held that if Applicant had established, as an indisputable fact, its proposition that temperature and moisture conditions in cooling tower plumes beyond a distance of one-quarter mile from the tower were insignificantly different from those in the ambient air, summary disposition would have been warranted. We would have so ruled because aircraft would not, indeed could not, reasonably remain within the influence of a plume within a quarter of a mile of the cooling tower for more than a few seconds[1] — too short a time period for carburetor icing to affect the aircraft. November 8, 1983 ("Summary Disposition") Order at 3-4.

A-9. At the summary disposition stage, we found that there could be a question about the applicability to Limerick of the 1981 Thomson Pennsylvania State University study relied on for Applicant's "one-quarter mile from tower significance proposition," because the design of the cooling towers of the Keystone Plant used in the study was different. *Id.* at 4. Based on the facts established at the evidentiary hearing, as set forth below, we find that the Applicant, without any reasonable contradiction, has established by the overwhelming preponderance of the evidence that the Limerick cooling tower plumes will not have temperature and moisture conditions significantly different from the ambient air beyond a quarter mile from the tower.

A-10. To dissipate the waste heat from the operation of the facility, the Limerick Generating Station will employ two large, natural draft hyperbolic cooling towers 507.5 feet in height. Markee, ff. Tr. 6883, at 3-5.

A-11. The operation of towers of the type used at Limerick creates visible plumes of water vapor under certain atmospheric conditions. The

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[1] For example, assuming both a slow airspeed of 90 mph, and an airplane flown through the long axis of the plume within a quarter mile of the tower, a plane would traverse the quarter mile in 10 seconds. Any other flight path would expose the airplane to potential icing conditions for an even shorter time.

plume emitted by the Limerick towers will always have a higher temperature and greater water content than the ambient air. Excess water vapor will condense to form a visible plume approximately 50 to 80% of the time. The plume will always be less dense than the ambient air and will rise due to buoyancy. *Id.* at 3-13; Tr. 6296, 6298-99, 6320, 6324 (Smith). The exact temperature and humidity content of the plume as it exits the tower will depend on the temperature of the ambient inlet air drawn into the tower and the amount of heat being dissipated from the plant (at different plant operating levels). Tr. 6317, 6322 (Smith).

A-12. As the plume rises it will be cooled by expansion, evaporation, radiation and mixing with the ambient air. Markee, ff. Tr. 6883, at 3-13 to 3-14; Tr. 6290, 6293 (Smith). The rate of heat dilution and consequent plume behavior are affected by the natural turbulence in the atmosphere, the vigor with which the plume exits the tower (1100 to 1600 ft/min at full power operation), and the humidity and temperature of the ambient air relative to the humidity and temperature of the plume. Tr. 6292, 6296, 6407 (Smith); Tr. 6630 (Boyer). Very rapid mixing occurs in the immediate vicinity of the tower. Tr. 6291-93 (Smith).

A-13. A temperature differential of as little as tenths of a degree (Fahrenheit) over the ambient air will result in a buoyant plume. Tr. 6681 (Smith). As they exit from the cooling towers, the plumes will be very close to or at saturation. Tr. 6639 (Smith). Strong winds expedite the mixing process and reduce the plume's buoyancy as its warmer, wetter air is dispersed. Tr. 6299 (Smith). On the other hand, if the atmosphere is relatively still, plumes will rise almost vertically to greater heights and will continue to rise, usually until they reach a layer in which temperature increases with height, i.e., an inversion layer. Tr. 6299-6300, 6407 (Smith). Normally, as a plume rises under nearly calm conditions it generates its own turbulence and mixing and either dissipates while rising vertically or reaches a layer in which there is transport wind and is carried away. Tr. 6302-03 (Smith). A plume rising into air that is already saturated and therefore has a cloud deck will blend into and become part of the ambient cloud deck. Tr. 6408-10 (Smith).

A-14. As testified to by both the Applicant and Staff, it is extremely rare for cooling tower plumes to assume a lateral orientation before reaching an altitude of 1000 feet. Tr. 6894, 6908-09 (Markee); Tr. 6298 (Smith). In their studies of natural draft cooling tower plumes, Applicant's witnesses did not find a single plume whose rise leveled off below 1000 feet. They found only one bent-over plume between 1000 and 1200 feet. Tr. 6298, 6334, 6619 (Smith). Additionally, the Staff testified that there is only an extremely small probability that a plume waft might reach the ground in the vicinity of Limerick. Such an event could only occur as a result of very turbulent, hurricane-type conditions, which would be conducive to plume dispersion in any event. Tr. 6894-95 (Markee).

3. Studies of Cooling Tower Plumes

A-15. Applicant's witnesses relied upon two cooling tower plume studies as part of the bases for their testimony that plumes will not affect carburetor icing in the Limerick area. Smith and Seymour, ff. Tr. 6234, at 5-7; Tr. 6423 (Smith). One of these studies, the Thomson (Pennsylvania State University) study of the Keystone cooling towers in Western Pennsylvania (App. Ex. 13), was conducted expressly to determine conditions inside and outside visible and invisible plumes. Tr. 6259, 6279, 6405, 6418 (Smith). The visible plume was tested by making airplane flights at right-angle cross-sections at various altitudes from top to bottom and at various distances along the length of the plume. Tr. 6259-60, 6419, 6458 (Smith). When the visible plume terminated, those procedures were employed downwind at the same altitudes and at increasing distances out to 10 miles to test the invisible plume. Tr. 6419, 6458, 6460-61 (Smith). This technique enabled the researchers to intersect the so-called invisible portion of the plume with great regularity. Tr. 6262, 6279, 6419-20, 6459 (Smith).

A-16. The Thomson study results indicate that in-plume temperature and humidity conditions vary sharply within one-quarter mile of the tower, with both quantities significantly exceeding ambient levels for very short periods. Smith and Seymour, ff. Tr. 6234, at 5-6. Beyond a quarter mile, however, in-plume temperatures were found to be almost indistinguishable from those of the external air, and the humidity difference dropped to 0.25 gm/kg or less. This is a very small excess as the natural atmosphere, when saturated, contains about 3.5 gm/kg of water vapor at 30°F. This figure increases to 22 gm/kg at 80°F. Smith and Seymour, ff. Tr. 6234, at 5-6 and Figs. 1 and 2; Tr. 7094, 7106-07 (Markee).

A-17. Contrary to AWPP's unsupported claims, the results of the Thomson Keystone study are valid for Limerick. The key climatic conditions applicable to carburetor icing are nearly identical at Keystone and Limerick. Smith and Seymour, ff. Tr. 6234, at 6; Tr. 6423-24 (Smith); Tr. 7033-34 (Markee). The plume and weather conditions at Keystone are not affected by the modest ridges located 40 miles away. Tr. 6444-45 (Smith).

A-18. As noted in our order denying summary disposition, the Keystone towers are smaller than the Limerick towers — 325 feet and 507 feet, respectively. However, the expert witnesses for the Applicant and Staff testified that based on American Electric Power data, there is little difference in comparative behavior of plumes from cooling towers from plants that are about 500 megawatts and larger. Tr. 6424-25 (Smith); Tr. 7033 (Markee). This was not contradicted by either other testimony or under cross-examination.

A-19. We agree with the Applicant's conclusion, supported by Staff's meteorologist (Tr. 7033, 7086-87, 7106-07 (Markee)), that as a result of the plume and ambient air mixing processes described above, the distance would not exceed one-quarter mile from the tower within which temperature and humidity in the plume could reasonably vary enough from the ambient air to cause or exacerbate carburetor icing. This is well supported by their expert knowledge of plume phenomena, their review of the literature, and the Thomson Keystone study. *See, e.g.,* Smith and Seymour, ff. Tr. 6234, at 5-6 and Figs. 1 and 2; Tr. 6267, 6286, 6312-13 (Smith); Tr. 6286, 6350-51 (Seymour).

4. AWPP's Disagreements Regarding Plume Behavior

A-20. AWPP's disagreements with the information and conclusions regarding plume behavior testified to by the Applicant's and Staff's experts are insubstantial and without foundation. The arguments by AWPP's representative show an unfortunate apparent inability to understand the testimony. Indeed, the arguments illustrate why the testimony of AWPP's representative is entitled to no weight. For example, AWPP seems to believe that the testimony that plumes will not affect carburetor icing beyond a quarter mile from the tower means that Applicant and Staff believe that plumes longer than a quarter mile will not exist. This is not correct. The testimony is that longer plumes will exist, at times as much as 5 or 10 miles long. Tr. 6264-65 (Smith). On rare occasions, the Applicant postulated that, based on American Electric Power studies performed by Mr. Smith, and a computer modeling run for Limerick, the Limerick plumes may even exceed 10 miles. Smith and Seymour, ff. Tr. 6234, at 7-8. This is not inconsistent with the well-supported, uncontradicted, and often repeated testimony at the hearing, regarding the lack of significant temperature and humidity deltas of the plume over the ambient air at distances greater than one-quarter mile from the tower.

A-21. Similarly, AWPP's argument (proposed finding 6) that the velocity of the plume as it exits the tower, of 1100 to 1600 feet per minute, contradicts the testimony of lack of significance beyond a quarter of a mile. This argument is a *non sequitur.* In the first instance, even if that velocity continued, we fail to see how a high-velocity plume could contradict the testimony and data of lack of significance of the conditions within the plume beyond a quarter mile. To the contrary, if such velocity continued it would appear to promote even more rapid mixing of the plume with the ambient air. In any event, the testimony was only that these velocities occurred at the point of exit of the plume from the tower, not that they persisted. *See* our Finding A-12.

A-22. AWPP postulated that saturated, stagnant ambient conditions could cause the cooling tower plumes to remain near the ground and concentrate in an inversion condition, causing a carburetor icing threat. This was unsupported by AWPP, and was authoritatively discredited by the expert testimony of the Applicant and the Staff. As noted above (Finding A-13), when the ambient air is saturated, the plume will rise into the atmosphere, continue to mix with the ambient air, merge with the cloud deck, and then be transported away over the course of about an hour. Tr. 6408-10 (Smith). Further, during stagnant ambient conditions, plumes would rise to greater heights than normal and would not cause a significant humidity increase in the airspace close to the tower or the ground. Smith and Seymour, ff. Tr. 6234, at 14; Tr. 6407, 6712-13 (Smith). There is no such thing as completely stagnant air — air always moves, although at slower rates in stagnant conditions. Tr. 7050-51 (Markee).

A-23. The plume phenomena described above show that even when ambient dispersion conditions are poor (i.e., stagnant), plumes will rise to heights of several thousand feet, where the stronger winds will disperse them. Markee, ff. Tr. 6883, at 2. The computer model run for Limerick by the Applicant is consistent with this expert view. It indicates that the Limerick plumes will always reach a height of at least 1000 feet above ground before leveling off, if they have not dissipated before reaching that altitude. Smith and Seymour, ff. Tr. 6234, at 7-8. *See also* our Finding A-14.

5. *Aircraft Carburetor Icing*

A-24. AWPP's assertion that the Limerick cooling tower plumes will lead to increased aircraft carburetor icing ignores the fact that the conditions causing carburetor ice formation are well understood and that steps have been taken to assure that it does not present a significant problem to pilots who are reasonably attentive. Smith and Seymour, ff. Tr. 6234, at 8; Geier, ff. Tr. 6883, at 2-4; Krug, ff. Tr. 6883, at 2-3. Carburetor icing occurs as follows: The vaporization of fuel, combined with the rapid expansion of air as it passes through the carburetor intake valve, causes that mixture to cool; the water vapor content of the intake air may then condense, and if the temperature in the carburetor reaches 32°F or below, the moisture can be deposited in the fuel intake system as frost or ice which may reduce or block the passage of the fuel/air mixture to the engine and cause engine failure. Due to the venturi effect of a partially closed throttle valve, carburetor ice is more likely to form when the throttle is not fully open. The temperature of air passing downstream of the throttle valve may drop as much as 60°F. Smith and Seymour, ff. Tr. 6234, at 8; Geier, ff. Tr. 6883, at 2.

A-25. On very dry days and when the temperature is well below freezing, the moisture content of the air is not sufficient to cause carburetor icing. But if the temperature is between 20°F and 90°F, and moderate humidity or visible moisture is present, there is a potential for carburetor ice to form. Smith and Seymour, ff. Tr. 6234, at 8-9; Tr. 6517-18 (Seymour).

a. Time for Formation

A-26. Experiments have been conducted on the ground using an automobile engine and an airplane carburetor to accumulate the greatest amount of carburetor ice in the least amount of time so as to establish the power losses associated with timed exposure to optimum icing conditions. Such studies are done in a laboratory because it is difficult to find optimum conditions for carburetor ice accumulation occurring naturally. Tr. 6507-08 (Seymour).

A-27. At such conditions (68°F and 100% humidity), the study found it would take 8 minutes of flying time for enough carburetor ice to accumulate to cause a 25-rpm reduction in engine speed. This result assumes that the proper preventive and remedial measure of using the carburetor heat control, discussed below, is not taken. Such a drop is not even significant enough probably to be noticed by the pilot. Smith and Seymour, ff. Tr. 6234, at 9; Tr. 6374-77, 6527-28 (Seymour). The FAA witness appearing on behalf of the Staff stated in his direct written testimony that although carburetor ice can form instantaneously under the proper conditions, it does not accumulate at such a rate that the pilot who pays attention to the signs cannot prevent engine stoppage due to blocking by ice of the carburetor throttle. Geier, ff. Tr. 6883, at 2.

A-28. On its face, the FAA witness' prepared testimony is not inconsistent with the Applicant's testimony based on the icing test studies. Instantaneous ice formation is not an accumulation of carburetor ice which would create a flying hazard. That this is what the FAA witness meant was clarified at the hearing. He and the other Staff pilot witness did not wish to testify to a particular time frame such as 5, 8 or 10 minutes, due to variation in aircraft and conditions. Tr. 7002-03 (Krug, Geier). However, he explained he agreed with and had no evidence to believe that the conclusion of the study relied on by the Applicant was wrong — i.e., that it would take some time (8 minutes according to the study) of flying through adverse conditions without carburetor heat to accumulate enough carburetor ice to present a significant hazard to an aircraft. Tr. 7001-03 (Geier).

A-29. Based on the above, even if an airplane would fly in the plume within a quarter mile of the tower, it would pass through that area in a matter of seconds — much too soon for hazardous carburetor ice to accumulate. The use of the quarter-mile distance as the maximum area of potential adverse effect was conservatively based on the premise that differential conditions between the plume and ambient air conditions of not more than 1°C or a half a gram of water vapor per kilogram of air would not have an effect on carburetor icing. Tr. 6249 (Smith). As discussed above (Finding A-16), the conditions beyond the quarter-mile distance would not exceed that. Actually, the one-quarter-mile distance proposition is conservative, because a differential between the plume and ambient air conditions of 2 or 3°C and 10 or 20% humidity would not significantly affect aircraft carburetor icing. Tr. 6267 (Smith).

A-30. Moreover, even if we believed, contrary to the evidence, that the cooling tower plumes could cause carburetor icing for distances beyond one-quarter mile from the tower, and that pilots would not apply carburetor heat to prevent or remedy icing, there is another factor which demonstrates that the contention has no merit. The record fully supports, and we agree with, Applicant's proposed findings (45-47), showing that it would be highly unlikely — indeed a nearly impossible, purposeful maneuver — for a pilot to keep a small general aviation airplane of concern in this contention within even the largest cooling tower plumes for their full extent long enough for enough carburetor ice to form to present a hazard to the airplane. *See, e.g.*, Smith and Seymour, ff. Tr. 6234, at 7-11.

b. Prevention and Elimination of Carburetor Icing

A-31. It is not necessary to make further findings in order to decide that the contention lacks merit. However, we do so to show that the conservative assumption used to this point that the pilot would not prevent or, if encountered, remedy carburetor icing, is unrealistic.

A-32. All airplanes with carburetors are required to have carburetor heat systems to prevent and eliminate icing. Geier, ff. Tr. 6883, at 3. All parties agree that aircraft manufactured since World War II have such systems, and therefore 99% of the airplanes flown in the Limerick area are so equipped. Tr. 6651 (Seymour); Tr. 6834 (Romano).

A-33. AWPP agrees that if carburetor heat is used, ice will not form. Tr. 6852 (Romano). Unless the ice were allowed to accumulate over a long enough time, during which the pilot would have to ignore seriously degrading engine performance, by design of the airplane carburetor ice can be removed in seconds by the use of carburetor heat. Tr. 6364-67, 6376-78, 6383-84, 6668-71 (Seymour); Tr. 7004-05 (Geier). Carburetor ice would not cause instantaneous engine failure without significant noticeable symptoms alerting the pilot to the problem. Tr. 6376-81, 6628-29 (Seymour). A trained pilot would not be likely to confuse the indications of other engine problems with the indications of the accumulation of carburetor ice. Geier, ff. Tr. 6883, at 4-5.

A-34. Beyond the fact that a pilot should be able to remedy a carburetor ice problem after detection, there are proper flight procedures for different maneuvers to prevent a carburetor ice problem. These procedures would prevent problems in the local Limerick area even though there are airplanes taking off and landing at local airports near Limerick.[2]

A-35. Carburetor heat is not used in normal flight as it reduces the output of the engine, but pilots are trained to apply carburetor heat at the first indication of an icing problem. Smith and Seymour, ff. Tr. 6234, at 12. Also, carburetor heat is not normally used during takeoff because full power is desired and the potential for carburetor ice is less when the throttle is fully open. Tr. 6673-75 (Seymour); Tr. 7042 (Krug). However, before taking off a pilot should test his carburetor heat control. This will assure that it is working. It will also indicate whether any ice is present based on the reaction of the engine to the application of the heat. If symptoms of ice occur during that preflight check, then the carburetor heat should be reapplied just before takeoff to assure the carburetor is clear at that time. Smith and Seymour, ff. Tr. 6234, at 12; Tr. 6673-74 (Seymour).

A-36. In making an approach for landing an aircraft which has a carburetor, the pilot normally applies carburetor heat on the downwind leg even if there is no indication of carburetor ice. An increase in engine rpm after the carburetor heat is applied is an indication that carburetor ice was present and that the heat has eliminated it. Such an increase is

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[2] Based on our findings on plume behavior, local airport traffic will not be affected by the plumes which, if they do not dissipate first, will rise to over a thousand feet above the ground. The typical airport traffic altitude is 800 feet for light aircraft and 1000 feet for heavy aircraft. Tr. 6688-89 (Seymour). The pattern altitude at the closest airport, Pottstown-Limerick, is 889 feet above the ground (1200 msl), well below the lowest heights at which plumes will level off. Tr. 7101-02 (Geier).

an indication that the pilot should continue to use the carburetor heat. "As required" in a flight manual instruction regarding the use of carburetor heat means that normal procedure is to leave the carburetor heat on throughout the approach. Tr. 6890, 7007-08 (Geier).

A-37. In the case of a "go-around," a situation in which a pilot must reapproach the runway after beginning his pre-landing descent, carburetor heat would have been applied during the pre-landing descent. Once a pilot realized that a go-around had become necessary, carburetor heat would be eliminated and full power applied, thus ameliorating any icing potential. Carburetor heat would again be applied upon reentering the landing approach. Tr. 6676 (Seymour); Tr. 6835-36 (Romano); Tr. 6890 (Geier).

A-38. It is not our conclusion that aircraft cannot be placed in hazardous circumstances, perhaps even to the point of a tragic accident, by carburetor icing. But it is our finding that this would occur only due to pilot failure to use well-established procedures and available equipment. The procedures are well established and the carburetor heat systems are required precisely because aircraft carburetor icing is a well-recognized potential hazard.

A-39. More to the point, any variation between the cooling tower plumes and the ambient air is insignificant when compared to the much larger normal temperature and moisture variations over relatively small changes in location that pilots face in routine flights through ambient air. Indeed, changes in altitude of a few hundred feet may result in differences of 5 to 10°F and 50 to 60% in humidity. Tr. 6997-98 (Krug); Tr. 6356 (Smith); Tr. 6367 (Seymour); Tr. 6644-47 (Smith, Seymour).

A-40. Based on all of the above, we find that AWPP Contention V-4 lacks merit.

B. FOE Contentions V-3a and V-3b: Natural Gas and Petroleum Pipeline Accidents

1. Background

B-1. On September 19, 1981, Mr. Robert L. Anthony filed a petition to intervene on behalf of himself and Friends of the Earth in the Delaware Valley (FOE), including some thirteen proposed contentions. In its Memorandum and Order of October 14, 1981 (unpublished), this Board scheduled a special prehearing conference for approximately the first week in January 1982 to consider, *inter alia*, the contentions, the objections to the contentions, and the responses by petitioners to the objections — from all participants in the proceeding at that time. We also required that all contentions be refiled, since coordination among petitioners had not taken place and some of the preliminary contentions were poorly organized, redundant and unclear.

B-2. On November 24, 1981, in a Supplemental Petition of Coordinated Intervenors, FOE, among eleven other petitioners, filed seven proposed contentions, which superseded those filed previously. FOE/Mr. Anthony was found to have standing to intervene in this proceeding. The Board denied six of FOE's seven contentions in its Special Prehearing Conference Order (SPCO) of June 1, 1982. LBP-82-43A, 15 NRC 1423 (1982). Our ruling on one of FOE's contentions (VIII-11, having to do with emergency planning) was deferred until after the Limerick emergency plans became available. While we denied FOE's Contention V-3, related to the danger of fire and explosions in connection with gas and oil pipelines and industry near the plant, we allowed FOE 30 days to file contentions which would allege specific deficiencies which FOE believed existed in the FSAR analysis of these matters. *Id.* at 1513-14. FOE responded to our SPCO on July 7, 1982, listing ten contentions that it characterized as severe deficiencies in § 2.2 of the FSAR. Generally, these related to explosions, fires and missiles arising from pipeline and industrial activities.

B-3. In our Order (Concerning Proposed FOE Contentions on Hazards from Industrial Activities) of November 22, 1982 (unpublished), we denied all but two of the newly proposed contentions, i.e., Contentions 3 and 5. To focus these contentions on the areas of concern, the Board rewrote and renumbered them, as follows:

> V-3a. In developing its analysis of the worst case rupture of the ARCO pipeline, the Applicant provided no basis for excluding consideration of siphoning. Thus, the consequences from the worst case pipeline accident are understated.
>
> V-3b. In discussing deflagration of gas and petroleum due to pipeline rupture, no specific consideration has been given to the effect of radiant heat upon the diesel generators and associated diesel fuel storage facilities.

B-4. We note that with respect to Contention V-3a, consequences from the worst-case pipeline accident were understood to encompass missiles of pipe fragment or rock damaging plant facilities as well as damage from overpressure. With respect to Contention V-3b we note that concerns about the impact of a pipeline fire on the diesel generators and the diesel fuel storage facilities were not discussed explicitly in the FSAR.[3] Although not explicitly part of FOE's contentions as admitted,

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[3] FOE/Anthony filed a response to and a motion to reconsider our November 22, 1982 order regarding FOE contentions on December 19, 1982. Upon reconsideration, we denied the motion on March 10, 1983 (LBP-83-14, 17 NRC 473).

the Board found that consideration of the detonation of natural gas from the Columbia Gas pipelines, which all parties had addressed in their prefiled testimony, should properly be considered for completeness, given the issues in controversy before us. "Memorandum and Order Ruling on Motions to Strike Testimony" (December 1, 1983) (unpublished).

B-5. As a preliminary matter, we note that the proposed testimony of Mr. Anthony on Contentions V-3a and V-3b was not accepted, because he does not possess the expertise necessary to testify as an expert witness. We did allow the testimony of Mr. Bevier Hasbrouck, on the basis that he was marginally qualified as a physicist to discuss pipeline explosions, even though he had no direct experience in this area. Evidentiary hearings on these matters were held on December 12-16, 1983; January 9-10, 23-25, March 8-9, 20-23, 1984.

B-6. The Board wished to ascertain from the Applicant and the Staff at the outset whether they depended, for any part of their cases on these contentions, on the probability of a breach in the pipelines occurring, as opposed to the nature of such a breach and its potential consequences. Both Applicant and Staff conceded that a pipe break could occur. Tr. 5076 (Wetterhahn); Tr. 5076-77 (Vogler). Consequently, we do not consider the probabilities of rupture of either the ARCO or the Columbia pipelines. We do consider the consequences of worst-case accidents potentially resulting from the rupture of these pipelines in the vicinity of the Limerick Generating Station. To do this we determine, in turn, the nature of the materials transported in the pipelines, how much of these materials could react to produce heat and blast overpressures and the ability of safety-related structures, systems and components to withstand such impacts, including interactions from the nonsafety-related structures, systems and components that could be damaged from the results of potential heat or blast impacts.

2. *Summary*

B-7. In consideration of FOE's Contentions V-3a and V-3b, the Board has carefully evaluated the potential effects on the Limerick Station of postulated ruptures of the ARCO and Columbia pipelines. We have not considered what might have been argued as to the low probability of such ruptures. We have considered what we believe to be very conservative postulates of accident scenarios that would lead to radiant heat and overpressure impacts on the Station. Such conservatisms include the distribution of material released from the pipelines, the meteorological conditions prevailing at the time of rupture, the transportation and dispersion of flammable mixtures toward the Station and the assumption that such unconfined mixtures could be detonated. Even assuming burning or detonation of such mixtures, conservative calculations of the radiant heat loads and overpressures on the safety-related structures at Limerick, and the effects of failure of nonsafety-related structures on the safety-related structures, demonstrate the adequacy of these structures to withstand the effects of postulated ruptures of the ARCO and Columbia pipelines. Accordingly, we find FOE's Contentions V-3a and V-3b to have no merit.

B-8. We find the Applicant's and Staff's witnesses to be qualified and competent in their respective disciplines and their testimony to be credible and persuasive. On the other hand, we find the qualifications of FOE's sole witness to be limited, in education, training or experience applicable to the issues raised in these contentions. Based on limited qualifications, and the content of his testimony, we assign no weight to his testimony.

3. The ARCO Pipeline

a. Description of Pipeline

B-9. The ARCO Pipe Line Company operates and maintains a pipeline that traverses Chester and Montgomery Counties in Pennsylvania. This is known as the 8" Northeast Boot (Pa.) to Fullerton (Pa.) Pipeline. It consists of an 8-inch-diameter, 0.250-inch wall thickness x 42 grade steel pipe coated with a coal tar enamel and additionally protected against corrosion by an impressed electrical current cathodic protection system. Christman, ff. Tr. 5093, at 1-3. The pipeline has a capacity of 31,700 barrels per day[4] and operates at a maximum pumping pressure of 1100 pounds per square inch gauge (psig). Normal operating pressures for gasoline are 850 to 875 psig and for diesel and furnace oil, 950 to 1000 psig. The pipeline was buried at least 3 feet below grade at the time it was constructed in 1955. Christman, ff. Tr. 5093, at 3.

b. Contents of Pipeline

B-10. The pipeline carries automobile gasoline, kerosene, diesel oil and home heating oil. ARCO Pipe Line Company has stipulated in an amendment to its right-of-way agreement with PECo that it will not carry propane through the line. The pipeline has never carried butane or

[4] One barrel of petroleum products is equivalent to 42 gallons. Thus, 31,700 barrels per day is equivalent to 55,475 gallons per hour (gph).

liquefied natural gas (LNG) and could not carry either product without physical modification of the pipeline. Tr. 5109 (Christman). Although the pipeline could carry aviation fuel, which is simply a higher octane gasoline than used for automobiles, the line has never been used for this purpose, to the knowledge of Mr. LeRoy A. Christman, who is the Montello District Manager for ARCO for approximately 1000 miles of pipeline in Pennsylvania and New York, including the 8" Northeast Boot to Fullerton Pipeline. The present tariffs on file with the Pennsylvania Public Utilities Commission (PUC) cover transportation of the following: gasoline, kerosene, jet engine fuel, tractor fuel, diesel fuel, and light and medium fuel oil. Christman, ff. Tr. 5093, at 1, 4. Kerosene and jet engine fuel would be less volatile than automobile gasoline. Tr. 5231 (Christman). Automobile gasoline was considered in the Applicant's analysis because it is the most volatile substance carried and has the highest energy content. Aviation gasoline has a lower volatility and lower heat content than automobile gasoline. Walsh, ff. Tr. 5411, at 4. No new product has been added since 1978. Tr. 5122 (Christman). If propane were added to the tariff, it would certainly be known by Mr. Christman and others well in advance. Tr. 5122 (Christman). *See also* Agreement attached to the Testimony of Vincent Boyer, ff. Tr. 5412.

c. *Location of Pipeline*

B-11. The Northeast Boot to Fullerton line is 48.87 miles long. Christman, ff. Tr. 5093, at 3. Within a radius of 5 miles of the Limerick site the pipeline runs generally in a south-to-north direction. FSAR Fig. 2.2-1. *See also* Fig. 1, taken from the SER (Staff Ex. 6) and reproduced at the end of this section of the decision solely to provide a general depiction of the orientation of the ARCO and Columbia Gas pipelines. Its location in the vicinity of the site is depicted in Applicant's Ex. 18, a site plan drawn with a scale of 1 inch equal to 200 feet. This plan includes 2-foot topographical contour lines. It shows the pipeline proceeding northward from the easternmost corner of the Limerick Information Center parking lot approximately 400 feet, then slightly west of north for approximately 850 feet, then north for approximately 500 feet, and then east of north for approximately 1200 feet. Almost directly east of the valve and meter house (located between the two cooling towers), the pipeline crosses Possum Hollow Run. Approximately 550 feet south of this crossing, the surface elevation reaches the nearest high point in this direction of approximately 244 feet msl (mean sea level). Approximately 1300 feet to the east of north of this crossing, the surface elevation reaches the nearest high point in this direction of approximately

272 feet msl. PECo's witness Walter C. Payne identified these high points as being approximately 270 feet elevation, approximately 1400 feet north and approximately 245 feet elevation, approximately 600 feet south of the Possum Hollow Run crossing. Tr. 5378-79 (Payne). The elevation of Possum Hollow Run at the point of the pipeline crossing is approximately 168 feet msl. The nearest approach of the pipeline to the Unit 2 reactor building is approximately 1603 feet. The Unit 2 Diesel Generator Building is 1665 feet away. Payne, ff. Tr. 5357, at 5. It should be noted, however, that the location of the pipeline itself, or the location of breaks in the pipeline, are not necessarily considered to be the actual locations of the fires or explosions that are postulated for the purposes of this decision. These latter locations are determined from the postulated break locations and other factors, such as topography, wind direction and speed, as discussed below.

B-12. FOE contended that the Applicant did not know where the ARCO pipeline was located (in the vicinity of the Limerick site) and that the Applicant could be wrong by 50 to 100 feet. Tr. 5135-36 (Anthony). Witness Payne testified that using a more refined technique than photogrammetry, PECo knew the location of the pipeline within less than 1 foot over 90% of its length and within a foot or two over the remaining 10%. Tr. 5380-81 (Payne). The more refined technique is described in detail by Payne, ff. Tr. 5357, at 3-4. From its recent investigation, the Applicant determined that the location of the pipeline as indicated in FSAR Fig. 2.2-4 deviates slightly from its true location. At its maximum deviation, it is actually 50 feet farther from the Station facilities than shown in the FSAR figure at the point where the pipeline exits from the northern boundary of the Station property. Payne. ff. Tr. 5357, at 10.

B-13. Staff witness Charles M. Ferrell testified that he checked this location of the ARCO pipeline in three ways, (a) by use of a high-altitude (24,000 feet) infrared photograph of the Limerick site (Attachment 1 to the prefiled testimony of Ferrell *et al.; see* Tr. 6133-35), (b) a high-altitude (12,000 feet) black and white photograph of the Limerick site (Attachment 2 to the same prefiled testimony), (c) and by flying over the site at low elevations. Ferrell *et al.*, ff. Tr. 6136, at 4, 5. He concluded that the ARCO pipeline is accurately indicated on Fig. 2.7 of the SER. This Figure appears to be a reduced replica of Applicant's Ex. 18.

B-14. FOE failed to controvert the evidence of the Applicant and Staff concerning the location of the ARCO pipeline. The Board finds that the location of the ARCO pipeline is accurately indicated on Applicant's Ex. 18.

B-15. In any event, the exact location of the pipeline is important only for the purpose of determining the location of potential flammable mixtures of gasoline and air that could result from a pipe break. Measuring distances to within 1/16 inch on Applicant's Ex. 18 permits distances to be determined within approximately 10 feet, which, as will become evident in our discussion of consequences, is clearly more than accurate enough for the analysis required for reaching our conclusions with respect to this contention. We rely, however, on the Applicant's survey, as presented in Mr. Payne's testimony. Payne, ff. Tr. 5357, at 3-5.

d. Nature of the Release

B-16. A number of "scenarios" were postulated for the release and distribution of gasoline from the ARCO pipeline, its evaporation and formation of an explosive volume within the atmosphere, its burning or detonation and the resulting heat and overpressure impacts on the Limerick structures. Initially, Applicant assumed a break to take place where the pipeline crosses Possum Hollow Run at a time when automobile gasoline was being transported. Gasoline was postulated because it is the most volatile substance transported by the pipeline and has the highest energy content. Because the pipeline is monitored by pressure sensors to detect sudden rises or decreases in pressure that would automatically shut off the pumps, Applicant assumed that the total amount of gasoline released would be limited to that contained in the pipe between the high points on either side of the break. This was calculated to be 4962 gallons. Walsh, ff. Tr. 5411, Attachment 1, at 1-2. By assuming the break at the low point — Possum Hollow Run — the maximum amount of gasoline would be released. In the case of a small leak, Applicant testified that it would be detected by the operators in a relatively short time by inventory procedures and the pipeline would be shut down. Walsh, ff. Tr. 5411, at 3-4. Applicant also initially omitted consideration of any siphoning effects that could increase the amount of gasoline escaping, because to achieve such siphoning, an additional opening to the atmosphere would have to occur at a location beyond an adjacent high point. *Id.* at 5-6. Intervenor challenged the lack of consideration of siphoning in its Contention V-3a. While the Board finds that siphoning could not be conclusively excluded, based on the record before us, we need not try to speculate on the additional amount of gasoline discharged from the break caused by siphoning, which might result from an additional opening in the pipe at some other undefined location. Rather, the Board notes that the record also does not support the reliability of automatic or manual shutdown of the pumps in the event of a leak from or break of the pipe. Thus, as a worst case, we consider the case where the pumps operate continuously after the break.

e. Formation of a Flammable Mixture

B-17. The "source term" for the quantity of gasoline that could lead to an explosive mixture with air is not the total amount that escapes the pipe, but instead the surface area of the gasoline as it spreads over the terrain after leaving the pipe. The surface area is the important consideration because it controls the rate at which the gasoline evaporates and permits the vapor and air to form an explosive mixture. Walsh, ff. Tr. 5411, at 6. We proceed to consider the surface area that might be covered with gasoline as a result of a pipe break not only at the low point where the pipeline crosses Possum Hollow Run, but at other locations as well. Breaks at locations other than the low point could produce a larger surface area of gasoline for evaporation.

B-18. Considering the topography traversed by the ARCO pipeline (*see* App. Ex. 18), it is clear that given a break in the pipeline at any point between the high points on either side of Possum Hollow Run, the escaping gasoline will flow downhill under the force of gravity toward Possum Hollow Run and thence downstream in Possum Hollow Run (generally to the southwest) to the Schuylkill River. Given a break in the pipeline on the other side of either high point (away from Possum Hollow Run), the escaping gasoline would flow downhill under the force of gravity in a direction generally away from the plant structures, to less proximate drainage systems, and therefore cause lesser effects. Walsh, ff. Tr. 5411, at 4. Thus, the worst case, and therefore the bounding case, that we need only to consider is a break between the high points on either side of Possum Hollow Run.

B-19. The size of a pipe break can, of course, range from a complete double-ended guillotine failure to a small crack. For the complete break, gasoline would be released from the upstream section of the pipe no faster than the quantity pumped per unit time. For the downstream section of the pipe, only that gasoline in the pipe which could flow out of that section under gravity and/or siphoning could escape. Flow under these conditions would be characterized as a gushing as opposed to a spray. For smaller cracks, gasoline would be sprayed at a rate depending on the crack size and existing pressure within the pipe. It is known from experience that under conditions similar to a break in the ARCO pipeline, the sprayed material from a crack can cover a significant area, certainly as much as the order of 9000 square feet. Staff Ex. 9, NTSB-

PAR-76-8, Fig. 3.[5] Assuming such a continuous discharge to be spraying an area on the east bank of Possum Hollow Run and just below the southern high point of the pipeline, the gasoline would then flow downhill to Possum Hollow Run, covering additional terrain. Assuming the area sprayed to be roughly circular, its diameter would be approximately 130 feet. Thus the width of the swath covered by the downward flowing gasoline would be approximately 130 feet. From the site plan (App. Ex. 18) the distance from the postulated break to Possum Hollow Run is approximately 500 feet. The total area on the east bank covered with gasoline would be not more than 500 x 1130 = 65,000 ft^2. In fact, the area would be much less, since the gasoline would flow in rivulets rather than uniformly covering the entire area. Tr. 5723 (Walsh).

B-20. In its initial analysis the Applicant assumed that the quantity of gasoline (4962 gallons) it assumed to be discharged from the break located at Possum Hollow Run was confined to the creek bed between the location of the break and the first downstream bridge in a pool 610 meters long by 1 meter wide by 3 centimeters deep. Walsh, ff. Tr. 5411, at 5. No credit was taken for outflow to the Schuylkill River or for absorption of gasoline into the soil. This 610-square-meter pool corresponds to 6566 square feet. The Staff, in its Supplemental Testimony, postulated the area of the spill from the hillside break as the sum of the area of the spill pathway on the hillside (3 m x 158 m) and the area of the pool 610 meters long, but 3 meters wide, i.e., 474 m^2 + 1830 m^2 = 2300 m^2, or 24,800 ft^2. Ferrell *et al.*, ff. Tr. 7136, at 2. Due to the width of Possum Hollow Run, the Staff considers the assumption of a 3-meter-width water surface of the pool to be conservative by a factor of 2. Tr. 7157 (Ferrell).

B-21. Applicant assumed the evaporation rate of gasoline to be 1 cm/hr, with all the butane being evaporated in the first hour at a uniform rate. From this, Mr. Walsh calculated that 1922 gallons of gasoline evaporated in the first hour. Then, using the explosive limits for gasoline vapor, of 1.3 to 6.0% by volume, he calculated that if layering and gradual upward expansion of the vapors in the valley are assumed (0.06 − 0.013 = 0.047) x 1922 = 90.3 gal. of gasoline would be within explosive limits. For gasoline at 5.75 lb/gal. this corresponds to 519 pounds, which would be equivalent to 5252 pounds of TNT equivalent, if all were detonated. Walsh, ff. Tr. 5411, Attachment 1, at 1-3. The Staff, using a conservative calculational technique to estimate the gasoline evaporation

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[5] From the figure the maximum distance gasoline was sprayed from the SOCAL 8-inch pipeline was approximately 130 feet. The area sprayed approximates one sixth of a circle with a radius of 130 feet. Thus, the area sprayed was approximately $\pi(130)^2/6 = 9000$ ft^2.

rate, and conservative atmospheric temperature and stability assumptions, derived the amount of gasoline vapor assumed to be in the valley to be 773 pounds (approximately 134 gallons). The Staff then, very conservatively, assumed all of this vapor to be in the flammable range and thus equivalent to 1856 pounds of TNT if detonated. Ferrell *et al.*, ff. Tr. 7136, at 5. Applicant initially used a conversion factor for TNT equivalent that was 4 times too great.

f. Overpressure Calculations

B-22. The actual volume of explosive vapor would be distributed over a length of some 600 meters along Possum Hollow Run. Both Applicant and Staff, however, assumed a point source for the blast. Such an assumption is clearly conservative, perhaps by a factor of as much as 10. Ferrell *et al.*, ff. Tr. 7236, at 5-6; Tr. 7158-59, 7263 (Ferrell); Tr. 6187 (Campe); Tr. 7165 (Markee); Tr. 5602 (Walsh). The Staff assumed the location of the point source to be 960 feet due east of the Unit 2 reactor building, whereas the Applicant assumed both 800 feet (where the slope of the valley toward the reactor building is most gradual) and at 550 feet (in the direction of the closest approach of Possum Hollow Run to the Station). Both Applicant and Staff took no credit for shielding effects of the topography on the calculated overpressure resulting at the reactor building from the assumed detonation of all of the explosive mixture. The Applicant's results were 1.9 psi at 800 feet and 3.0 psi at 550 feet (using the incorrect, overly conservative conversion factor for TNT equivalence). Walsh, ff. Tr. 5411, at 7-8; Tr. 5575-78, 5583-88 (Walsh). The Staff calculated a peak reflected blast overpressure, from a detonation 960 feet due east, on the Unit 2 containment building of 1.1 psi for an assumed wind speed of 1 m/sec and 1.2 psi for 2 m/sec. Ferrell *et al.*, ff. Tr. 7136, at 6. For a wind speed of 1 m/sec and 550 feet the Staff calculated 2.1 psi. Tr. 7344 (Campe).

B-23. With respect to the postulated break in the ARCO pipeline, Mr. Hasbrouck's scenario included the following: 42,000 gallons of gasoline sprayed over 10,000 square meters (approximately 108,000 square feet), for which he had no scientific basis, Tr. 5995, 6004, 6100-01, 6115 (Hasbrouck), resulting in 10,500 gallons of gasoline in an explosive mixture. This compares with Applicant's result of 90 gallons and the Staff's conservative estimate of approximately 135 gallons. The sprayed patch of brush and trees on the side of the hill supposedly would generate dense vapor which then slides down the hill. This movement supposedly sucks in fresh air which causes added evaporation. Thus the vapor density supposedly powers a convection current down through the patch. With an unlucky selection of slope, breeze, etc., this convection current consists of an explosive mixture, i.e., any value between 1.3% and 6% by volume. Hasbrouck 1, ff. Tr. 5750, at 2-3.

B-24. Other FOE postulates, i.e., two simultaneous explosions, transport of a flammable mixture to the Schuylkill River and upstream along the railroad track and suction by the cooling towers of an explosive mixture out of Possum Hollow towards the plant, were similarly unsupported. Tr. 5257-58 (Ferrell, Markee); Hasbrouck 2, ff. Tr. 5750, at 3; Tr. 7352-53 (Hasbrouck); Tr. 7353, 7488-89 (Markee).

B-25. The Board assigns no credence to the FOE postulates and resulting calculations of overpressure on the Limerick structures resulting from a breach of the ARCO pipeline. Rather, the Board finds that the peak positive reflected pressure of 2.1 psi calculated by the Staff is conservative.

4. The Columbia Gas Pipelines

a. Description of the Pipelines

B-26. Columbia Gas Transmission Corp. operates two pipelines that transport only natural gas (methane). These pipelines share a common right of way and run parallel to each other 20 to 30 feet apart, generally southwest to northeast through Montgomery County, Pennsylvania (*see* Fig. 1 at the end of this section). Pipeline No. 1278 is 14 inches in diameter. It was constructed in 1949 and operates at a normal pumping pressure of 750 psig and a maximum pumping pressure of 938 psig. Pipeline No. 10110 is 20 inches in diameter. It was built in 1965 and operates at a normal pumping pressure of 1100 psig and a maximum pumping pressure of 1200 psig. Each pipeline was constructed of steel commensurate in thickness and grade with its maximum operating pressure and, when constructed, was buried a minimum of 3 feet below grade. Both pipelines are protected against corrosion by an impressed current cathodic protection system which prevents rusting in the same manner as a battery cathode is protected. Brown, ff. Tr. 5261, at 3-4.

B-27. The nearest compressor stations (i.e., pumping stations) to the Limerick Station are the upstream Eagle Compressor Station, located 9.7 miles south of the point where the pipelines cross the Schuylkill River (6000 feet southeast from the Limerick Station structures) and the downstream Easton Compressor Station located 44.4 miles north of this point. The valves in the pipelines closest to the Limerick Station are at the Schuylkill River and 4 miles north of the river for line 1278 and 4.3 miles north of the river for line 10110. *Id.* at 6. These are manual valves. Tr. 5330-31 (Brown).

B-28. Suction and discharge pressures are monitored at both the Eagle and Easton Stations and by the gas control center at Bethel Park, Pennsylvania. High pressures (938 psi on line 1278 and 1200 psi on line 10110) are designed to cause automatic shutdown of compressors. Tr. 5322 (Brown). Low pressures (425 psi on line 1278 and 770 psi on line 10110) trigger alarms at the control centers and at the Eagle and Easton Stations. Tr. 5321 (Brown). If a low pressure alarm occurred, the compressor units would be shut down manually and no additional gas would be introduced into the lines. Tr. 5288 (Brown). Under worse conditions, where a line break or large leak occurs in the middle of the night and crews must be called out, it was estimated that valves could be closed and the flow of gas stopped within approximately 2 hours. Brown, ff. Tr. 5261, at 6. Neither line 1278 nor line 10110 has experienced any leak or rupture in the history of its operation. *Id.* at 6. Breaks in other natural gas lines of similar design, structure and usage have occurred. In 1960, a 30-inch pipeline operating at 936 psig suffered a linear fracture of approximately 625 feet. A fire occurred at the moment of rupture, burning trees and landscape 400 to 500 feet on either side of the line, but no damage occurred beyond 500 feet. In 1982, a 10-inch pipeline operating at about 980 psi completely severed, resulting in an instantaneous fire which burned trees and the landscape 250 to 300 feet on either side. Brown, ff. Tr. 5261, at 6.

b. Contents of Pipelines

B-29. The Columbia Gas pipelines transport only methane in the gaseous state. There are no plans to transport either propane or butane and the existing compressors would have to be replaced before these materials, in either gaseous or liquid form, could be transported in any event. Tr. 5318, 5325-27, 5341, 5349-50 (Brown). Further, approval by the Federal Energy (Regulatory) Commission would be required to transport anything other than natural gas. Tr. 5349 (Brown).

c. Location of the Pipelines

B-30. The Columbia Gas pipelines cross the Schuylkill River at a point approximately 6000 feet from the Limerick Station structures and proceed approximately in a straight line somewhat north of northeast for more than 2½ miles. Staff Ex. 6 (SER), Fig. 2.6. The actual location, at their closest approach to the Limerick site, is depicted in Applicant's Ex. 18 from which it can be determined that the closest approach is at least 3400 feet. Applicant verified that the closest approach is approximately

3500 feet. Payne, ff. Tr. 5357, at 7-10. His attempt to determine the possible error in the location of the pipelines from comparison of a U.S. Geological Survey map and photogrammetric interpretation of pipeline traces and Columbia Gas Transmission Company plans indicated possible mean errors ranging from 15 to 51 feet. Payne, ff. Tr. 5357, at 8, 9. Intervenor FOE/Anthony indicated that he had a lot of confidence in Applicant's site plan and that even if the location of the pipelines were off by 100 feet, he didn't think that would be a controlling factor. Tr. 5361 (Anthony). We agree.

d. Nature of the Release

B-31. Disregarding the reality or probability of a break in the larger (20-inch) pipeline, for purposes of analysis, a double-ended rupture was assumed by the Applicant to occur at the closest approach of the pipeline to each of the safety-related structures of the Limerick plant. Boyer *et al.*, ff. Tr. 8213, at 6, 7. For such a break it would be possible for the entire contents of the pipeline between the Eagle Compressor Station and the Easton Compressor Station to be released. Since the gas is immediately dispersed in the atmosphere by its own momentum, by diffusion and by wind, the nature of the cloud formed that is potentially explosive depends upon the rate at which the gas is released, not upon the total quantity released during an incident. Thus, it is irrelevant whether or not the compressor stations are shut down after the breaks. The rate of release of gas from a break depends upon the size of the opening in the pipe and the sonic velocity of the released gas. Walsh, ff. Tr. 5411, at 11.

e. Formation of Flammable Mixture

B-32. When the gas is first released from the pipe, the concentration of methane in air is too rich to be flammable or explosive. As the gas disperses into a cloud, the concentration decreases to the upper limit of flammability and continuing dispersal reduces the concentration below the lower limit of flammability. The flammable limits of natural gas are between 6 and 14% by volume in air. Walsh, ff. Tr. 5411, at 12. This dispersion is a continuous process, so that for a constant rate of release of gas, a constant stability condition and constant temperature of the ambient atmosphere and a constant wind speed, a fixed region in space will result within which the methane-air mixture will be within flammable limits. The dimensions of this region define the amount of methane that could burn or explode.

B-33. To calculate conservatively the potential blast and heat effects on the Limerick structures, the Applicant made a number of conservative assumptions. First, the maximum openings in the two ends of the ruptured pipe were assumed to be the full cross-sectional area of the pipe. Second, both pipe ends were assumed to be forced into a vertical orientation. Any other configuration would result in additional turbulence and consequent increased dispersion, causing the point at which the methane-air mixture decreased below the flammable limit to be further from the Limerick plant. Walsh, ff. Tr. 5411, at 11; Tr. 5424 (Walsh). Third, Applicant conservatively assumed an atmospheric stability of Pasquill "F," an inversion condition. Atmospheric conditions actually are more conducive to dispersion 95% of the time. Fourth, Applicant assumed a 1-m/sec wind, moving the gas cloud directly toward the Limerick Station, during Pasquill "F" conditions, a situation that occurs only 0.004% of the time. Walsh, ff. Tr. 5411, at 10, 11; Tr. 5432-35, 5458, 5470 (Walsh). If the wind were blowing in any other direction, the effects of a potential detonation on the Limerick facility would be less, since the location of the detonation would be further from the Station. Similarly, if the wind speed were higher, greater dilution of the methane-air mixture would occur and the region of flammability would be further from the Station. Walsh, ff. Tr. 5411, at 12. Fifth, Applicant assumed the escaping gas first rose above the ground level from momentum velocity to an elevation of approximately 500 feet, before traveling toward the plant. Tr. 5421 (Walsh). This assumption results in the maximum concentration of the methane-air mixture to occur as far downwind as possible. If the mixture traveled at ground level there would be more mixing with air which also would cause the region of flammability to be further from the plant. Walsh, ff. Tr. 5411, at 12; Tr. 5463-65 (Walsh).

B-34. The Applicant calculated the concentration of natural gas in the atmosphere both downwind, crosswind and vertically as a function of distance at 100-meter intervals downwind from the source of natural gas, under the assumed conservative conditions. From the results of these calculations, Applicant calculated the volume of the region in which the methane-air mixture would be within explosive limits to be 3.74×10^5 m^3.[6]

[6] Volume of ellipsoid = $V = 4 \times abc/3$, where a, b, and c are the lengths of the semi axes. $a = 840/2 = 429$ m, $b = 50/2 = 25$ m, and $c = 25/2 = 12.5$ m, for the ellipsoid whose surface corresponds to the points where the concentration of methane is at 4.31×10^7 micrograms/m^3, the lower explosive limit. $a = 480/2 = 240$ m, $b = 35/2 = 17.5$ m, $c = 20/2 = 10$ m, for the ellipsoid whose surface corresponds to the points where the concentration of methane is at 1.01×10^8 micrograms/m^3, the upper explosive limit. Walsh, ff. Tr. 5411, Attachment 3, at 3-5.

f. Overpressure Calculations

B-35. Assuming the average concentration of the gas within the upper and lower explosive limits to be (14% + 6%)/2 = 10%, the volume of natural gas contained within the volume of detonable mixture is 0.10 x 3.74 x 10^5 = 3.74 x 10^4 m^3. Also, assuming the density of methane to be 0.0448 lb/ft^3 at 0°C, this volume is equivalent to 5.92 x 10^4 lb of natural gas at explosive mixture concentration, or 347 tons of TNT equivalent. Walsh, ff. Tr. 5411, Attachment 3, at 3-5. Since the density of methane decreases with increasing temperature, the assumption of 0°C is conservative most of the time and would not affect the result significantly if the temperature were below 0°C.

B-36. Using Staff Ex. 7 (NRC Regulatory Guide 1.91) and assuming that the explosion centroid is located at an elevation 500 feet above ground and approximately 700 meters downwind (toward the Limerick Station structures, which would be approximately 1200 feet from the Unit 2 containment building), triggered by some undefined high-energy ignition source, the calculated peak positive normal reflected pressure was determined to be 10 psi at the nearest safety-related structure, i.e., the Unit 2 reactor building. Walsh, ff. Tr. 5411, at 5. Additional conservatisms (*see* B-33, above) in this analysis include:

a. break at exactly the nearest point of approach to the Limerick Station.
b. vertical rise of the gas column to 500 feet above plant grade (where the momentum energy decays), without dilution. Tr. 5428 (Walsh).
c. natural gas clouds seldom, if ever, detonate in an unconfined space.
d. it is difficult to hypothesize an ignition source to trigger a detonation in an elevated cloud.

B-37. FOE postulated a number of conditions which it alleged would cause a flammable mixture to be transported to the vicinity of the Station, i.e., Possum Hollow. These included the assumption of a negatively buoyant (i.e., much colder than ambient) cloud being transported to reach the closest location to the Station.[7] FOE performed no calculations and did not provide any credible technical basis to support this postulation. Tr. 5990-94, 6085-86 (Hasbrouck). In fact, practical experience in purposely blowing down a natural gas pipeline indicates a reduction in temperature of the gas of 7°F/100 psi reduction in pressure, but

[7] At 0°C the density of air is 0.081 lb/ft^3; the density of methane is 0.045 lb/ft^3. Walsh, ff. Tr. 5411, Attachment 3, at 1.

the gas does not stay cold because of immediate mixing with the air around it. Tr. 5298, 5346, 5353-54 (Brown); Tr. 5430 (Walsh).

B-38. Consideration also was given by the Applicant to simultaneous rupture of both Columbia Gas pipelines, notwithstanding the lack of basis for such a postulated event. Enhancement of the effects resulting from the simultaneous rupture of the 14-inch line and of the 20-inch line would be minimal because of several factors. The difference in diameters and the difference in operating pressures would cause the two plumes to enter the atmosphere at different elevations, causing the zones of flammability to occur at different distances from the Station. Thus, for simultaneous detonations or simultaneous rupture, the overpressure effects would arrive at the Station at different times and therefore not be directly additive. Merging of the two plumes, which could only take place under much less favorable meteorological conditions, would result in the flammable mixture being located closer to the point of release, reducing any overpressure effect. Tr. 5604-05, 5727-28 (Walsh).

B-39. With respect to the Columbia pipelines, Mr. Hasbrouck *assumed* 350 tons of TNT equivalent at a distance of 800 feet. Hasbrouck 1, ff. Tr. 5750, at 4. Applicant calculated 347 tons of TNT equivalent using a TNT equivalence factor of 10, which is 4 times too great according to Regulatory Guide 1.91, Rev. 1 (Staff Ex. 7). Ferrell, ff. Tr. 9401, at 5; Tr. 7467 (Campe); Tr. 9170 (Ferrell). Staff used a TNT equivalence factor of 2.4 to obtain 71 tons and used the Applicant's calculated horizontal distance to the cloud centroid of 1200 feet. Ferrell, ff. Tr. 9041, at 6-9; Tr. 9138, 9147 (Ferrell). Mr. Hasbrouck *chose* 800 feet, by assuming the methane gas would not rise above ground until after reaching Possum Hollow Run and then rising before detonation. Hasbrouck 1, ff. Tr. 5750, at 3-4. In fact, he believed it was possible for a flammable mixture to be caused by a break in the pipeline where it crosses Possum Hollow Run and to travel 5500 feet and remain in a concentration that would be flammable. He did not have a technical basis for this (scenario) and characterized it as half-baked. Tr. 6008-09 (Hasbrouck). The Board gives no weight to this testimony and finds the testimony of the Applicant and Staff to be credible and uncontroverted with respect to the overpressure and radiant heat load impacts of potential ruptures of the ARCO and Columbia pipelines on the Limerick Station.

B-40. For further explication of the Applicant and Staff results of overpressure calculations, we provide, as Figs. 2, 3 and 4, tabular summaries of overpressure calculations. Boyer *et al.*, ff. Tr. 8213, Tables I and II and Staff Ex. 23. Using the correct value for TNT equivalence, the maximum overpressure calculated by the Applicant was 8.3 psi from an air burst on the reactor building and diesel generator building exterior walls (Fig. 3). The comparable calculations by the Staff resulted in overpressures of 7.4 psi on the diesel generator building Unit 2 exterior wall and 7.3 psi on the reactor building Unit 2 exterior wall (Fig. 4). Figure 2 values were calculated using the conservative (by a factor of 4) value for TNT equivalence.

5. *Radiant Heat Load Calculations*

a. ARCO Gasoline Pipeline

B-41. Both the Applicant and the Staff calculated the radiant heat load on the Limerick Station safety-related structures resulting from burning gasoline released from the ARCO pipeline. The Applicant's calculation assumed that the total amount of gasoline contained in the pipeline between high points adjacent to the break (4962 gallons) burned in 15 minutes. The 15-minute period was conservatively used to maximize the heat generation rate. Walsh, ff. Tr. 5411, at 8. Based on 20,000 Btu/lb of gasoline, this would amount to 5.71×10^8 Btu released in 15 minutes or at a rate of 2.28×10^9 Btu/hr. *Id.*, Attachment 2, at 5-6. The radiant heat may be calculated using the formula, *id.* at 5,

$D = (FQ/(4 K))^{½}$, where
D = distance in feet from flame midpoint to receptor
F = fraction of heat radiated
Q = heat release in Btu/hr
K = heat radiated in Btu/ft^2-hr,
$D^2 = FQ/12.57\,K$
$K = FQ/12.57\,D^2$
For F = 0.3 (based on Butane values)
D = 800 feet, the distance to Possum Hollow Run in the direction in which the valley wall is least steep on the Station side, to minimize the effects of shielding by the valley wall.
$K = 0.30 \times 2.28 \times 10^9/12.57 \times 6.4 \times 10^5$
= 85 Btu/ft^2-hr. This is equivalent to approximately 270 W/m^2.

B-42. Applicant also calculated the radiant heat load on the Unit 2 reactor building arbitrarily assuming 21,000 gallons of gasoline burned in 15 minutes, a scenario it does not believe to be credible, to demonstrate the effects of 4 times as much gasoline burned as in its original calculations. Using the same method and 800-foot distance, the result was 350 Btu/ft^2-hr. Walsh, ff. Tr. 5411, at 9. This would be approximately 1100 W/m^2.

B-43. The Staff's calculation proceeded differently. It believes that ignition of a gasoline vapor cloud would cause burning in less than 1 minute, or would flash back to the point of issuance of gasoline from the pipe rupture. This was considered reasonable, since the liquid gasoline on the hillside and along the creek would be rapidly consumed. Ferrell *et al.*, 7136, at 12. It believes the potential thermal effects of such burning would be insignificant because of the distance from the Unit 2 reactor building and because of the expected short duration of the fire. To estimate the radiant heat from a sustained fire of the gasoline issuing from the rupture, it assumed a 100-foot-diameter vertical column of burning gases located at the pipe break, i.e., at the nearest approach of the pipeline to the Unit 2 reactor building, a distance of 1625 feet. The result was 265 W/m^2. Ferrell *et al.*, ff. Tr. 7136, at 12-13; Tr. 7431 (Ferrell).

B-44. The Staff noted that the average solar flux in Washington, D.C., is 170 W/m^2 and the peak solar flux in Albuquerque, N.M., is in the range of 1000 to 1250 W/m^2. *Id.*

B-45. The Board finds, based on the uncontroverted testimony of the Applicant and Staff, that the radiant heat load on the safety-related structures of Limerick Station resulting from burning gasoline released from a rupture of the ARCO pipeline will not pose an undue hazard to the Station.

b. Columbia Gas Pipelines

B-46. With respect to a rupture of the Columbia 20-inch gas pipeline, the Applicant calculated the radiant heat load on the safety-related structures of the Limerick Station using the same formula as above.

B-47. Applicant assumed the heat release to be the volume of gas burned per second times the heat content released per unit volume, i.e., 4800 $ft^3/sec \times 1050\ Btu/ft^3 = 5.04 \times 10^6$ Btu/sec or 1.814×10^{10} Btu/hr. Walsh, ff. Tr. 5411, Attachment 2, at 1. The record does not show the basis for the 4800-ft^3/sec number, but the heat release clearly is conservative, since the Applicant assumed extended burning of the vapor cloud at its closest approach to the Station. Assuming that the cloud burns at 1200 feet from the Station,

$$K = 0.25 \times 1.814 \times 10^{10}/12.57 \times (1200)^2$$
$$= 250\ Btu/ft^2\text{-}hr$$

B-48. The Staff also calculated the consequences of burning of natural gas released from the 20-inch Columbia pipeline. It considered a double-ended rupture occurring at the closest approach (3500 feet) of the pipeline to the Station, resulting in a natural gas fireball of 300-foot diameter and infinite height. The 300-foot diameter is believed by the Staff to be characteristic of previous experience. Even if the initial diameter were larger, it would diminish in seconds and the Staff analysis assumed sustained burning over a long period of time. The infinite height was assumed for calculational simplicity. Tr. 7436-37 (Campe). The Staff concluded that the potential heat flux from a burning natural gas cloud would be insignificant with respect to the plant structures. Campe, ff. Tr. 6131, at 3. This conclusion is corroborated by reference to Staff Ex. 14, NUREG/CR-1748, which estimates the thermal radiation (mean emissive power) from a turbulent methane flame to be 100 kW/m². Using the formula, *id.* at F-2,

$\bar{F} \doteq F \;;\; F(D/r)^2$, where
$\bar{F}$ = radiant heat at the receptor
F = radiant heat at the flame edge
D = diameter of flame
r = distance from flame to receptor
τ = transmissivity of the atmosphere

And using a conservative value of τ as 0.66, *id.* at F-3, a diameter of 300 feet and a distance of 3350 feet,

$$\begin{aligned}\bar{F} &= 100(300/3350)^2 \times 0.66 \\ &= 0.802 \times 0.66 = 0.53\ \mathrm{kW/m^2} \\ &= 530\ \mathrm{W/m^2}\end{aligned}$$

B-49. This is the result reported in the SER, Staff Ex. 6, at p. 2-13. While comparable to solar heat radiation, the effect on Station structures would indeed be insignificant.

6. *Effects of Postulated Detonation on Safety-Related Structures*

B-50. In response to a request by the Board, the Applicant and Staff analyzed the ability of safety-related structures at the Limerick Generating Station to withstand the effects of postulated detonations resulting from the assumed rupture of the ARCO and Columbia Gas transmission pipelines. The Board expressed an interest in both the ability of the structures to withstand such postulated detonations and the margins of structural safety above the calculated blast overpressures inherent in the design of the structures. Tr. 5934-35. Evidentiary hearings on the ability of the structures to withstand the postulated explosions and the margins of structural safety took place on March 8, 9, and 20-23, 1984.

B-51. In assessing the ability of a structure to resist the effects of explosions, the effect to be considered is the resulting pressure on the structure. This pressure (or overpressure) is in the form of a shock wave which expands through the air radially from the center of the explosion and diminishes with distance. As the shock wave impinges on the structure, the structure will experience a structural loading. The magnitude of the loading is measured in units of pressure — commonly pounds per square inch (psi). Given the size of the explosion in TNT equivalence and the distance to a given structure, the overpressure on the structure in psi can be calculated. The structure can then be assessed as to its ability to withstand the applied overpressure loading. Both Applicant and Staff, using conservative explosion scenarios, assessed the ability of the safety-related structures at the Limerick Station to withstand the postulated explosions. Boyer *et al.*, ff. Tr. 8213; Ferrell, ff. Tr. 9041; Kuo and Romney, ff. Tr. 9043.

B-52. Applicant calculated the highest overpressures that would result from the worst-case ARCO or Columbia Gas pipeline explosion on the roof and exterior walls of each safety-related structure. Boyer *et al.*, ff. Tr. 8213, at 6-13. *See* Fig. 2 at the end of this section.

B-53. The pressures resulting from the postulated rupture and detonation of gasoline from the ARCO pipeline were always significantly less than that resulting from an assumed detonation of the vapor from the Columbia Gas transmission line rupture. The maximum peak positive reflected pressure from an ARCO pipeline explosion calculated by the Applicant (Walsh) was found to be 1.9 psi. *Id.* at 7.

B-54. For the postulated Columbia Gas pipeline rupture, both Staff and Applicant utilized the methodology set forth in Regulatory Guide 1.91 (Rev. 1), for determining TNT equivalency to hydrocarbons and graphs provided in the Army Technical Manual TM 5-1300 "Structures to Resist the Effects of Accidental Explosions." *Id.* at 6-11; Ferrell, ff. Tr. 9041, at 2. Staff Ex. 7 and 20. The peak pressures shown as design/assessment values for the Columbia pipeline explosion in Applicant's Table I (*see* Fig. 2 at the end of this section), represent the maximum pressures that would be developed assuming a surface burst and a detonable mixture approximately 4 times that suggested by Regulatory Guide 1.91 (Rev. 1). Applicant recalculated the blast overpressures in accordance with the guidance of Regulatory Guide 1.91 (Rev. 1). The recalculated values are shown in cols. 1 and 2 of Applicant's Table II (*see* Fig. 3, attached), and are lower than the values in Table I. The pressures used in Applicant's structural margin assessments were taken from Table I and represent an additional conservatism. The highest overpressure for a Columbia gas explosion shown in Table I is 10 psi while the highest value shown in cols. 1 or 2 of Table II is 8.3 psi. Boyer *et al.*, ff. Tr. 8213, at 7, Tables I and II.

B-55. Neither Staff nor Applicant agreed that the detonation of unconfined or open-air natural gas cloud is a credible event. Ferrell, ff. Tr. 9041, at 2, and Tr. 9066; Boyer *et al.*, ff. Tr. 8213, at 5. Uncontroverted evidence established that unconfined natural gas can only be detonated with high energy sources such as TNT and even then with difficulty. No such sources of energy are known to be available at the Limerick site. Tr. 6157-58, 7423, 7450-52 (Campe)

B-56. Regardless of the evidence presented as to the improbability of an open-air gas detonation, as a conservatism, both Applicant and Staff assumed a gas explosion at a horizontal distance of 1200 feet from the structure and at 500-foot elevation, the maximum height to which the natural gas could rise as a result of momentum from the postulated pipeline breach. The Board notes that no sources of ignition exist at 500 feet, let alone a source of sufficient energy to cause a detonation. Boyer *et al.*, ff. Tr. 8213, at 6, 8; Ferrell, ff. Tr. 9041, at 2.

B-57. Applicant also calculated overpressures assuming an air burst and a surface burst. From these calculations, Applicant determined that estimated overpressure produced from the postulated TNT-loaded railroad boxcar explosion used in the design basis and elevated natural gas (500-foot elevation) explosions were greater than those of all other postulated pipeline scenarios. Boyer *et al.*, ff. Tr. 8213, at 11.

B-58. Staff and Applicant calculations for the 500-foot-elevation gas explosion and employing the guidance used in Regulatory Guide 1.91 (Rev. 1) are in close agreement. Tr. 8815 (Walsh); Tr. 9067-68 (Ferrell). Any differences in the numbers are attributed to the analyst's accuracy in picking the numbers off the table in Army Technical Manual TM 5-1300. Tr. 8815 (Vollmer). The comparable values are contained in col. 2 of Applicant's Table II and col. 1 of Staff's Table 1 (Boyer *et al.*, ff. Tr. 8213, and Staff Ex. 23, ff. Tr. 9055, respectively). The largest difference between comparable Applicant and Staff Columbia blast overpressure calculations was 1.0 psi (for the reactor building wall). This is larger than might be expected to result from inaccuracy in reading values from a graph. The difference might be explained by the Staff's use of 1300 feet as the distance from the structure. Ferrell, ff. Tr. 9041, at 7. It appears that Applicant used a horizontal distance of 1200 feet in its calculations, not the slant distance of 1300 feet. Boyer *et al.*, ff. Tr. 8213, at 6.

B-59. Staff calculations indicated that the railroad boxcar explosion generated greater overpressures than any postulated explosions of either the ARCO or Columbia Pipeline materials. Ferrell, ff. Tr. 9041, at 10 and Table 1 (Staff Ex. 23), ff. Tr. 9055. (Fig. 4 of this Decision.)

7. *Margin Analysis of Margins of Structural Integrity to Postulated Overpressures*

B-60. After determining the critical overpressure for each safety-related structure (Reactor Buildings and Diesel Generator Buildings for Units 1 and 2, the Control Building and the Spray Pond Pumphouse), Applicant identified the critical wall of each structure and the critical element of that wall for detailed analysis. The critical element selected was a 1-foot-wide beam element with fixed ends. This is a conservative selection of the critical element because if the wall slab had been evaluated as a whole rather than as a beam section, considerable additional support would have been provided by the adjacent walls. Tr. 8417, 8479-81, 9018 (Vollmer); Kuo and Romney, ff. Tr. 9043, at 4.

B-61. Applicant then isolated the 1-foot-wide wall strip and applied the highest determined overpressure as a uniform load on the length of the strip. The criterion used for structural adequacy was the ductility ratio of the element. Tr. 8822-23 (Wong).

B-62. The response of a structure or structural member to load is deformation. Loading up to a certain level results in elastic deformation. For any loading imposed up to the elastic limit, the structure will return to its original shape when the load is removed. Any loading greater than the elastic limit puts the material into the plastic range and results in permanent deformation. Materials or structural elements that have deformed into the plastic range will not return to their original shape. Ductility is the ability of a structure or structural member to deform beyond its elastic limit without rupturing. The "Ductility Ratio" is the ratio of the total deformation (elastic plus plastic) to the deformation that would occur at the limit of the elastic range. Kuo and Romney, ff. Tr. 9043, at 5.

B-63. Applicant calculated the ductility ratios for the loaded critical sections and compared the calculated values against the maximum code allowable, which is forth set in Regulatory Guide 1.142 as a mid-span ductility ratio of 3.0 and an end-point ductility ratio of 10. Tr. 8948 (Palaniswamy).

B-64. After applying the maximum blast overpressures to the structures and calculating the ductility ratios, the ratios were compared with the code-allowable value of 3.0 for mid-span and 10.0 for the end-point ratio. In all cases the determined ductility ratios were within the limits established by the code. The highest mid-span ratio calculated was 2.2

and the worst-case end-point ratio was 2.9. Tr. 8947-48 (Palaniswamy); Tr. 9069 (Kuo).

B-65. The Applicant then determined the blast overpressure that would cause deformation up to a ductility ratio of 3.0 at mid-span and compared that value with the calculated blast overpressure. The result was expressed as a percent of margin. Boyer *et al.*, ff. Tr. 8213, at 13-15; Tr. 8822-24 (Wong).

B-66. Staff did not make independent calculations of ductility ratios, margins, or shear and moment calculations of the safety-related structures. They did, however, make a detailed review of the assumptions, models, techniques and methodologies employed by Applicant and found them to be appropriate and conservative. Kuo and Romney, ff. Tr. 9043, at 3-4; Tr. 9069-70, 9221 (Romney); Tr. 9206-08, 9221-23 (Kuo).

B-67. Regarding the conservatism of the bounding ductility ratio of 3.0 for mid-span deformation, tests have indicated that beam elements, such as the wall panel strips used in the structural analysis here, do not actually fail until they reach ductility ratios of 20 and beyond. Tr. 9019-20 (Palaniswamy). The one-way slab analysis, used by Applicant in its assessment, rather than a two-way analysis, is conservative in that no credit is taken for support from adjacent walls. If a two-way analysis were to be used, the structural safety margins would be larger. Tr. 9206-07 (Kuo); Tr. 8417, 9018 (Vollmer). The calculated safety margins are not predicated on the ultimate failure threshold of the structure. They are based on code values acceptable for structures of the type considered here. Accordingly, some additional unquantified safety margin above the calculated margins exists for these structures. In Applicant's Table II (ff. Tr. 8213) (Fig. 3, attached), a comparison of cols. 3 and 4, respectively, which are the pressures calculated using the conservative TNT equivalent (by a factor of 4), with the pressures used in structural assessment (col. 5), margin is shown to be available in both the reactor building and the diesel generator building. For the control structure and the spray pond pumphouse the values of 4 times the Regulatory Guide values exceed the structural assessment values. For those cases, using the proper TNT conversion factor, margins do exist, as is apparent from the values listed in column 2 of Fig. 3. Applicant's demonstration of a structural safety margin for the reactor and diesel generator buildings even when using 4 times the TNT-equivalent explosion suggested by Regulatory Guide 1.91 (Rev. 1) is a significant additional conservatism in assessing the adequacy of the Limerick structures to resist the effects of blast overpressures. Boyer *et al.*, ff. Tr. 8213, at 12, 13; Tables I and II, ff. Tr. 8213.

B-68. Applicant also conducted an evaluation of the global response margins inherent in the design of the safety-related structures at Limerick. This evaluation consisted principally of a determination of the overturning moment and story shear on entire structures as a result of the postulated explosions and a comparison with the moments and shears resulting from the design basis safe shutdown earthquake (SSE). In each case, the overturning moment and the story shear associated with the SSE were found to be larger than that associated with the postulated explosions. Since the plant has been designed to withstand the safe shutdown earthquake loading values, there is more than adequate structural capacity to resist the forces associated with the postulated explosions. Global response safety margins were calculated by dividing the SSE loading values by the loading values calculated as a result of the explosions. Kuo and Romney, ff. Tr. 9043, at 8 and 9; Tr. 9361-62 (Kuo); Vollmer *et al.*, ff. Tr. 8213, at 11; Tr. 8824-26 (Wong); Tr. 8826-27 (Vollmer).

8. Factors Allegedly Not Considered in Margin Analysis

a. General

B-69. FOE alleged that the Applicant's margin analysis did not consider the effects of deadload, vibratory loads, inside/outside pressure and temperature differentials, hydrostatic pressure and differential settlement on the safety-related structures at the Limerick Generating Station. Testimony indicated that each of these factors was adequately considered. Tr. 8368-83, 8442-54, 8463-73 (Wong, Boyer, Vollmer, Palaniswamy, Walsh, Benkert); Tr. 9181-9247 (Romney, Kuo).

B-70. Regarding the consideration of gravity and deadload, uncontroverted evidence established that the deadload consisting of the weight of the walls and equipment attached thereto is transmitted to the ground as a vertical compressive load. Since the forces associated with the postulated explosions would act horizontally and thus perpendicular to the walls, the effect of the deadload and the blast overpressure would not be directly additive. Tr. 8442-45 (Vollmer, Palaniswamy); Tr. 9201 (Romney). Structural members are designed for combination of deadload, liveload, earthquake and tornado loads. Forces resulting from the appropriate load or loads are combined with the blast overpressure and were considered in the margin calculations. Tr. 9236-37 (Kuo), Tr. 9202-03 and 9245 (Romney). Applicant's witnesses further testified that the compression resulting from deadload is actually beneficial in terms of the ability of a structural wall to withstand bending since it acts as a pre-stress. Tr. 8445 (Palaniswamy). The roof slab deadload acts in the same direction as a downward-acting blast pressure and was therefore considered additive as appropriate. Tr. 8372 (Vollmer); Tr. 8442-43 (Palaniswamy); Tr. 8442-45 (Vollmer).

B-71. FOE's allegation that vibratory load from equipment operating within the reactor building was not considered in the structural analysis was likewise unsupported by the evidence. Tr. 8372-73 (Vollmer, Palaniswamy). Evidence indicated that vibratory loads were considered and found to be negligible. Tr. 8374, 8378-79 (Palaniswamy). Applicant's witnesses further testified that any portion of the vibratory load not eliminated by the damping effect of the 1½- to 2-foot-thick floors would primarily be transferred from the floor slab to the supporting beams and columns, thus leaving the wall slabs largely unaffected. Tr. 8375 (Boyer); Tr. 8377 (Wong). The roof slabs would not experience vibratory loading since there is no moving equipment on them. Tr. 8378 (Wong); Tr. 8378-79 (Palaniswamy).

B-72. FOE's claim that Applicant's margin analysis did not examine pressure or temperature differentials between the interior and exterior of the reactor building was also found to be without merit. The evidence indicated that the reactor building is operated under a negative pressure of about 0.01 psi to prevent releases from escaping the building. Such a small pressure difference would have no effect on the results of a detonation or on the margin analysis. Tr. 8446 (Vollmer). As regards temperature differences, the evidence indicates that temperature loading is considered in the design of safety-related structures as required by Regulatory Guide 1.142, but is not required to be considered in the analysis of blast overpressures. Tr. 9181-83 (Romney). Further, any difference between the inside and outside temperatures would have a negligible effect on the margin analysis since the containment wall is over 30 inches thick and is well insulated from temperature changes. Tr. 8447-50 (Vollmer).

B-73. Hydrostatic forces were considered in the design of below-grade walls of the safety-related structures at Limerick. Tr. 8463-64 (Vollmer); Tr. 9189-92 (Romney). Both Applicant and Staff testified that hydrostatic pressure exerts force only on the portions of the wall that are below grade level. Walls above grade level are not affected by hydrostatic pressure. In evaluating the effects of an explosion on a building structure, only the walls above grade need be considered. Tr. 8464, 9191-96, (Kuo, Romney); Tr. 8468-69 (Vollmer).

B-74. FOE's allegation that differential settlement was not considered is without merit. Stresses that would be caused by differential settlement were considered in the design of the structure. The Limerick structures, however, are located on a competent rock foundation and on foundations of this type there is no differential settlement. Tr. 8469 (Vollmer); Tr. 9215-17 (Romney).

b. Reactor Building Openings

B-75. FOE postulated that the blast wave would enter the reactor building through a 9-foot-high by a 40-foot-wide louver in the south wall and/or a 2-foot by 2-foot roof opening of the reactor building and damage the safety-related equipment and systems inside. Both Applicant and Staff testified that the louver in the south wall is not safety-related and opens into a compartment which houses nonsafety-related HVAC equipment. Its failure would in no way affect the integrity of the reactor building or the ability to safely shut down the facility. Tr. 9110-13 (Kuo, Romney, Lefave); Tr. 9132-33 (Kuo, Romney); Tr. 8956-57 (Wong). Additionally, the walls surrounding the compartment housing the HVAC equipment are 1 foot thick and would resist any residual overpressure that is not absorbed by the louver. Tr. 9114 (Kuo); Tr. 8955-58, 8965 (Wong). Applicant's calculations indicate that even if the pressure from an explosion were not absorbed in any way, by the louver, inter-compartment walls or plenum, the average pressure inside the reactor building would increase by no more than 0.016 psi and would have a negligible effect on the building and any equipment contained therein. Tr. 8965-66 (Walsh). By comparison it takes 0.1 psi to break a normal house window. Tr. 8958 (Ashley).

B-76. The 2-foot-square roof opening in the reactor building which is covered by a sheet metal blowout panel is designed to relieve pressure inside the building and does not serve any structural purpose. Tr. 8959-60 (Wong). Even if the sheet metal blowout panel were displaced, the resulting pressure differential would be insufficient to dislodge any pipes that might be nearby and the pressure wave would quickly be reduced to ambient as it expanded inside the large volume of the reactor building. The increase in pressure within the building's interior would be less than 0.01 psi. Tr. 8960-61 (Ashley); Tr. 8960-63 (Wong, Ashley).

B-77. The sheet metal buildings on the north and south sides of the reactor building roof could conceivably be damaged by a postulated natural gas explosion. These buildings, however, are not required for the safe shutdown of the Station and, even if destroyed, would not provide an opening into the reactor building since the conduits passing between these buildings and the reactor building are sealed and would not be affected by an explosion. Tr. 8969-70 (Wong).

c. Effect of Detonation on Underground Structures

B-78. Applicant and Staff also determined that the blast pressure or deflagration would have no effect on underground related structures or equipment since buried safety-related pipes and ducts must have a minimum cover of 4 feet of soil or the equivalent in concrete or other material. Kuo, Romney, ff. Tr. 9043, at 11; Tr. 8864-65 (Boyer). Four feet of soil or equivalent cover can withstand a minimum of 3000 to 4000 lb/ft^2, which is an order of magnitude greater than the load that would result in any of the postulated explosions. Similarly, the manhole and duct-to-bank covers are at least that strong since they are designed for high-impact loads such as would result from a tornado missile. Tr. 8805-06 (Wong); Tr. 8806 (Vollmer).

9. The Effects of a Postulated Cooling Tower Collapse

B-79. FOE speculated that the cooling towers would rotate about their base and overturn from explosive forces, thereby causing potential damage up to a radius of greater than the 550-foot height of the towers. Both Staff and Applicant testified that this event is highly unlikely because the relatively thin-shelled cooling tower structure is not likely to maintain its rigidity as it collapses. Kuo and Romney, ff. Tr. 9043, at 11; Tr. 9278, 9284-85 (Romney); Boyer *et al.*, ff. Tr. 8213, at 15, 16.

B-80. Applicant postulated a concrete missile 5 ft x 5 ft x 5 ft resulting from the failure of a cooling tower falling directly onto buried safety-related piping. Using conservative assumptions (200-ft/sec velocity as compared to a free-fall velocity of 188 ft/sec from the top of the 550-foot tower and orientation such that the corner strikes the ground first), Applicant calculated that the concrete section would only penetrate 2.8 feet into the soil and would not affect the safety-related facilities buried below. The analysis further showed that the impact would not overstress the buried pipes or concrete duct banks due to compression. The analysis included the duct-bank manholes which would be adequately protected by their steel and concrete covers. Boyer *et al.*, ff. Tr. 8213, at 16-17. Staff agreed with Applicant's analysis, stating also that it is conservative in that the cooling tower collapse would likely produce much smaller pieces of debris than assumed by Applicant. Kuo and Romney, ff. Tr. 9043, at 11-12.

B-81. FOE then postulated several scenarios involving pieces of cooling tower debris. One such scenario involved steel reinforcing rod by itself or extending from a dislodged concrete section penetrating greater than the 2.8 feet calculated by Applicant and causing damage to buried structures. Unrebutted evidence established that individual steel rods will not fall separately or protrude in any significant length from broken pieces of concrete. Tr. 8876 (Vollmer), Tr. 8876-77 (Buchert).

B-82. FOE also speculated that the 70-foot-tall column supporting the cooling tower and the 500-kV transmission towers would also fail and penetrate nearby buried safety-related structures. Evidence established that the 70-foot cooling tower support columns would pivot on their bases and fall, penetrating about 1 foot into the ground. Since the nearest buried safety-related structures are 100 feet away and buried at a minimum of 4 feet or equivalent, they would not be affected. Tr. 8913-14 (Vollmer); Tr. 8914 (Boyer); Applicant's witnesses testified that even if the transmission towers failed, they would buckle and fold over. The effect of their impact on falling would be less than the missiles for which the buried safety-related ducts (e.g., power lines, to spray pond) are designed to resist. Tr. 8923-24 (Vollmer); Tr. 9260 (Romney).

B-83. FOE postulated failure of the walls of the cooling tower basin and subsequent flooding of the turbine building and allowing water to enter the reactor building and control building, preventing a safe shutdown of the plant. FOE, in the alternative, postulated that even if the walls of the cooling tower basin were to remain relatively intact, cooling tower debris falling into the basin would result in increased flooding. Both Staff and Applicant addressed the possible consequences of water loss from the cooling tower basins. Each agreed that the worst-case scenario for a basin-related flooding accident was a breach in the south wall of the basin. Wescott, ff. Tr. 9045, at 2, 3; Boyer *et al.*, ff. Tr. 8213, at 18. A complete breach of the basin wall or a break in other than the south wall would send most of the flood water away from the power block complex and towards the Schuylkill River or Possum Hollow Run. *Id.* Even in the event of a failure of the south wall of either basin, the circulating water pumphouse, which is between the cooling towers and the power block complex, would tend to divert water to the east or west and away from the turbine building. Wescott, ff. Tr. 9045, at 2.

B-84. Both Applicant and Staff assumed a 50-foot breach in the basin wall and in order to maximize the amount of flooding in the turbine building, each also assumed that all of the turbine building main doors on the north side were open. Even with the north wall turbine building doors open, Applicant calculated a water height rise of about 4 feet. Because the walls of the reactor building and central building are water- or steam-tight to above that level, there would be no entrance for water into the category 1 structure and no adverse impact on the ability to safely shut down the reactor. Tr. 9028 (Buchert).

B-85. Staff and Applicant also evaluated the possible effects of erosion by escaping water on buried safety-related structures. Each concluded that no adverse effects would occur. Wescott, ff. Tr. 9045, at 4; Tr. 9324-25, 9335-36 (Wescott); Boyer *et al.*, ff. Tr. 8213, at 19-20; Lefave, ff. Tr. 9047, at 2-3.

10. Integrity of the Spray Pond

B-86. FOE raised questions concerning the integrity of the spray pond — which is the ultimate heat sink for the Limerick decay heat removal from the reactor cores — with respect to missiles that could be generated as a result of blast pressure from an explosion resulting from a pipeline break. The Applicant testified that missiles generated by destruction of the cooling towers could not reach the spray pond. Tr. 8900 (Vollmer). Mr. Vollmer was not aware of any other missiles from an explosion that could reach the spray pond. *Id.* Missiles from an explosion would not be similar to missiles from a tornado. *Id.* Because the design explosion is an air blast, at an elevation of 500 feet above ground, there is going to be a force radiated downward which would not have a tendency to lift missiles up, as in a tornado which rotates them and lifts them. *Id.* at 8900-01 (Vollmer). Various structures that appear in an aerial photograph around the towers would not be exploded by an explosive force from a gas pipeline explosion and carried in the direction of the spray pond. *Id.* at 8901. The photograph showed some temporary structures, including a concrete batch plant that will be removed as well as some old structures that were used for the fabrication of the reactor vessel. Tr. 8901 (Boyer). There is one permanent one-story Butler-type building located somewhere exceeding 800 feet from the spray pond pumphouse building. Since the spray pond pumphouse was designed against tornado missiles, failure of the Butler building would have zero impact on the spray pond building. *Id.* The Applicant estimated that whatever missiles were generated — side panels, disks or whatever — might be moved 50 feet, but not to exceed 100 to 200 feet away from the building. *Id.* at 8908. Mr. Boyer did not think that sheet metal would have any effect on the spray pond fixtures or the pipes leading to the fixtures. *Id.* at 8908-09. We agree.

B-87. The spray nozzles and the piping within the spray pond are safety-related. Tr. 9368 (Lefave). The Applicant is doing a probabilistic risk assessment of the tornado event to determine the probability of how many nozzles and trains in the piping can be affected by tornado missiles. *Id.* Presumably, the results will be evaluated against the required function ability for this system. The Staff considers this to be an open item in its review of externally generated missiles. SER § 3.5.2. It was not conceivable to the Staff, however, that the postulated pipeline accidents could generate missiles which could impact the spray nozzles. This conclusion was based on the belief that the blast wave travels so fast that it would be unable to pick up anything and carry it. Tr. 9368 (Romney). For a detonation of 56 tons of TNT the positive phase pulse time of the blast wave at 1200 feet would be approximately 170 milliseconds. Staff Ex. 21.

B-88. The Staff had not, and did not know whether the Applicant had, conducted an analysis of what potential effects a blast wave would have on the spray pond nozzles. Tr. 9369 (Romney). The Staff did think they are strong enough to take the blast pressure, since they and related piping are designed to withstand the safe shutdown earthquake and because the pressure the blast wave would exert on the piping is not going to be a pressure large enough to affect the structural integrity of the piping system. Any effect would be rather small. Tr. 9371 (Kuo). The calculated pipeline accident blast pressure on the surface of the spray pond water is approximately 1.9 psi. Tr. 9373 (Ferrell).

B-89. The Applicant also testified that if a cooling tower were to fail from a blast from the southwest direction, it would collapse within its own perimeter and would not reach the spray pond pumphouse. Tr. 9284, 9364 (Romney). A cooling tower has never failed as a rigid body. Tr. 9341-42 (Romney).

B-90. We find that all of FOE's allegations and speculations of sequences of events omitted from the Applicant's and Staff's analyses to be without merit. Applicant has demonstrated reasonable assurance that the safety-related structures at Limerick will withstand the postulated pipeline accidents. Accordingly, FOE's Contentions V-3a and V-3b are without merit.

C. LEA I-42: Environmental Qualification of Electric Equipment

C-1. LEA Contention I-42, admitted as respecified, states:

> The Applicant has not shown compliance with the Commission's rule, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, Jan. 21, 1983, 48 Fed. Reg. 2729, 10 C.F.R. § 50.49. Particularly, it has neither established a program for qualifying all of the electrical equipment covered by § 50.49, nor performed an analysis to ensure that the plant can be safely operated pending completion of equipment qualification, as required by § 50.49(i). Failure to comply will threaten the health and safety of the public.

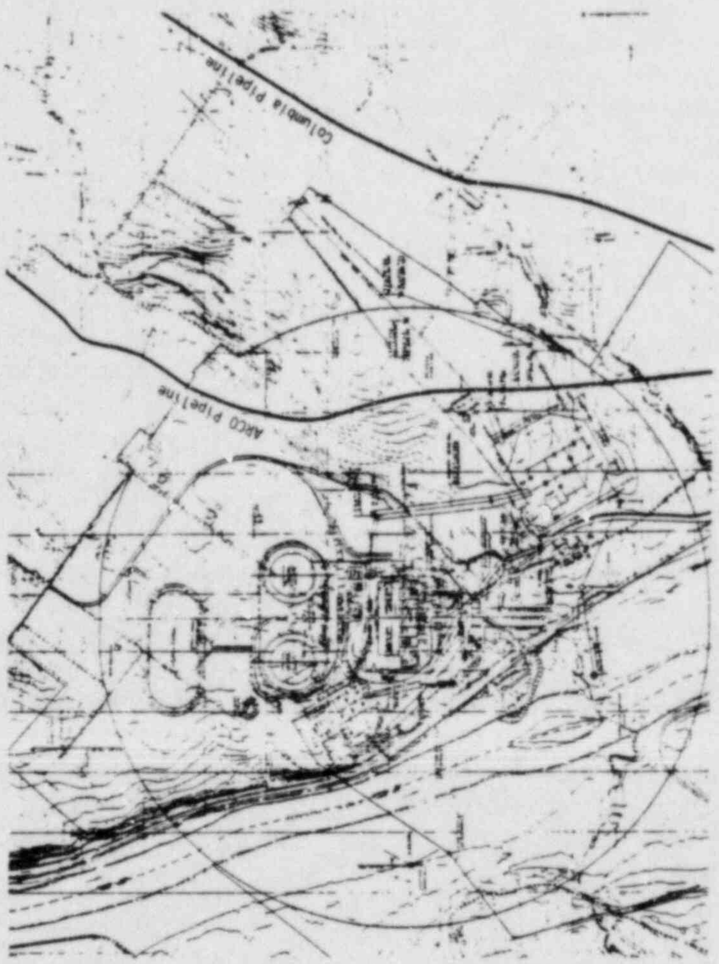

FIGURE 1. ARCO and Columbia gas pipelines
Source: Limerick SER (p. 2-12), Staff Ex. 6

LIMERIC PROJECT
JOB 8031

TABLE I

SUMMARY OF ACCIDENTAL EXPLOSION PRESSURES

LOADING ON STRUCTURE / BUILDING FACILITIES	DESIGN/ASSESSMENT VALUES: POSITIVE PEAK REFLECTED PRESSURE-PSIG — COLUMBIA PIPELINE NATURAL GAS EXPLOSION		ARCO PIPELINE GASOLINE EXPLOSION		READING RAILROAD BOX/TANKCAR EXPLOSION		MARGINS (%) OVER DESIGN/ASSESSMENT PRESSURE FOR EXPLOSION				N-S DIRECTION COMPARISON OF GLOBAL BLDG RESPONSE — EXPLOSION PRESSURES		SAFE SHUTDOWN EARTHQUAKE		REMARKS
	ROOF	EXT WALL	ROOF	EXT WALL	ROOF	EXT WALL	ROOF BENDING	ROOF SHEAR	EXT WALL BENDING	EXT WALL SHEAR	OVER-TURNING MOMENT	STORY SHEAR	OVER-TURNING MOMENT	STORY SHEAR	
REACTOR BLDG. UNIT 1	NC [1]	NC	NC	NC	5.3	16.1	NC		19	15	FT-K 1.37×10^7	K 89,630	FT-K 1.51×10^7	K 110,440	
REACTOR BLDG. UNIT 2	5.4	10.0	1.9	1.9	NC	NC	35	3.3	NC						
DIESEL GEN. BLDG. UNIT 1	NC	NC	NC	NC	5.7	16.4	NC		14	28	1.55×10^5	8330	4.65×10^5	9,060	
DIESEL GEN. BLDG. UNIT 2	6.7	10.0	1.9	1.9	NC	NC	84	20	NC		1.55×10^5	8330	4.65×10^5	9,060	
CONTROL BLDG.	4.9	10.0	<1.9	<1.9	3.3	10.0	83	14	15	20	NA	NA	NA	NA	
SPRAY POND PUMPHOUSE	3.0	5.0	<1.0	<1.0	2.1	4.7	143	9	900	101	1.72×10^4	2,025	1.48×10^5	4,842	

NOTES:

1. NC MEANS NOT COMPUTED. ELEMENT IS LESS CRITICAL THAN IN CORRESPONDING STRUCTURAL UNIT.
2. NA MEANS NOT APPLICABLE. THE ELEMENT OR LOADING CASE DOES NOT EXIST OR APPLY TO THE STRUCTURE UNDER CONSIDERATION.

FIGURE 2.
Source: Boyer *et al.*, Attachment

TABLE II
SUMMARY OF PRESSURES RESULTING FROM
A NATURAL GAS PIPELINE DETONATION

Pressure (Pr) PSI / BLDG.	COLUMN 1 REG. GUIDE 1.91 REV. 1 SURFACE BURST		COLUMN 2 REG. GUIDE 1.91 REV. 1 AIR BURST		COLUMN 3 4 x REG. GUIDE SURFACE BURST		COLUMN 4 4 x REG. GUIDE AIR BURST		COLUMN 5 PRESSURES USED IN STRUCTURAL ASSESSMENT	
	ROOF	EXT. WALL	ROOF	EXT. WALL	ROOF	EXT. WALL	ROOF	EXT. WALL	ROOF	EXT. WALL
DIESEL GEN.	1.9	5.8	3.5	8.3	4.0	13.0	2.5	16.0	6.7	16.4
REACTOR BLDG.	1.2	5.8	2.8	8.3	2.6	13.0	5.2	16.0	5.4	16.1
CONTROL STRUCTURE	1.6	5.0	2.8	6.9	3.3	11.0	4.7	14.0	4.9	10.0
SPRAY POND PUMP HOUSE	0.8	2.5	1.2	3.3	1.8	5.0	1.4	6.0	3.0	5.0

FIGURE 3.
Source: Boyer *et al.*, ff. Tr. 8213, Attachment

IN FORMAT OF APPLICANT'S TABLE I

[illegible]

TABLE I

SUMMARY OF ACCIDENTAL EXPLOSION PRESSURES

LOADING ON STRUCTURE / BUILDING FACILITIES	DESIGN/ASSESSMENT VALUES — POSITIVE PEAK REFLECTED PRESSURE-PSIG					
	COLUMBIA PIPELINE NATURAL GAS EXPLOSION		ARCO PIPELINE GASOLINE EXPLOSION		READING RAILROAD BOX/TANKCAR EXPLOSION	
	ROOF	EXT. WALL	ROOF	EXT. WALL	ROOF	EXT. WALL
REACTOR BLDG. UNIT 1	NC(1)	NC	NC	NC	5	12.2
REACTOR BLDG. UNIT 2	3.2	7.3	<1	1	NC	NC
DIESEL GEN. BLDG. UNIT 1	NC	NC	NC	NC	5.2	12.5
DIESEL GEN. BLDG. UNIT 2	3.6	7.4	<1	<1	NC	NC
CONTROL BLDG.	3.6	7.2	<1	<1	3.6	7.2
SPRAY POND PUMPHOUSE	1.9	3.7	<1	<1	2	4.5

NOTES:

1. NC MEANS NOT COMPUTED. ELEMENT IS LESS CRITICAL THAN IN CORRESPONDING STRUCTURAL UNIT.
2. NA MEANS NOT APPLICABLE. THE ELEMENT OR LOADING CASE DOES NOT EXIST OR APPLY TO THE STRUCTURE UNDER CONSIDERATION.

NRC CALCULATIONS

FIGURE 4.
Source: ff. Tr. 9055, Staff Ex. 23

1. Summary

C-2. Testimony by the Applicant and the Staff supports the conclusion that the Applicant has an acceptable program, although not completely implemented, for qualification of electric equipment important to safety at Limerick, which is in compliance with 10 C.F.R. § 50.49, as adopted in January 1983. This testimony described how items to be qualified were identified and how the program was developed and implemented. Proper identificatic . was assured by an independent verification program conducted by a qualified contractor. The Staff's review, while also not complete, verified the adequacy of the program.

C-3. Based on qualification efforts so far, it is not anticipated that completion of the program would identify any components not properly qualified. Should this occur, however, the Applicant would then have to perform and have approved by the Staff an analysis, as required by § 50.49(i) to ensure that the plant can be safely operated pending completion of equipment qualification. Such an analysis is called a Justification for Interim Operation (JIO) by the Staff. Subject to that possibility, we find that the Applicant has met its burden of proof on this contention by demonstrating, (1) that it has a proper program in place for qualifying all of the electrical equipment covered by § 50.49; and (2) that those particular components of concern to LEA, as set forth in the bases for the contention, have been properly considered by the Applicant.

C-4. The Applicant and the Staff provided expert witnesses and testimony; LEA and the City of Philadelphia cross-examined these witnesses, but did not provide their own witnesses. Evidentiary hearings were held on April 9 and 10, 1984, in Philadelphia, Pennsylvania.

2. Compliance with the January 1983 Environmental Qualification Rule

C-5. As a framework for discussing the merits of this contention, we begin by considering the state of compliance of the Applicant with the subsections of 10 C.F.R. § 50.49, adopted in January 1983, as applicable to the contention.

C-6. Section 50.49(a) states each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section. Section 50.49(b) states that electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment[3]: This equipment is that relied upon to remain functional during and following design basis events to ensure

(i) the integrity of the reactor coolant pressure boundary,

(ii) the capability to shut the reactor down and maintain it in a safe shutdown condition, and

(iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 C.F.R. Part 100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

(2) Nonsafety-related electric equipment whose failure could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment. (Footnote omitted)

[3] Safety-related electric equipment is referred to as "Class IE" equipment in IEEE (standard) 323-1974.

C-7. LEA asserts, in part (a) of its Basis for the contention, that Applicant's environmental qualification (EQ) program, designed prior to issuance of the new rule, was designed to qualify safety-related equipment only (and therefore does not include nonsafety-related equipment whose failure under postulated environmental conditions could mislead the operator or otherwise prevent satisfactory accomplishment of specified safety functions, and certain post-accident monitoring equipment). Applicant argues that even though its program for EQ was designed before the promulgation of the new rule, because of its anticipation of the new requirements and because of its conservative equipment classification practice, its program does comply with the new rule. Boyer *et al.*, ff. Tr. 9529, at 1-2. Further, Applicant avers that all Limerick equipment within the scope of 10 C.F.R. § 50.49 will be qualified by the fuel load date. *Id.* at 4.

C-8. LEA, also in part (a) of its Basis, asserts that the Applicant should promptly develop a list of the equipment at Limerick, subject to § 50.49(b)(2), that is "important to safety" (and not just safety-related) and that will be tested in its EQ program as required by § 50.49(d). Examples given by LEA of systems or equipment that should be reviewed for inclusion in the Applicant's EQ program were the feedwater control, emergency lighting and communications systems, the plant process computer system, and computer software.

C-9. The Limerick Project "Q-List" was developed and established as the controlling document identifying the safety-related structures, systems and components [including electric equipment] to meet the requirements of § 50.49(b)(1). *Id.* at 4-5.

C-10. The Applicant testified that there is no equipment at Limerick in the subset § 50.49(b)(2). *Id.* at 3, 7. The interfaces between safety-related electrical components are evaluated as part of the plant design process. Whenever cases are identified in which failure of nonsafety-related components could prevent attainment of the safety function objectives, they are eliminated by implementing design modifications or by adding (such components) to the Project Q-List and qualifying them as necessary. The Electrical Equipment Separation Program is an example of such an interface evaluation. *Id.* at 7. All electrical equipment on the Q-List is reviewed to determine its environmental qualification requirements. If the electrical equipment is determined to be located in a harsh environment, the appropriate environmental qualification parameters for the component are identified. *Id.* at 8.

C-11. "Certain post-accident monitoring equipment" is defined by the footnote to § 50.49(b)(3), which references Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." This Guide defines three categories of design and qualification criteria. Category 1 criteria are similar to the criteria applicable to safety-related systems. Category 2 criteria include selected criteria normally associated with safety-related systems, but the same environmental requirements as Category 1. Category 3 criteria specify only a high-quality commercial-grade installation, for which there are no environmental qualification requirements. *Id.* at 5-6.

a. Independent Component Classification Program

C-12. To assure the identification, in the Limerick Environmental Qualification Program, of all electrical equipment required to perform a safety function, the Applicant contracted with Quadrex Corporation to perform an independent verification, the Component Classification Program. Boyer *et al.*, ff. Tr. 9526, at 9. Quadrex had conducted five identical independent review analyses of the overall environmental qualification programs at other nuclear power plants prior to the Limerick program. Tr. 9551 (Stanley). The extensive effort at Limerick showed that of the approximately 30,000 components considered, of which approximately 1600 were different (i.e., nonidentical) electrical items, 16 differences in electrical equipment classification from the original Applicant architect-engineer classifications were identified. Nine of the sixteen components were found to be located in a mild environment. Four of the sixteen were to be reclassified as not requiring environmental qualification. The remaining three are included in the EQ Program. Boyer *et al.*, ff. Tr. 9526, at 22-23; Tr. 9622-23 (Boyer).

C-13. A comparison of the Component Classification Program (CCP) rules against § 50.49 was performed and it was determined that the classification rules fully complied with the requirements of § 50.49, even though they were prepared and implemented prior to publication of the new rule. This determination was also based on a comparison of the CCP rules with draft Regulatory Guide 1.89, Rev. 1, "Qualification of Class 1E Equipment for Nuclear Power Plants." Boyer *et al.*, ff. Tr. 9526, at 23.

3. *Systems Excluded from the EQ Program*

C-14. As a part of the basis for its Contention I-42, LEA asserted that the emergency lighting system, in-plant communications system, plant process computer system and computer software were examples of systems that were improperly excluded from PECo's qualification program. The evidence indicated that the exclusions were proper in that the systems cited by LEA are not important to safety as the term is used in 10 C.F.R. § 50.49; that is, they are not relied on during a design basis accident in areas subject to a potentially harsh environment and their failure would not prevent achievement of safety function objectives. Boyer *et al.*, ff. Tr. 9529, at 11-15; Masciantonio, ff. Tr. 9640, at 7-8.

a. *Emergency Lighting System*

C-15. The Applicant testified that this system was not included in the CCP because it is not safety-related as defined by § 50.49, it is not relied upon to provide lighting during a design basis accident in areas which could produce a harsh environment, and its failure could not prevent achievement of the safety function objectives defined in subparagraphs (i) through (iii) of § 50.49(b)(1). Boyer *et al.*, ff. Tr 9526, at 12. The Staff concurs. Masciantonio, ff. Tr. 9640, at 7.

b. *In-Plant Communications Systems*

C-16. The Applicant testified that these systems were not included in the CCP because they are not safety-related, they are not relied upon during a design basis accident in areas that could produce a harsh environment, and their failure could not prevent the achievement of the safety function objectives defined in subparagraphs (i) through (iii) of

§ 50.49(b)(1). Boyer *et al.*, ff. Tr. 9526, at 13. The Staff concurs. Mascian'onio, ff. Tr. 9640, at 7.

c. *The Plant Process Computer System*

C-17. The Applicant testified that this system and the computer software were not reviewed because the computer is not safety-related; it is not relied upon to provide information during a design basis accident in areas that could produce a harsh environment, and its failure could not prevent achievement of the objectives defined in subparagraphs (i) through (iii) of § 50.49(b)(1). The computer software has not been reviewed because it is outside the scope of § 50.49. Information obtained via the plant process computer is not required during or following these accidents. The computer system interfaces with other systems that are safety-related, but these electrical interfaces are designed in compliance with Regulatory Guide 1.75, "Physical Independence of Electric Systems." Boyer *et al.*, ff. Tr. 9526, at 14. The Staff concurs. Masciantonio, ff. Tr. 9640, at 7.

d. *Feedwater Control System*

C-18. The Applicant testified that this system was included in the CCP. The review showed, however, that it contains no equipment having a safety function as defined by § 50.49. Boyer *et al.*, ff. Tr. 9526, at 14-15. The Staff concurs. Masciantonio, ff. Tr. 9640, at 7.

e. *Standby Liquid Control System*

C-19. The Applicant testified that the squib values, in this system, have been added to the EQ List of Equipment Important to Safety. Boyer *et al.*, ff. Tr. 9526, at 3. The Staff concurs. Masciantonio, ff. Tr. 9640, at 10.

C-20. The keylock switch is located in the control room which is maintained by a safety-related ventilation system and therefore is not subject to harsh environments. Boyer *et al.*, ff. Tr. 9526, at 2[illegible]. The Staff concurs. Masciantonio, ff. Tr. 9640, at 10.

f. *Human Interaction Problems*

C-21. In part (b) of its Basis for its contention, I[illegible]A contends that failure of nonsafety-related valves, but which are important to safety, could mislead an operator into miscategorization of an accident for emergency planning purposes. Since there is no electrical equipment in the class defined by § 50.49(b)(2), this could not happen for such equipment. With respect to the post-accident monitoring equipment defined by § 50.49(b)(3), the operators will be directed by written procedures to rely only on the equipment that is qualified in accordance with Regulatory Guide 1.97, Rev. 2, if the equipment is subjected to a harsh environment, and thus will not be misled by unqualified equipment. Boyer *et al.*, ff. Tr. 9526, at 3, 25-32.

C-22. The Limerick-specific Transient Response Implementation Plan (TRIP) procedures are initiated and keyed to entry condition symptoms to treat these symptoms and are specific to Limerick. The procedures are organized in such a manner as to control those plant parameters important for protecting the plant safety barriers against the release of radioactive material to the environment. Whenever a symptom develops, the operator immediately enters the applicable procedure and takes the corrective action directed by the procedures, until its exit conditions are satisfied. If the particular transient continues to degrade, the operator enters contingency procedures to handle the more degraded conditions until he can return to the main procedures. Boyer *et al.*, ff. Tr. 9529, at 25-27.

C-23. Review of the listing of Regulatory Guide 1.97 instrumentation reveals that all entries into the TRIP procedures are monitored by environmentally qualified instrumentation. The impact on execution of TRIP procedures is minimal since the qualified instrumentation that must be used is either the instrumentation which the operator would normally choose to use under those conditions or the only qualified instrumentation available to monitor the parameter. The operator is specifically instructed in the TRIP procedures to utilize only certain instrumentation in the event of an indication of adverse environmental conditions. In accordance with the requirements of Regulatory Guide 1.97, the applicable instrumentation will be highlighted by special markings on the control panel to aid in its identification and assure that only such instruments will be used under the circumstance of adverse environmental conditions. Boyer *et al.*, ff. Tr. 9529, at 28-30; *see* Tr. 9601-10 (Doering).

C-24. Many TRIP procedures use only environmentally qualified instrumentation. However, that instrumentation may cover a broader range than nonqualified equipment and may, therefore, be less precise. The instrumentation an operator normally relies on is generally restricted to a narrow band around the operating range and is, therefore, more exact. Absent an indication of actual adverse environmental conditions in the reactor building, the operator is not restricted to the use of environmentally qualified instrumentation. Tr. 9607-09 (Doering); Masciantonio, ff. Tr. 9640, at 8.

C-25. A "human interaction review," *per se*, is not a requirement of § 50.49. *Id.* at 8.

4. Aging of Equipment

C-26. In part (c) of its Basis, LEA contends that where the qualified life of a piece of equipment does not equal the 40-year plant life, no action is identified to correct the deficiency. The environmental qualification of electrical (and other) equipment is contingent upon replacing such equipment at the end of its designated life and upon performing required maintenance during its designated life. The Limerick Plant Staff Maintenance Group has a systematic program to determine required replacement intervals for the equipment whose designated life is less than 40 years and to define the maintenance and frequency thereof for equipment whose environmental qualification is required to be sustained. Boyer *et al.*, ff. Tr. 9526, at 32-35; Masciantonio, ff. Tr. 9640, at 9.

5. Completeness of EQ Program

C-27. At the time of hearing, the Applicant's EQ Program was 95% complete. Final completion was anticipated to occur in June 1984. For the remaining 5%, the work on the qualification packages was sufficiently along the way that an informed judgment was that there would be no unqualified equipment for which a Justification for Interim Operation would be requested. Tr. 9617 (Boyer).

6. Staff Review of the Limerick EQ Program

C-28. The Limerick EQ program is reviewed by the Staff for completeness, accuracy and conformance — to determine proper definition of the scope of the program, proper definition of postulated environments, and demonstration of qualification in accordance with NRC rules and regulations, which include 10 C.F.R. § 50.49, Regulatory Guide 1.89 ("Qualification of Class 1E Equipment for Nuclear Power Plants"), NUREG-0588 ("Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment") and Institute of Electrical and Electronics Engineers (IEEE) standards. Masciantonio, ff. Tr. 9640, at 4. In addition, the Staff reviewed the total number of components and equipment types in the Limerick EQ program as compared to other plants of similar design to assure consistency, and reviewed the process used for selecting components, as described in the EQ report. *Id.* at 6. Conformance to § 50.49(b)(2) concerning nonsafety-related equipment whose failure under postulated accident conditions could prevent the satisfactory accomplishment of safety functions is determined by the Staff's review of Limerick with respect to the issues in IE Information Notice 79-22 (Qualification of Control Systems) and conformance with Regulatory Guide 1.75 ("Physical Independence of Electric Systems"). *Id.* at 6. Tr. 9665-66, 9678-79 (Masciantonio). *See also* Tr. 9683-88 (LaGrange). The Staff review of conformance of Limerick to Regulatory Guide 1.75 is complete and Limerick has been found acceptable. Masciantonio, ff. Tr. 9640, at 7. Tr. 9709 (LaGrange, Masciantonio). Review of the Applicant's response to Information Notice 79-22 (Qualification of Control Systems) was not yet complete. Masciantonio, ff. Tr. 9640, at 7. The Staff testified that similar reviews, which analyze the effects of high-energy line breaks on the interactions between nonsafety-related and safety-related components, had been completed for several plants and it had no reason to believe it would be a special problem for Limerick. Tr. 9710 (LaGrange). In addition, the Staff had not completed its review of the pressure-temperature profile following a loss-of-coolant accident submitted by the Applicant. This "profile" is substantially lower than for typical boiling water reactors that have been reviewed and therefore needs special Staff review. Tr. 9711-12 (Masciantonio). The equipment has been environmentally qualified against the Applicant's proposed profile. Tr. 9712 (LaGrange).

C-29. An audit of the Applicant's Equipment Qualification files, including a plant walkdown, was conducted by the Staff, primarily to verify the bases of the information submitted. Twelve EQ files, representing approximately 10% of the equipment items in the EQ program, were selected for detailed review. In all cases it was determined that adequate proof of qualification was provided to establish qualification as claimed. Masciantonio, ff. Tr. 9640, at 11.

C-30. The Staff has determined that the Applicant has established a program for qualifying electric equipment important to safety within the scope of § 50.49, but its review is not complete and no approval of the program has been issued. Its review was expected to be complete within a few months (from April 1984). *Id.* at 11. Should there be any unqualified equipment, Applicant will be required, according to § 50.49(i), to perform an analysis to ensure that the plant can be safely operated pending completion of environmental qualification. This analysis (Justification for Interim Operation) must be submitted and approved by the Staff before the Staff would support issuance of a license. *Id.* at 12.

7. Discussion

C-31. LEA would have the Board find in its favor that there is no basis in the present record for a finding that Limerick is in compliance with 10 C.F.R. § 50.49. Further, it would have us retain jurisdiction until several actions by the Applicant and Staff are taken as preconditions for a finding of such compliance. LEA's proposed findings (June 21, 1984), at 13. Applicant and Staff would have us find, on the basis of the present record, that the Applicant has fully complied with the requirements of § 50.49. App. PF (June 8, 1984), at 26; Staff PF (July 2, 1984), at 19.

C-32. All parties agree that Applicant's EQ program has not been completely implemented and Staff's review is not complete. Prior to the time of hearing, Staff had received a report from the Applicant indicating approximately 80% of the equipment items as being qualified. (As noted in Finding C-27 above, at hearing the Applicant stated that its program was 95% complete, although all of this had not been officially reported to the Staff.) The Staff Safety Evaluation Report (SER) will not be closed out until full compliance with § 50.49 has been demonstrated. Tr. 9698 (Masciantonio). The Staff must conclude that compliance with the requirements of § 50.49 has been demonstrated before an operating license is issued. Masciantonio, ff. Tr. 9640, at 14.

C-33. When governing statutes or regulations require a licensing board to make particular findings before granting an applicant's requests, a board may not delegate its obligations to the Staff. The responsibilities of the boards are independent of those of the Staff under the Commission's system, and the boards' duties cannot be fulfilled by the Staff, however conscientious its work may be.[8]

C-34. Applicant argues that the prerequisite to the issuance of a decision in a case such as this where the Staff's review is not yet complete, is a basis in the present record on which to reach an informed conclusion, citing *Cincinnati Gas & Electric Co.* (Wm. H. Zimmer Nuclear Power Station, Unit 1), LBP-82-68, 16 NRC 741, 748 (1982). In that case, however, the Board found that "[w]e have no basis in the present record on which to reach an informed conclusion with regard to the

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[8] *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-298, 2 NRC 730, 737 (1975). *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-124, 6 AEC 358, 360, 361-62 & n.4 (1973).

FEMA (emergency planning) review. Consequently, we require that the results of the FEMA review be served on the Board and parties . . . ." The Applicant also claims there is specific precedent for the action it seeks — post-hearing resolution of this matter by the Staff — in the *Shoreham* proceeding. In that proceeding, the Atomic Safety and Licensing Board (two of whose members also serve on the instant Board) found that in the area of environmental qualification the deficiencies were minor and would be resolved by the Staff subsequent to the Board's order, but prior to issuance of a license. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), LBP-83-57, 18 NRC 445, 544 (1983). Consequently, the Board concluded that the environmental qualification program and the intended further revisions to implement § 50.49(b)(2) were acceptable.

C-35. On the basis of the evidence before us we can and do conclude that the Applicant has established, in the words of the contention, an acceptable *program* for qualifying all of the electrical equipment covered by § 50.49. Classification of components by the Applicant, verified by an independent contractor and audited by the Staff, with no evidence of any component currently improperly qualified, gives us a basis to reach an informed conclusion with respect to the adequacy of the *program* for compliance with § 50.49.

C-36. Implementation of the EQ program admittedly is incomplete. It is a close question, in our view, whether we can conclude, based on the present record, that the remainder of the implementation, including Staff review, constitute minor procedural difficulties (*see Consolidated Edison Co. of New York* (Indian Point Station, Unit No. 2), CLI-74-23, 7 AEC 947, 951 (1974)), or minor documentation deficiencies (*see Shoreham, supra*).

C-37. The Appeal Board, relatively recently, had occasion to deal specifically with the question of reliance on predictive findings and post-hearing verification, albeit in the context of contentions with respect to emergency planning. *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076, 1103 (1983). First, the Board said:

> We are in agreement with the basic principles upon which Joint Intervenors rely. The Commission, in fact, has long held that, "[a]s a general proposition, issues should be dealt with in the hearings and not left over for later (and possibly more informal) resolution." *Consolidated Edison Co. of New York* (Indian Point Station, Unit No. 2), CLI-74-23, 7 AEC 947, 951 (1974). "[T]he 'post-hearing' approach should be employed sparingly and only in clear cases" — for example, where "minor procedural deficiencies" are involved. *Id.* at 952, 951, n.8. *Accord, Marble Hill, supra*, 7 NRC at 318; *Cleveland Electric Illuminating Co.* (Perry Nuclear Power

Plant, Units 1 and 2), ALAB-298, 2 NRC 730, 736-37 (1975); *Washington Public Power Supply System* (Hanford No. 2 Nuclear Power Plant), ALAB-113, 6 AEC 251, 252 (1973).

C-38. Second, the Board noted that the Commission takes a slightly different course with respect to emergency planning:

> At one time, the [Commission's] regulations required a finding that "the *state* of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." 10 C.F.R. § 50.47(a)(1) (1982) (emphasis added). In July 1982, the Commission amended this provision by clarifying that "the findings on emergency planning required prior to license issuance are predictive in nature" and by eliminating the reference to the "state" of emergency preparedness.

C-39. In the *Waterford* case the Appeal Board did allow predictive findings in five areas of emergency planning, but made no such concession on other issues.

C-40. The record may be summarized as follows. The evidence shows that the Applicant has established a program for qualifying all of the electrical equipment covered by § 50.49. No equipment specified by LEA in the bases for its contention has been shown to be misqualified. The program has been audited by the Staff and found acceptable. With respect to the 5% of the EQ program yet to be completed, there is reasonable assurance that it will be completed in compliance with § 50.49, based on the adequacy of the program itself and the Staff commitment to conclude its review of the entire program prior to issuance of a license. Further, the work on the remaining 5% was sufficiently far along that an informed judgment by the Applicant was that there would be no unqualified equipment for which a Justification for Interim Operation would be requested (thus obviating the need for any analysis required by § 50.49(i)).

C-41. With respect to completion of the Staff review of the Applicant's response to questions related to IE Information Notice 79-22, there is reasonable assurance that this will be completed to the Staff's satisfaction. Similarly, there is reasonable assurance that the Staff review of the temperature and pressure behavior following a loss-of-coolant accident will be completed to the Staff's satisfaction. LEA raised no particular concern with either of these Staff reviews, other than the general complaint of incompleteness. If the results of the Staff review of Applicant's response to IE Information Notice 79-22 show a high-energy line break interaction which was not designed for, then additional components may have to be included in the environmental qualification program (in the absence of design changes to correct any such interaction).

This still does not detract from our finding that the allegation in the contention, of the lack of a proper environmental qualification program, is without merit. Similarly, if the results of the Staff review of the temperature and pressure profile following an accident show that those parameters would be higher than assumed for the EQ program, then the environmental qualification of the affected components will have to be reanalyzed by the Applicant, following the same approved program, but against different postulated temperature and pressure conditions.

C-42. We find that we cannot strictly characterize the incomplete aspects of the Applicant's implementation of its EQ program and the Staff's review thereof as minor procedural or documentational deficiencies. Within the scope of the contention as worded, however, we can and do find that this is a clear case where reasonable assurance exists that the Applicant will comply with § 50.49 before any license will be issued. In other words, no specific complaint of LEA (including particular components alleged by LEA to be improperly qualified) remains to be explored in the Staff's overall review of electric equipment qualification at Limerick, which review is broader than the litigated issues. This situation could change only if, contrary to the record before us, the Applicant decides to seek a Justification for Interim Operation under § 50.49(i). In such an eventuality, the parties obviously are obligated to bring such change in the record promptly to the attention of the parties and any adjudicatory body with jurisdiction. Subject to this possibility, we find this contention without merit and do not retain jurisdiction.

D. Confirmation of Findings of Fact Made on the Record That AWPP Contention VI-1 (QA/QC of Welding) Lacks Merit

1. The Contention Lacks Merit as Previously Determined in the Bench Decision

D-1. AWPP Contention VI-1, as admitted by the Board, states:

> Applicant has failed to control performance of welding and inspection thereof in accordance with quality control and quality assurance procedures and requirements, and has failed to take proper and effective corrective and preventive actions when improper welding has been discovered.

D-2. This contention was admitted as an issue in controversy on reconsideration by the Board (after earlier conditional admission and then rejection given the issue specified by AWPP). The reconsidered admission was subject to the important requirement that, after discovery, AWPP specify in advance of the hearing the particular instances of alleged improper actions of Applicant with regard to quality control and quality assurance of welding at Limerick, which AWPP would rely upon to litigate its contention.[9] This particularization of the contention was accomplished in the course of prehearing filings by the parties and rulings by the Board.[10]

D-3. This contention was litigated on May 7-10, 1984. Expert and factual testimony was presented by separate witness panels for the Applicant and NRC Staff. The proposed direct testimony offered by AWPP's representative, Mr. Frank R. Romano, was not admitted into evidence for the reasons set forth in the Board's May 2, 1984 "Memorandum and Order on Pretrial Motions Regarding Testimony on Contention VI-1" (unpublished), which granted the motions by the Applicant and Staff to strike Mr. Romano's testimony. In addition, at the hearing the Board rejected the late-filed testimony of Professor Gudmund R. Iverson proffered by AWPP (AWPP Ex. 3 for identification), because it was inexcusably late (it had been filed at the hearing), did not relate to any of AWPP's specified instances, and in any event was not sufficiently probative towards any matter relating to quality assurance of welding to be admitted as late testimony. Tr. 10,428-35, 11,931 (Brenner, J.)

D-4. The evidentiary hearing on this contention involved extensive written testimony by the Applicant which detailed the facts involved in each instance relied on by AWPP for its allegation of improper welding and quality assurance thereof. Boyer *et al.*, ff. Tr. 10,321. The NRC Staff's testimony fully supported the Applicant's. Durr and Reynolds, ff. Tr. 10,977. The extensive oral testimony, including cross-examination by AWPP and Board questions, also fully supported and confirmed the accuracy and completeness of the written direct testimony.

D-5. Accordingly, at the conclusion of the hearing on the contention, the Board announced that at that time it was its provisional judgment that, based on the entire record, there are no facts upon which it could be concluded that the Applicant had not overwhelmingly met its burden of proof on the contention. We noted our view that the facts were straightforward, fully stated in the Applicant's direct testimony and

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[9] *See* "First Special Prehearing Conference Order," LBP-82-43A, 15 NRC 1423, 1517-18 (1982); "Memorandum and Order (Concerning Objections to June 1, 1982 Special Prehearing Conference Order)," slip op. at 6 (July 14, 1982) (unpublished); "Second Special Prehearing Conference Order," LBP-83-39, 18 NRC 67, 88-91 (1983); "Memorandum and Order Confirming Rulings Made at Prehearing Conference," slip op. at 5-7 (October 28, 1983) (unpublished).

[10] AWPP filed its list of specified allegations of improper welding and related quality assurance actions on March 6, 1984. Thereafter, the Board ruled on the Applicant's and Staff's objections to some of the alleged instances as being beyond the scope of welding-related matters. "Memorandum and Order Ruling on Applicant's Motion to Strike Specific Instances Advanced by AWPP in Support of Contention VI-1" (April 2, 1984) (unpublished).

not contradicted in any way under cross-examination or Board questions. Tr. 11,047 (Brenner, J.). *See also* Tr. 11,050-54 (Brenner, J.). We also noted our provisional view that the witnesses were straightforward, truthful and candid and that they had fully disclosed the bases for the facts and conclusions in their written testimony. Tr. 11,048 (Brenner, J.)

D-6. Given our provisional view, we held it was unnecessary for the Applicant to follow the normal course and file its proposed findings of fact first. It was not necessary to have all the facts and conclusions in the record regurgitated in lengthy findings, which the Applicant, as the party with the burden of proof, would have had to file if the Board had not revealed and announced its provisional decision on the merits. Tr. 11,048-49 (Brenner, J.) However, the Board refrained from making final its provisional ruling — that the conclusions in the testimony of the Applicant and Staff were correct and fully supported and that therefore the contention lacked merit — in order to give AWPP the opportunity to file proposed findings of fact and conclusions of law. The Board informed AWPP that it should point out in its proposed findings evidence in the record which it believed showed that there was merit in any of its instances alleged in support of its contention. The Applicant and Staff would then have an opportunity to file reply findings discussing the matters covered in AWPP's proposed findings. Tr. 11,049-50 (Brenner, J.) *See also* Tr. 11,052, 11,055-58 (Brenner, J.).

D-7. As scheduled, AWPP filed its proposed findings on May 22, 1984, and the Applicant and Staff filed their separate replies on May 29. On the record of May 31, 1984, the Board heard oral argument and set forth its reasons as to why none of the matters raised in AWPP's proposed findings raised any item which contradicted the Applicant's and Staff's evidence as had been previously ruled upon by us. *See* Tr. 11,915-94. We found the reply findings of the Applicant and Staff to accurately and fully reflect the record. We found that AWPP's proposed findings were inaccurate on several points. Tr. 11,935-36 (Brenner, J.). Therefore, there was no item meriting further deliberation by the Board and we entered our ruling that AWPP's contention lacked merit. As we stated we would, that bench ruling hereby is confirmed and becomes the partial initial decision that AWPP Contention VI-1 lacks merit. Tr. 11,964, 11,993-94 (Brenner, J.).

D-8. Before setting forth the Board's conclusions, which are based on those of the Applicant's and Staff's testimony which we find to be correct, we summarize the points raised in AWPP's proposed findings with which the Board disagreed for the reasons stated in our May 31 bench ruling: AWPP continuously ignored the testimony showing there is reasonable assurance that 100% of all safety-related welds were inspected. The sampling procedures, which we also find to be acceptable, were for audits of the inspection program. *See* Tr. 11,923-35, 11,945, 11,984-85. AWPP was totally incorrect in its belief that Applicant's witnesses did not fully answer its questions. We find the witnesses to be qualified, truthful and accurate and worthy of belief. *See* Tr. 11,940-46, 11,953-58. We also set forth why an instance in a Staff inspection report regarding the apparent lack of certified qualification for a receipt of materials inspector could not be related to any alleged welding problems. Tr. 11,946-48. We also set forth why an old matter involving the calibration of weld oven thermometers, raised for the first time in AWPP's findings, was beyond the scope of the contention because it could have been, but was not, set forth as one of AWPP's specified instances in support of the contention. *See* Tr. 11,948-51.

D-9. The Board, on its own, also noted the potential concern it had harbored before the evidentiary hearing regarding the Applicant's remedial actions on the scope of its search of all types of QA records, given the fact that its initial search of QA weld records had been incomplete. Indeed, it was this incomplete search by Applicant, which incompleteness was discovered and corrected by Applicant because of this proceeding and the pending AWPP contention, which led the Board to admit AWPP's welding contention after reconsideration. *See* Tr. 10,708-10 (Boyer). We were satisfied that the scope of Applicant's remedial and preventive actions were appropriate. *See* Tr. 11,958-62, 11,989-91. We also stated why the facts on welds of hangers, and the deficiencies found, did not undercut the conclusion that the contention lacked merit. Tr. 11,985-88.

D-10. The Board finds, as applied to the instances of improper welding activities advanced by AWPP to form the scope of its contention, as follows:

D-11. The Limerick Quality Assurance (QA) program meets the requirements of 10 C.F.R. Part 50, Appendix B, and is effective in assuring that the welding meets the quality requirements and satisfies the design criteria required for the safe operation of the plant. Throughout the course of construction of Limerick, the Applicant has monitored, through audits, all welding-related activities. These audits have confirmed that the QA program has been properly and effectively implemented. Boyer *et al.*, ff. Tr. 10,321, at 3 and 89-90. *See also* Durr and Reynolds, ff. Tr. 10,977, at 23.

D-12. Since there are in excess of 2 million safety-related welds at Limerick, there is the potential for occasional welding deficiencies as have occurred at Limerick. Most of these have been discovered and corrected as the result of the effective implementation of Applicant's QA program. Although the NRC Staff has also identified a few such welding deficiencies, the deficiencies have not formed any pattern of repeated similar instances. Boyer *et al.*, ff. Tr. 10,321, *passim* and particularly at 89. Durr and Reynolds, ff. Tr. 10,977, *passim* and particularly at 11, 13, 15, 17, 18 and 23.

D-13. The circumstances relating to two structural weld deficiencies, emphasized by AWPP, which were not discovered by the Applicant's Quality Control inspector, as well as all the other instances cited by AWPP, and the Applicant's evaluations and corrective and remedial actions as audited by the NRC Staff, have been fully and truthfully described in the Applicant's and Staff's testimony. The testimony clearly establishes that AWPP's instances, all of which were taken from NRC Staff inspection reports and/or Applicant's own audit reports and responses to the NRC Staff, are isolated, nonprogrammatic, and, particularly given their source, in general, indicative of the effectiveness of the Limerick QA program. There has been no "breakdown" of the Limerick QA program for welding. Boyer *et al.*, ff. Tr. 10,321, *passim* and particularly at 4. Durr and Reynolds, ff. Tr. 10,977, *passim* and particularly at 11, 13, 15, 17, 18 and 23.

D-14. Additional expert views finding that the Applicant's welding quality assurance program was effective were provided by the NRC Staff's 1983 programmatic evaluation (1983 "SALP Report"). It states:

> Observations by the Resident Inspector and Construction Inspection Team indicated that a strong construction QC program was in place. In addition to the E-C's well staffed and trained QC organization, the Licensee's QA organization also is staffed by well trained and knowledgeable QA engineers. The Resident Inspectors have noticed that the Licensee's QA engineers have performed more than the required inspections and surveillances in this area.

App. Ex. 52, at 12-13; Boyer *et al.*, ff. Tr. 10,321, at 90.

2. *AWPP's Post-Hearing Motions*

D-15. Subsequent to the close of the record (as well as after the filing of its proposed findings and our May 31, 1984 bench decision on the merits), AWPP filed a motion to reopen the record on this contention (June 8, 1984), followed by its "Motion to Withhold Final Decision Re AWPP Contention VI-1" (June 11, 1984). We agree with the answers of the Applicant and Staff that there is no basis in support of these motions and accordingly deny them.

D-16. The subject of AWPP's motion to reopen is a finding in an NRC Staff inspection report regarding deficiencies in the placement of pipe support hangers resulting from interferences with other structures. Although AWPP cites a May 21, 1984 letter to the Applicant from the NRC Staff, this letter is simply a followup acknowledging Applicant's responses to the underlying Staff inspection report findings and notice of violation issued on January 10, 1984. This is an old matter, arising from combined NRC Staff IE Report 50-352/83-19 & 50-353/83-07, which AWPP previously had included in its list of instances specified in support of this contention, designated by AWPP as the second of its two items "AWPP 260A." In our unpublished "Memorandum and Order Ruling on Applicant's Motion to Strike Specific Instances Advanced by AWPP in Support of Contention VI-1" (April 2, 1984), slip op. at 4-5, we ruled that the hanger interferences violation was not related to welding quality or welding-related quality assurance and that therefore this alleged instance would be stricken as being irrelevant to the contention. AWPP now simply again brings this instance to our attention, and mentions test welding in the same pleading. No reason to reconsider our prior ruling is shown or apparent, even if we consider AWPP's very untimely attempt to seek, in effect, reconsideration after the close of the record. We adhere to the previous determination in our April 2 order.

D-17. AWPP's June 11 "Motion to Withhold Final Decision" cites the fact that the NRC Staff informed the Applicant in a June 4, 1984 letter that it would be conducting routine verifications, by nondestructive examinations, of construction activities and materials. AWPP asserts, without basis and inconsistently with the routine nature of this facet of the NRC Staff's ongoing inspection program, that the plans for this inspection confirm that there is a basis to doubt the previous inspections of welds. Given the actual routine nature of the situation, there is no reason to defer this decision to await and consider on this record the results of the Staff's inspection. This is reason enough to deny the motion. In any event, even if the inspections were related to the contention, AWPP's motion does not address, let alone satisfy, the standards for reopening the record to admit a late-filed contention, and is denied for this reason as well.

E. Onsite Emergency Planning

1. Summary

E-1. In this section of the decision we rule on seventeen contentions or parts of contentions which Limerick Ecology Action (LEA) puts forward on the Applicant's emergency plan, generally called the onsite plan.[11] Issues involving the Commonwealth's and local governments' offsite plans are still pending for litigation and will be considered in a later partial initial decision. The hearings were held April 23-25, 1984 in Philadelphia. The Commonwealth took part in them under the provisions in 10 C.F.R. § 2.715(c) for the participation of interested governments. In accord with its rights under § 2.715(c), the Commonwealth also filed proposed findings, which we have considered in coming to our decisions.

E-2. LEA's contentions allege shortcomings or insufficient development in many areas of the Applicant's onsite planning: the spectrum of accidents covered by the Plan; the operation centers for emergency response; the length of time which might pass before offsite authorities were notified of an emergency; the Applicant's capabilities for predicting and assessing the radiological consequences of an accident; its capabilities for determining the location of all onsite personnel at the start of an emergency, and for monitoring them for radiation and decontaminating them if necessary; hospital care for onsite personnel who are both injured and contaminated; and the agreements with offsite organizations which would provide onsite support, the training of their personnel, and the backups for these organizations. The number and range of the contentions which were dealt with in the hearings were even greater than the number and range of the seventeen we rule on here, for LEA withdrew some contentions and parts of others between the hearings and the filing of its Proposed Findings. The course of the litigation also brought about enough changes in the contentions which remain to cause their texts as admitted to no longer adequately reflect them. Thus, in our rulings below, we paraphrase the contentions when setting out what they now allege. Their full texts may be found in a November 14, 1983 compilation by LEA.

E-3. At the hearings, the Applicant presented a panel of witnesses which included some of the Applicant's senior management officials, the Applicant's Director of Emergency Preparedness, and the Senior Health Physicist at Limerick. The Staff's one overall general witness was a Senior Reactor Safety Engineer in the Emergency Preparedness Branch, Division of Emergency Preparedness and Engineering Response, Office of Inspection and Enforcement. Both LEA and the Commonwealth took part in cross-examination of these witnesses but presented none themselves.

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11 The pertinent parts of the Plan are in the record as Applicant's Exhibit 32. However, for the sake of brevity, our citations to the Plan will be of the form, "Plan, § 6.1.1."

E-4. As set forth in our Findings of Fact on each contention detailed below, we rule in favor of the Applicant on all seventeen contentions. Except on Contention VIII-12(a), hospital arrangements for contaminated injured, our rulings are unanimous.

E-5. With a number of contentions we have found it necessary to go to the Plan's implementing procedures to decide a controversy. We are aware that by going to the procedures we may appear to have run counter to the ruling in *Louisiana Power and Light Co.* (Waterford Steam Electric Station, Unit 3), ALAB-732, 17 NRC 1076 (1983), which may appear to say that no implementing procedure is to be subject to scrutiny in a licensing hearing. *Id.* at 1107. However, we read *Waterford* less broadly. It does say that the whole body of implementing procedures need not be ready in time for challenge in a hearing, and the case wisely counsels against getting bogged down in the detail of the procedures. *Id.* We give similar counsel below in our discussion of Contention VIII-6(c), and we believe we have avoided getting bogged down in detail. However, we do not construe *Waterford* to rule that we cannot examine implementing procedures which are — as were the ones we consider below — already available and arguably necessary to determine whether certain plan provisions meet NRC planning standards and guidelines. Examining such procedures has the adequacy of the *plans* foremost in mind, and thus is in keeping with *Waterford's* reminder that the proper object of litigation is the adequacy of the plan. *See also* our Special Prehearing Conference Order, LBP-84-18, 19 NRC 1020, 1040 (1984).

E-6. As the reader may note, almost none of our citations to implementing procedures are to the record. This is because only early revisions of the pertinent implementing procedures appear in the record, in App. Ex. 33, and yet we early on discovered that the latest revisions of these procedures, filed by the Applicant after the completion of the hearing on this subject, made moot some of the controversies in this proceeding. Thus, we acquired the habit of referring to the latest revisions, even on matters which have remained unchanged from revision to revision. The parties were given an opportunity to set forth, in writing, any specific objections or other points they wished to make regarding these revisions.

2. *LEA Contention VIII-1: Spectrum of Accidents Envisioned in Plans*

E-7. Contention VIII-1 as admitted and Contention VIII-1 as argued in LEA's Proposed Findings are not the same. As admitted, this contention had alleged the onsite plan did "not encompass the spectrum of credible accidents for which emergency planning is required." The narrow factual basis of the contention was that although § 4.2 of the Plan said that the adequacy of the Plan could be demonstrated by, among other things, noting that the provisions of the Plan encompassed the radiological consequences of the "postulated accidents," Table 4-1 showed that the only accidents postulated were design basis accidents.

E-8. In reply, the Applicant argued that Table 4-2 of the Plan, which sets out responses to a variety of events, in fact included some accidents which were beyond design basis. Boyer *et al.*, ff. Tr. 9972, at 1-2. Both the Applicant and the Staff argued that the provisions of the Plan encompassed the accident-initiating conditions listed in NUREG-0654, Rev. 1, in Appendix 1. *Id.* at 2; Sears, ff. Tr. 9776, at 5.

E-9. On Contention VIII-1 as admitted, we find for the Applicant. LEA neither proffered witnesses on the issues raised by the contention nor cross-examined the witnesses of the other parties. Thus, all the evidence in the record points to the conclusion that the Plan does indeed encompass accidents beyond design basis.

E-10. As argued in LEA's Proposed Findings (PF), this contention is much broader than it was as admitted. It alleges that, whether or not the Plan recognizes initiating conditions which could lead to a severe core melt accident, the Plan does not *adequately* encompass "severe core melt accidents which are likely to result in doses exceeding the PAGs [Protective Action Guides] and to require protective actions, including evacuation of the plume exposure pathway emergency planning zone." LEA Proposed Findings at 2 (footnote omitted) and 3 n.1. The issue now is not the narrow one of whether the Plan in fact covers accidents beyond design basis, but the broader one of whether it does so adequately.[12]

E-11. The bases of this new version of Contention VIII-1 are likewise broader. As bases, the Proposed Findings on Contention VIII-1 proffer not merely a table, as Contention VIII-1 in its admitted form did, but rather "the entire record . . . established on all other contentions," and all the findings LEA proposes we make on all the other contentions. LEA PF at 1-2, 5. Thus, LEA argues, the Applicant cannot carry its burden of proof by merely citing a table of initiating conditions. "The *Plan* in its entirety must be examined to determine whether the

[12] The Board notes that the NRC dces not intend that emergency plans must aim at the impossible in an emergency, namely the prevention of any dose which exceeds the relevant PAG, or on the other hand, that PAGs are acceptable dose levels in situations other than emergencies. *See* NUREG-0396/EPA 520/1-78-016 (December 1978), at 4. Rather, PAGs are intended by the NRC to be simply levels of radiation dose which when predicted or exceeded trigger protective actions designed to minimize the impacts of the actual or threatened doses.

Plan's operation in fact will encompass the sequence of events which would occur in a severe accident." LEA PF at 7 (footnote omitted).

E-12. It is difficult to view this new version of Contention VIII-1 as more than a kind of summary of LEA's other onsite planning contentions. It cites them as bases and proposes no remedy of its own. It is arguable that given its newness and redundancy we are not obliged to rule on it at all.

E-13. However, treating the VIII-1 of the Proposed Findings as both admitted and distinguishable from a mere summary of the other onsite contentions, we nonetheless again find for the Applicant. The Findings of LEA we accept on the other contentions are far too few to support so broad a claim as that the onsite plan taken as a whole does not adequately encompass the spectrum of credible accidents, both design basis and beyond.

3. *LEA Contention VIII-3: Onsite Monitoring Systems*

E-14. As admitted, this contention was quite broad, alleging that the onsite plan did not identify and establish the onsite monitoring systems called for by Evaluation Criterion H.5 in NUREG-0654, ch. II. These systems cover a variety of phenomena, among them wind speed and direction, reactor coolant levels, radioactivity, and fire. The data from these monitoring systems would be used to initiate emergency action levels. In its written testimony, the Applicant listed the sections of the Final Safety Analysis Report (FSAR) in which the monitoring systems called for by Criterion H.5 are discussed. Boyer *et al.*, ff. Tr. 9772, at 2-5. The contention now concentrates on the adequacy of three of these systems. We find that the first of them is adequate, and that, in the circumstances, the Staff should make the final evaluation of the other two.

E-15. The first of the three systems monitors for certain toxic chemicals which could incapacitate control room operators. Criterion H.5 does not explicitly call for a chemical release monitoring system, but the Applicant has installed one nonetheless, and its inclusion seems necessary given the goals of the Criterion. Thus there can arise an issue over its adequacy. LEA claims that the system does not cover all the chemicals which might present a hazard to control room operators. For the reasons given below, the claim is true, but not significant.

E-16. The Applicant's determination of which chemicals present a hazard to control room operators is set out in § 2.2.3.1.3 of the FSAR. The determination rests on this definition: "A chemical is considered a potential hazard if it is stored or transported nearby in such quantities that its concentration at the control room air intake following a spill could exceed the toxic incapacitation level." FSAR at 2.2-7. After consultation with Conrail, surveys of nearby manufacturers and users of toxic chemicals, and a modeling of toxic plume transport, the Applicant determined that 6 of 154 chemicals evaluated fit the definition just quoted. All six are covered by the Applicant's chemical release monitoring system. *See* FSAR § 2.2.3.1.3. Thus, in testimony LEA does not mention, one of the Applicant's witnesses could say, "we are monitoring for all the chemicals which have the capability of resulting in concentrations in the control room which would incapacitate the operators." Tr. 10,207 (Boyer).

E-17. Of course, it is possible, but extremely improbable, that one of the chemicals not covered by the monitoring system would be released, say by a train derailment, in such a way as to threaten the control room. However, the Applicant has already exceeded the standards of Criterion H.5 in this regard, and LEA has raised no question about the adequacy of the consultation, surveys, and modeling which the Applicant used to determine which chemicals the monitoring system would cover. Much of the analysis which led to the determination followed NRC guidelines in various documents. *See* FSAR § 2.2.3.1.3. We see no legal or practical point in requiring that the Applicant's monitoring system cover more chemicals than the six it now covers.

E-18. The second of the monitoring systems LEA is concerned about is the meteorological system. Data from two meteorological towers, called Met-Towers 1 and 2, are direct inputs in a system the Applicant would use to predict cumulative population dose. Tr. 10,187-88 (Murphy). The dose prediction would be used in determining what emergency measures to initiate. LEA notes that the Staff has said that Met-Tower 1 is close enough to the cooling towers for there to be distortion of Met-Tower 1's readings of wind speed and direction. *See* NUREG-0991, Safety Evaluation Report related to the operation of Limerick Generating Station, Units 1 and 2 (SER), August 1983, at p. 2-19. The Staff has said that it will include this subject in its review of emergency preparedness. *Id.* LEA proposes that we "require, as part of any order, a Staff report on the evaluation and resolution of these concerns prior to any fuel loading or testing." LEA PF 18.

E-19. We find that any such requirement is unnecessary. First, in the course of its review of emergency preparedness, the Staff will be preparing a report which will include evaluation of the impact on emergency planning of the possible distortions in the data from Met-Tower 1. SER at p. 2-19, p. 13-17. LEA has offered no evidence that

that report will be inadequate. We see no gain to safety from simply including that report in one of our orders.

E-20. Perhaps more important, a glance at the SER passage on Met-Tower 1 reveals that the Staff's concern about its location is minimal. There the Staff says that meteorological measurements at Met-Tower 1 "will probably be affected by the cooling towers less than 10% of the time," and probably not at all in a slow wind. *Id.* at p. 2-19. Also, the Staff says that the potential for *significant* distortions of Met-Tower 1's measurements of wind speed and direction is "small." *Id.* Indeed, the Staff concludes that the location of Met-Tower 1 is "satisfactory." *Id.* LEA does not dispute any of these statements.

E-21. The last of the three systems or pieces of equipment LEA is concerned about under Contention VIII-3 is the wide-range water level transmitter used to monitor the level of the coolant in the reactor. As is the case with the other systems and equipment considered in this contention, data from the wide-range water level transmitter would be used in an emergency to help determine the appropriate level of emergency response. Regulatory Guide 1.97 calls for the reference leg of the transmitter to be located at the required tap at centerline of the main steam lines, but the Applicant, excepting to this guidance, has put the reference leg 5 feet below the location the Regulatory Guide prefers. *See* the FSAR at 7.5-27, in App. Ex. 38. Moreover, the Staff is in the midst of reviewing the whole of Applicant's treatment of Regulatory Guide 1.97. *See* the SER, § 7.5.2.3, and SER, Supp. 1, at p. 1-2. LEA would have us therefore conclude that the water level monitoring system is not yet "established" and so does not conform to Criterion H.3, the legal basis for all parts of Contention VIII-3.

E-22. We do not so conclude. First, it must be remembered that Regulatory Guide 1.97 is guidance, not regulation. Therefore, an Applicant need not conform to some particular guideline in the Guide if it has good reason not to. The Applicant has chosen to place the reference leg of the wide-range water level transmitter below where Regulatory Guide 1.97 would have it placed in order to "eliminate long runs of exposed sensing line tubing that contribute to erratic indication." FSAR at 7.5-27, in App. Ex. 38. LEA doesn't even mention this reason, let alone criticize it. Nor is there in the record any indication that the Staff will find the reason inadequate in the course of its review of the Applicant's treatment of Regulatory Guide 1.97.

E-23. Thus, we have ruled against LEA on all three parts of Contention VIII-3. In relation to the second and third parts, our rulings have been the result largely of LEA's nearly identical approaches to the issues of the locations of Met-Tower 1 and the wide-range water level transmitter: In both cases LEA has chosen to second a concern the Staff has raised in the SER, but LEA has added nothing to the record on either issue, either by testimony or cross-examination. The result is that LEA has in effect asked us to be not adjudicators of conflicting claims each backed by a part of the record, but solely reviewers of Staff work. It is not our function to review Staff work except in the context of adjudication proper. Therefore, we leave to the Staff the final determination of the adequacy of the locations of Met-Tower 1 and the wide-range water level transmitter.

4. LEA Contention VIII-6(a): Mutually Agreeable Bases for Notification of Organizations with Responsibility for Onsite Augmentation

E-24. Evaluation Criterion E.1 of NUREG-0654, ch. II, says that "[e]ach organization shall establish procedures which describe mutually agreeable bases for notification of response organizations . . . ." LEA contends that the onsite plan does not demonstrate that mutually agreeable bases exist for notification of organizations with responsibility for onsite augmentation. Arguing more specifically, LEA says that each of the three organizations it regards as having responsibilities for onsite augmentation — Linfield and Limerick Fire Companies, and Goodwill Ambulance Corps[13] (LEA PF 27) — has offsite responsibilities which can conflict with its responsibilities on site, and that for there to be the mutually agreeable bases called for in Criterion E.1, there should be something in either the Plan or the letters of agreement with these organizations which "provides a resolution . . . of conflicting claims upon these very limited resources," or which "describes how these resources already committed off site would be notified and required to leave offsite duties to travel to the site." LEA PF 31.

E-25. For the reasons set out below, we find that the letters of agreement between the Applicant and the three organizations LEA names in this contention conform to Evaluation Criterion E.1 of NUREG-0654, ch. II, and that the real issue which LEA raises in this contention — the adequacy of the resources of these three organizations — is litigated in other contentions.

E-26. LEA is confusing two possible agreements, one on the allocation of allegedly scarce resources, and the other, more properly the sub-

[13] The Applicant argues that Goodwill cannot be construed to have any responsibilities for onsite augmentation. Applicant's Reply Findings at 5. Given the grounds of our decision on this contention, we need not determine whether Goodwill's responsibilities include augmentation of onsite functions.

ject of the cited Criterion E.1, on the means of notification of the need for the resources. The contention alleges nothing about how the three organizations in question are to be notified of the need for their resources, only that the Applicant and the three organizations have not agreed on whether and when onsite needs should take priority over offsite. Thus, the issue the contention raises is whether the resources of these organizations are adequate where conflicting needs for these resources might arise. This issue is the principal one in Contentions VIII-11 and VIII-12(b), and thus is redundant here.

E-27. Evaluation Criterion E.1 seeks not adequacy of numbers but rather agreement which is likely to preclude confusion during an emergency about what constitutes official notification. During an emergency, a response organization should not have to wonder whether a call for its resources was made by a responsible party. The agreements with each of three organizations LEA names in this contention appear to preclude such confusion. Each of the two fire company letters says that the fire company which is the subject of the letter will receive notification from the "Montgomery County Division of Public Safety, Office of Communications." App. Exs. 44 and 45. According to unchallenged testimony of one of the Applicant's witnesses, the Office of Communications is aware of these agreements. Tr. 10,007-08 (Kankus). The letter of agreement between Goodwill Ambulance Corps and the Applicant says that Goodwill and the Applicant's Medical Director have "reviewed arrangements for the Goodwill Ambulance Unit to respond to a call for assistance" to the Limerick plant. Plan, Appendix A, Item 10.

5. *LEA Contention VIII-6(c): Notification to Offsite Authorities*

E-28. As did other onsite emergency planning contentions, VIII-6(c) changed in the course of being litigated. The contention in its admitted form is now only a secondary part of the contention in its litigated form. As admitted, VIII-6(c) is aimed only at one provision of the onsite plan. Section 6.1.1 provides that notification to governmental authorities of an emergency event "shall be within about fifteen minutes after classifying the event." LEA alleges that this provision does not conform to the guidance in NUREG-0654, Appendix 1, at p. 1-3, which LEA interprets as saying that notification should take place within 15 minutes "not from *classification,* but from the time that operators recognize that an emergency event has occurred." LEA PF 37 (footnote omitted).

E-29. However, during litigation VIII-6(c) expanded and became aimed not only at the Plan but also at some of the implementing procedures under it. LEA claims that given the provisions of certain implementing procedures, the time between classification of the emergency event and notification of offsite authorities — let alone the time between recognition that the event has occurred and notification — may "easily" be longer than 15 minutes. LEA PF 48.

E-30. Thus Contention VIII-6(c) now has two parts; they can be summarized thus: First, the plan measures the 15 minutes to notification from too late a moment, and second, even if it should be measured from the later moment, notification may well be delayed beyond 15 minutes. Each of the two parts of the contention is a fall-back position for the other, but the second part has been foremost in the litigation of VIII-6(c). Below, we consider the second part first. Happily, the issue it raises has become largely moot because of revisions of the implementing procedures, revisions LEA and, surprisingly, the Applicant did not inform the Board of. We end our discussion of VIII-6(c) with an examination of the NUREG-0654 guidance on which LEA relies in claiming that the Plan measures the 15 minutes from too late an event. For a number of reasons we conclude that NUREG-0654 intends that the 15 minutes be measured from classification of the emergency event. Thus, the Plan conforms to the guidance.

E-31. To support its claim that notification could easily be delayed beyond 15 minutes after classification, LEA examined in some detail EP-103, the implementing procedure which provides guidelines for the site response to the Alert level of emergency action. EP-103 lists several tasks to be performed by the Emergency Director, or the Interim Emergency Director if the Emergency Director is not available. The task of filling out the Alert Notification Message to be sent to offsite authorities is the seventh item in the list, after such apparently time-consuming tasks as directing evacuation of the site. Citing testimony by one of the Applicant's witnesses, LEA claims that just the first listed task alone, verification of the emergency classification, could well take anywhere from 10 minutes to an hour. LEA PF 46. LEA could have made similar arguments about what, at the time of the hearing on this contention, were the current texts of EP-102, EP-104, and EP-105, the other three documents which provide guidelines on site response at one of the four levels of emergency action the NRC has established. *See* NUREG-0654, Appendix 1.

E-32. However, in the latest revisions of EP-102 (Unusual Event), EP-103 (Alert), and EP-104 (Site Emergency) — Revision 3 of each — the notification tasks are listed immediately after verification of the emergency classification, which is still listed first in each of the three documents. No Revision 3 has been issued yet for EP-105 (General

Emergency), the last of the four implementing procedure documents on site response at the four emergency action levels, but, given the latest revisions of the first three documents, there is no reason to think that there will not be a revision of EP-105 which will list notification tasks right after verification.[14]

E-33. With these latest changes in implementing procedures, the claim in Contention VIII-6(c) that notification might well come more than 15 minutes after classification of an emergency event depends wholly on whether verification of the classification could take more than 15 minutes, for verification is now the only step between classification and notification. As we've said, LEA claims that verification could take up to an hour. LEA PF 46.

E-34. The claim is misleading. It is stated generically, without mention of the single example on which it rests, and rests not at all firmly. The example is a wreck on site of a train carrying toxic chemicals. It could take up to an hour to obtain a report from Conrail on the contents of damaged cars. Tr. 10,101 (Boyer). However, if the chemicals were identified by labels on the cars which carried them, as they usually are, it would take only 10 to 15 minutes for someone sent from the Limerick plant to the site of the wreck to learn what the chemicals were. *Id.* at 10,100 (Boyer). Moreover, under EP-101, Rev. 1, and EP-102, Rev. 3, the mere fact of a train derailment within the site boundary is enough to trigger notification of offsite authorities. Therefore, there is no evidence in the record that verification of a classification could delay notification.

E-35. Thus, as the relevant implementing procedures now stand, there is reasonable assurance that notification of offsite authorities will occur within 15 minutes of the classification of an emergency event.[15]

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[14] Even though the Applicant sent these latest revisions to the Board and the other parties on June 11, 10 days before LEA filed its Proposed Findings and nearly a month before either the Applicant or the Staff filed theirs, it appears that no party knew of the changes we have just described. We might have expected LEA and, in particular, the Applicant to have noted changes in documents which figured so prominently in their Proposed Findings. On the other hand, there is illustrated here one of the difficulties which inheres in trying to cope with implementing procedures in litigation, rather than focussing on the plans, as case law would generally have us do. *See Waterford*, ALAB-732, *supra*, 17 NRC at 1107. Taken altogether, the implementing procedures are a maze of details undergoing more or less constant revision in a process which sometimes can be beyond the reach of even the Applicant's counsel, as apparently it was here.

[15] Even if the latest revisions of the implementing procedures had not made largely moot the issue of the length of time between classification and notification, we might well have found for the Applicant on this issue, principally because it would appear that, with the exception of site evacuation, none of the Emergency Director's tasks which in the earlier texts of the procedures came before notification would consume more time than a quick telephone call would; and even "directing" site evacuation requires the Director to perform what is arguably only a short series of simple acts. *See* EP-305, Rev. 1, § 9.1.

The Applicant makes two other arguments about the earlier versions of the procedures, but neither is persuasive. The first is that site evacuation, which in the earlier versions preceded notification, would be initiated and "directed" by the Emergency Director but that classification of an event and notification of

(Continued)

All that remains of Contention VIII-6(c) therefore is the original part of it, the claim that the onsite plan should measure the 15 minutes not from classification, but from the time onsite personnel recognize that an emergency event has occurred. LEA rests its claim on the following sentence from NUREG-0654: "The [15 minutes] is measured from the time at which operators recognize that events have occurred which make declaration of an emergency class appropriate." *Id.*, Appendix 1, at p. 1-3. The meaning of this sentence is not crystal clear. LEA's reading of it is certainly plausible, but three arguments point to a conclusion that the sentence means that the Applicant should be able to notify offsite authorities within 15 minutes of *classification* of an emergency event.

E-36. The first two arguments are textual. First, immediately before the sentence we just quoted from NUREG-0654 comes this one: "Prompt notification of offsite authorities is intended to indicate within fifteen minutes for the unusual event class and sooner (consistent with the need for other emergency actions) for other classes." *Id.* Here the time to notification is a function of the emergency class and therefore must be measured from classification.

E-37. Second, the 15-minute requirement is stated less ambiguously in Appendix E of 10 C.F.R. Part 50: "A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency." *Id.*, § IV.D.3. LEA acknowledges that this regulation measures the 15 minutes from classification, but apparently, LEA also wants to treat the regulation in Part 50 and the

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offsite authorities would be performed by the Shift Superintendent. Thus, the Applicant argues, site evacuation would not have to precede notification: The different personnel assigned these tasks could perform them simultaneously. Tr. 10,121-22, 10,124-25 (Ullrich). However, this argument is difficult to square with the texts of the implementing procedures. EP-103, Rev. 3, is typical. It assigns all three tasks — classification, direction of site evacuation, and notification — to what it calls the "(Interim) Emergency Director." The Interim Emergency Director is the Shift Superintendent (Plan, § 5.2.1.1); he is to serve until the Emergency Director, who is the Station Superintendent (*id.*, § 5.2.1.2), takes over (*id.*, § 5.2.1.1). Thus, although the Applicant's witness says that EP-103 assigns the Shift Superintendent and the Emergency Director to different tasks, it appears that EP-103 actually assigns them at most to different *times*, and therefore that if the Shift Superintendent were to stay long enough, or the Emergency Director to come early enough, under EP-103, Rev. 1, either officer could well have to perform all three tasks.

The Applicant's other unpersuasive argument is that notification and site evacuation could be simultaneous because "[t]here is no evidence in the record that the effectiveness of Applicant's implementing procedures . . . is dependent upon the execution of steps within a procedure in any particular order." Applicant's Reply Findings at 7. Such a claim is implausible *a priori*, but it is also difficult to square with certain particulars in the procedures. For instance, even a witness for the Applicant testified that in EP-305, Rev. 1, which governs site evacuation, the Emergency Director would have to perform § 9.1.1.3, notification of Security, before § 9.1.1.7, activation of the alarm, so that Security would have time to prepare for evacuation. Tr. 10,102-04 (Ullrich). Indeed, the very revisions which have placed notification just after verification would indicate that the order in which the tasks are listed is intended to be the order in which they are to be performed.

guidance in NUREG-0654 as different requirements, as if the Applicant had to be capable of notification within 15 minutes of two quite different moments. LEA PF at 14 n.1. We do not see how this makes sense.

E-38. The third and last argument is practical: Recognition of an emergency event and classification of it for the purposes of site response are, in relation to notification, barely separable; thus measuring the 15 minutes from classification could not cause significant delay. Apparently, LEA imagines that plant personnel will first recognize that something has gone wrong and then may have to spend some time determining how serious it is before they put it in an emergency level classification: LEA claims that classification may be delayed "for as long as 20 minutes beyond event recognition under some circumstances, e.g., a transient plus failure of the core shutdown system, in which the symptoms of the event will be the initiation of the liquid control system, but the failure of the core to become subcritical [sic]." LEA PF 38, *citing* Tr. 10,085-86 (Boyer).

E-39. While one witness of the Applicant did say that it could take "20 minutes say" after the initiation of the liquid control system to determine whether the reactor was becoming subcritical (*id.*), another witness of the Applicant pointed out that under EP-101, Rev. 1, at 15, even while the operator was initiating the liquid control system an Alert level of emergency response would probably be declared because of the failure to automatically scram, combined with a failure of a scram to bring the reactor subcritical. Tr. 10,087-88 (Kankus). Notification of offsite authorities would follow declaration of the Alert level, not the determination of whether the liquid control system had brought the reactor subcritical. Tr. 10,088 (Kankus); *see also* EP-101, Rev. 3. Similarly, as we've noted before, in the case of a train derailment on site, notification of offsite authorities would follow recognition of the derailment, not determination of whether toxic chemicals were released in the accident.

E-40. Thus, no period of uncertainty about how threatening an initial event was would delay notification, for while *re*classification might come more than 15 minutes after an initial event, notification would not, since even the initial event would fall within a classification which required notification to offsite authorities. We note also that as the implementing procedures now stand, reclassification would bring about renotification well within 15 minutes.

E-41. In conclusion, we find that NRC regulations and guidance require that notification of offsite authorities follow within 15 minutes of classification of an emergency event, and that as the implementing procedures now stand, there is reasonable assurance that this time constraint would be met in an emergency.

6. *LEA Contention VIII-8(b): Adequacy of Emergency Facilities, Equipment and Supplies*

E-42. In this contention, as in VIII-3, LEA focuses on areas still under review by the NRC Staff. Here, unlike in VIII-3, the Staff has not identified a possible shortcoming in the Applicant's work, but at the time of the hearing on onsite planning, the Staff's review was still far from complete.

E-43. At the time of the hearing, in April 1984, the Applicant was still in the process of establishing three emergency facilities called for by NRC guidelines in various documents: the Emergency Operations Facility (EOF), the Technical Support Center (TSC), and the Operations Support Center (OSC). The Staff's witness estimated that the three facilities were about 75% complete (Tr. 10,062 (Sears)), and that the Staff's review of the facilities would not be available for about another 3 months (Tr. 10,273 (Sears)).

E-44. In view of the importance of these three facilities, and the work which at the time of the hearing remained to be done on them, LEA asks that before we make findings on the three facilities, the Staff make its review of them available to the Board and the parties and the parties be given opportunity after the review becomes available to propose additional findings on the adequacy of the facilities. LEA PF 54.

E-45. Having balanced certain considerations, we have decided to close the record on these facilities now. On the one hand, it is crucial that these facilities be adequate to the uses which would be made of them in an emergency. Moreover, determining their adequacy would appear to require some judgment, considerably more than determining the adequacy of, say, the location of Met-Tower 1 or a wide-range water level transmitter. *See* our discussion of Contention VIII-3. Thus an outside observer such as an intervenor could be both interested in the outcome of the Staff's review and in a position to reasonably and fruitfully disagree with the Staff's review.

E-46. On the other hand, the review work which the Staff had yet to do at the time of the hearing was hardly novel, nor have such facilities been the objects of great controversy in proceedings on other plants. Limerick is not the first plant to use the instrumentation and equipment which will be in the three facilities. Tr. 10,065 (Sears). Moreover, the criteria for judging the facilities — NUREG-0696 and -0818 — are well known and not particularly controversial — and not at all controversial in this proceeding.

E-47. But last and perhaps decisive, litigation on emergency planning is first and foremost concerned with the plans; yet, even though a certain amount of information about these three facilities is available in §§ 7.1.2., 7.1.3, and 7.1.4 of the onsite Plan, LEA has raised no issue based on any of this information. Even now, LEA raises no specific concern that any of these facilities will not meet a particular requirement.

E-48. On balance, we find that LEA has not shown any justification for keeping the record open.

7. LEA Contention VIII-10(a): Delineation of Authority in Certain Letters of Agreement

E-49. LEA contends here that the Applicant's agreements with local agencies do not conform to Evaluation Criterion B.9 of NUREG-0654, ch. II, because they do not delineate the authorities, responsibilities, and limits on the actions of the agencies, but merely briefly describe the general nature of the service to be provided. Though stated quite broadly, the contention deals only with the Applicant's agreements with the Linfield and Limerick Fire Companies and the Goodwill Ambulance Unit.

E-50. The issue LEA raises about the agreements with the fire companies is that although the letters do say that the fire companies will be "under the direction and control of Philadelphia Electric Co." (App. Exs. 44 and 45), the letters do not reflect, but should, what LEA thinks is the more complicated division of authority which the Applicant actually has in mind: The fire companies would not have authority to decide how to fight an onsite fire, but would to decide what equipment to bring, though not to decide where to place it; they would also have authority to decide which of their personnel to bring, but not to decide how long they would fight a given fire. LEA PF 58 (*citing* Tr. 9968-69 (Kankus)). LEA claims that unless such divisions of authority are delineated in the agreements, there is likely to be conflict and confusion when the Applicant's fire-fighting personnel, who have had only a 2-day course in fire fighting, try to assert authority over experienced municipal fire fighters. LEA PF 59.

E-51. We find that the agreements are adequate as they stand. All the divisions of authority which LEA elicited in cross-examination from one of the Applicant's witnesses, and which LEA apparently thinks are too confusingly arranged to be left out of the agreements, follow directly from the single principle laid down by the same witness: "Again, before they [the fire companies] come to the site, they have — the decision is theirs to determine what they will bring. Once they're on the site they're under the direction of our fire-fighting personnel." Tr. 9969 (Kankus). And this principle is only a paraphrase of the one already stated in the letters of agreement, that while on the site the fire companies will be under the direction and control of Philadelphia Electric. There is no need for the letters to spell out the direct consequences of so simple a principle.

E-52. There is no reason either to think that the fire companies will resist the application of the principle. They have, after all, agreed to it, and it makes good sense, for, of all the fire-fighting personnel, only the Applicant's will be well informed about the layout of the plant, the location of electrical equipment that may be feeding the fire, ventilation systems, and the like. Tr. 10,012-13 (Ullrich). Moreover, personnel named by the fire companies will be trained by the Applicant (App. Exs. 44 and 45) and so will be accustomed to the division of responsibility the principle entails.

E-53. Last, we note that the Applicant's fire-fighting personnel have something more than just a superficial 2 days of training in fire fighting. Unrefuted testimony has it that the 2 days will be "intensive." Tr. 9970 (Kankus). The course is well established, being given by the Applicant's fire school, which has been in service for a number of years. *Id.*; Tr. 9971 (Reid, Boyer). Finally, there will be annual retraining. Tr. 10,008-09 (Ullrich).

E-54. There is even less reason to make a finding that the Applicant's agreement with Goodwill Ambulance is inadequate. One of the Applicant's witnesses testified that the only authority the Applicant would exercise over Goodwill's personnel would be that exercised by an escort who would keep them away from areas where they were not needed and would lead them to where they were needed. Tr. 9967-68 (Kankus). Such "authority" is more aptly called "help," and is so self-evidently what Goodwill personnel would need in an environment with which they were not familiar that it need not be spelled out.

8. LEA Contention VIII-11: Offsite Augmentation of Onsite Fire-Fighting Capabilities

E-55. LEA once again contends that the agreements between the Applicant and Linfield and Limerick Fire Companies for augmentation of the Applicant's own fire-fighting capabilities are not adequate. *See also* our discussions of LEA Contentions VIII-6(a) and VIII-10(a). Here the difficulty LEA sees is that there is a chance that the two fire companies would have offsite duties that would keep them from performing their onsite duties. Under the offsite emergency plan for the Limerick plant, both fire companies are assigned to do route-alerting if notification to the public should be required while the siren system is inoperable. Tr. 9982 (Kankus). LEA admits that the probability of there being both a general emergency and a failure of the siren system "may be relatively low." LEA PF 63. Nonetheless, asserting the principle that the adequacy of emergency plans is to be measured "in light of the circumstances of accidents which may require evacuation of the plume exposure EPZ" (LEA PF at 27 n.1), LEA claims that the Applicant should make some further arrangements, ones which will secure offsite augmentation even when route-alerting is necessary.

E-56. The Applicant and the Staff emphasize that the plant is "basically self-sufficient in fire-fighting capabilities." *See* App. PF 40-41, and Staff PF 24. The Applicant goes so far as to claim that its fire detection and suppression capabilities, together with the configuration and safety systems of the plant, are enough to suppress any credible fire at the plant, or to assure that if the fire could not be suppressed the damage would be limited enough to permit the plant to be safely shut down. Boyer *et al.*, ff. Tr. 9772, at 12. Both the Applicant and the Staff also claim that in the eighty-six times the Linfield Fire Company was called out last year, it was unavailable only once. *Id.* at 13; Staff PF 24.

E-57. These arguments are not very persuasive. The Applicant is not so self-sufficient in fire fighting that there has not been the need to arrive at an agreement with a second fire company. Moreover, it may be that the Linfield Company was unavailable only once in eighty-six times to fight an offsite fire, but that is not quite relevant, for the question here is not how often a fire company might be called on to fight two offsite fires at once, but whether it might be called on to fight an onsite fire and do route-alerting at the same time.

E-58. Nonetheless, we find that it is unnecessary for the Applicant to make further arrangements for augmentation of its fire-fighting capabilities. The principle that emergency plans must be judged with evacuation in mind is a good one. But probabilities must be kept in mind. It is prudent to assume, given the emergency planning regulations, that offsite evacuation could be required while there is a fire at the Limerick site. However, the further possibility that the fire companies could be called on to fight a fire at the plant and do route-alerting at the same time is just too remote. Not only is it improbable, as LEA admits, that the siren system would fail in a general emergency, it is also improbable that during the same emergency there would be a fire which exceeded the Applicant's considerable fire-fighting capabilities, the "basic self-sufficiency" of which LEA chooses not to question. The Applicant's planning for augmentation of its fire-fighting capabilities already goes beyond what prudence would suggest as a minimum. We will not require that it go still further.

9. *LEA Contention VIII-12(a): Emergency Hospital Care for the Contaminated Injured*

a. Unanimous Board Findings

E-59. LEA here contends that there is not yet reasonable assurance that adequate measures would be taken in a radiological emergency to care for onsite personnel who suffer both traumatic injury and contamination. Such persons are called "contaminated injured." *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-83-10, 17 NRC 528, 535 (1983).

E-60. Planning Standard (b)(12) in 10 C.F.R. § 50.47 requires that "arrangements are made for medical services for contaminated injured individuals." The first Evaluation Criterion under this Standard, Criterion L.1 of NUREG-0654, ch. II, would require that "each organization shall arrange for local and backup hospital services having the capability for evaluation of radiation exposure and uptake, including assurance that persons providing these services are adequately prepared to handle contaminated individuals."

E-61. Standard (b)(12) and the evaluation criteria which elaborate on (b)(12) aim principally to secure adequate planning for emergency treatment of traumatic injury, not of severe radiation exposure. Only in extreme cases does such exposure require immediate treatment. *San Onofre, supra,* 17 NRC at 535-36. Standard (b)(12) and the criteria under it are concerned with radiation exposure principally because medical personnel treating traumatic injury sustained in a radiological emergency may well have to reckon with contamination as an obstacle to adequate treatment of the traumatic injury.

E-62. The Applicant has made arrangements for the treatment of contaminated injured with two hospitals. Under these arrangements, Pottstown Memorial Medical Center (PMMC) would be the main receiving point for onsite personnel who are contaminated injured. *See* App. Ex. 42. Through an agreement with the Radiation Management Corporation (RMC), which is the Applicant's contractor, the hospital of the University of Pennsylvania (HUP) in Philadelphia would receive contaminated injured when it could provide specialized personnel and equipment PMMC could not. *See* App. Ex. 43. HUP would also assist with the treatment of persons suffering severe radiation exposure with no traumatic injury. *Id.*; Tr. 9804-05 (Linnemann); and App. Ex. 40.

E-63. However, PMMC is less than 2 miles from the Limerick plant (Tr. 9831 (Linnemann)), and HUP is a 45-minute drive from the plant (Tr. 9844 (Linnemann)). LEA wants us to rule that the Applicant should also make arrangements for care of the contaminated injured with a hospital less vulnerable to evacuation than Pottstown is, but also closer than HUP is, and thus more accessible for the treatment of traumatic injury. LEA PF 103. The majority rules against LEA on this issue. As noted in Judge Brenner's dissent, he would find for LEA on this part of Contention VIII-12(a).

E-64. LEA also wants us to rule that the implementation of the Applicant's arrangements with PMMC is in its "utter infancy" and therefore that there is not yet reasonable assurance that in a radiological emergency PMMC would be able to give adequate care to the contaminated injured. LEA PF 102. We do not so rule. We discuss the implementation of the Applicant's arrangements with PMMC first.

E-65. As of late April 1984, the time of the evidentiary hearing on onsite emergency planning, and 3 months before the scheduled emergency preparedness exercises, PMMC personnel were neither trained nor equipped to perform their roles under the agreement between PMMC and the Applicant. Tr. 9813-14, 9818 (Linnemann). Thus, LEA speaks of the "infancy" of the implementation of that arrangement. However, on the record before us, it would appear that 3 months would be ample time for training and equipping PMMC personnel, given the training and equipment required and the experience of the trainer.

E-66. As to training, PMMC personnel will not be wholly unfamiliar with the plans for treating contaminated injured, for those plans are an elaboration of plans already in effect at PMMC for the treatment of traumatic injury. Trauma is the first concern of treatment of the contaminated injured. PMMC's current disaster plan is adequate for trauma and requires only an addition dealing with contamination. Tr. 9813-14 (Linnemann). The addition will cover such important, but not especially complicated, matters as selecting a radiation emergency area, limiting contamination to that area, and seeking consultation and dose evaluation. Tr. 9814-15 (Linnemann). Training in accord with the addition is a matter of days only. Although specialized treatment procedures for contaminated injury victims have not been finalized, Dr. Roger Linnemann stated that RMC, PECo, and Pottstown Hospital are compiling these procedures which, along with training, will be completed by mid-July. Tr. 9812-13 (Linnemann). The training documents to be used at Pottstown will be similar to those used at HUP and other hospitals across the country. Tr. 9828-29, 9932 (Linnemann). The training for Pottstown Hospital employees shall include instruction in the biological effects of ionizing radiation, classification of acute radiation injuries, and in the initial and emergency room treatment of radiation injuries. Tr. 9830 (Linnemann). It is expected to consist of three sessions lasting 2 days each, three drills, and a field exercise, the drills and exercises to be evaluated by FEMA and the NRC. Tr. 9903, 9954 (Linnemann). The Pottstown Memorial Hospital will receive training on a semiannual basis. Tr. 9828 (Linnemann). Finally, the trainer, RMC, is experienced, maintaining, as it does, similar programs for a number of nuclear power plants. *See* Boyer *et al.*, ff. Tr. 9972, at 9-10; *see also* Tr. 9915 (Linnemann).

E-67. As to equipment, again on the record it appears that, with one exception, nothing is required which is especially difficult to acquire: Radiation instrumentation, bath arrangements which permit collection of contaminated water, decontamination supplies such as soaps known to be effective in removing radiation from the skin, and containers for taking samples to determine a patient's dose. Tr. 9816-18 (Linnemann). One piece of radiation instrumentation is both expensive and difficult to maintain: a whole-body counter, which is used to determine the dose a patient has received internally. However, RMC maintains a whole-body counter in a mobile unit in the Philadelphia area. Therefore, there is no need for PMMC to acquire such a counter as a prerequisite to implementation of the Applicant's arrangements with PMMC. As for the other equipment listed above, the Applicant has agreed to supply whatever is necessary and not already in PMMC's possession. Tr. 9818-21 (Boyer).

E-68. In conclusion, we see no obstacle to the timely completion of the training and equipping of PMMC personnel. LEA's sole argument in this part of Contention 12(a) appears to be that the 3 months between the hearings and the preparedness exercises would not be time enough for the training and equipping we've just described. However, LEA said nothing to counter the indications in the record that 3 months would be enough. Therefore, we find that there is reasonable assurance that PMMC will be trained and equipped to give adequate care to the contaminated injured in a radiological emergency. Of course, any particular deficiencies which may be disclosed by the emergency planning exercises will have to be corrected under the auspices of FEMA and the NRC Staff.

E-69. LEA's principal concern is about the locations of the hospitals with which the Applicant has made arrangements. PMMC, being less than 2 miles from the plant, appears to be potentially vulnerable to having to be evacuated in a general emergency, while HUP, being 45 minutes away, might appear, in LEA's view, to be too far away to be adequate backup for treatment of traumatic injury if PMMC had to be evacuated.[16] LEA is contending that HUP should not be the sole backup for PMMC, not that either PMMC or HUP should not be among the hospitals assigned responsibility for the contaminated injured. The Applicant and the NRC Staff both agree that since traumatic injury is much more likely than evacuation, prudence requires that the hospital assigned the treatment of traumatic injury be reasonably close to the plant. *See* Tr. 9929-30 (Sears) and Tr. 9906 (Linnemann). Contamination is really the secondary part of the whole problem. It is the patient's life that is important. Tr. 9844-45 (Linnemann). LEA appears to acknowledge this counsel of prudence. *See* LEA PF 90. We agree.

E-70. Borrowing a phrase from the Staff, the Applicant argues that the probability of a hospital having to evacuate during a radiological emergency is "vanishingly small." *See* Tr. 9941 (Linnemann) and Tr. 9930 (Sears). The Applicant's chief witness on this contention, one of the officers of RMC and a medical doctor as well as an Associate Professor at the University of Pennsylvania's School of Medicine, says, "Evacuating a hospital is a pretty serious matter, or an immediate life-threatening situation, and I don't see a release from a nuclear power plant that would be life-threatening." Tr. 9941 (Linnemann).

E-71. The Applicant further argues that even if PMMC had to evacuate, adequate backup would exist. If time permitted, the contaminated injured could be taken to HUP (Tr. 9906-07 (Linnemann)), and if the injury required earlier treatment than HUP could provide, the patient could be taken to one of the several hospitals which are nearer the plant than HUP is. Tr. 9912-14 (Linnemann); *see also* Tr. 9906-11 (Linnemann). Neither the Applicant nor RMC have made arrangements with any of these other hospitals to receive contaminated injured from the plant, but the Applicant argues that, even so, none of these hospitals would refuse to accept a contaminated injured patient, for all of them are accredited by the principal national accrediting organization, the Joint Committee on Hospital Accreditation (JCHA). The JCHA requires that each accredited hospital have some plans for treating contaminated injured patients. Tr. 9912-14 (Linnemann).

[16] We do not assume availability of helicopter med-evac transport for this purpose, given the testimony on such availability which the Board relies on in its findings on Contention VIII-12(b).

b. Majority Findings by Judges Cole and Morris

E-72. While the Commission's decision in *San Onofre* is directed primarily to consideration of offsite emergency response plans, important guidance is given that is relevant here. In discussion of § 50.47(b)(12), the Commission teaches that:

> The emphasis is on *prudent* risk reduction measures. The regulation does not require dedication of resources to handle every possible accident that can be imagined. The concept of the regulation is that there should be core planning with sufficient planning flexibility to develop a reasonable *ad hoc* response to those very serious low probability accidents which could affect the general public. (Emphasis in original.)

San Onofre, supra, 17 NRC at 533. The Commission explicitly noted that NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," and NUREG-0654 were considered in its examination of this regulation. Also, the Commission noted the conclusion of the Appeal Board that "relatively few people [one to 25] are expected to be both contaminated and traumatically injured in a nuclear accident." *Id.* at 532. *See Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), ALAB-680, 16 NRC 127, 137 (1982). *See also* Tr. 9806 (Linnemann).

E-73. Regarding the availability of other hospitals in the highly unlikely event that Pottstown Memorial is evacuated, the County Radiological Emergency Response Plans (RERPs) show that there are twenty hospitals in the three county risk areas listed with radiation exposure/contamination treatment capability (Montgomery County-12, Berks County-3, Chester County-5). While the Board has no detailed knowledge of the specific abilities and training of the emergency medical service personnel at these potential alternative receiving hospitals, who might handle "contaminated injured," it is not unreasonable to assume that they are adequately prepared. Also, when a contaminated injured individual is transported, a health physicist would accompany him and provide assistance in controlling any radiological hazard both during transport and at the receiving facility. Tr. 9842-43 (Boyer). In the event of a large number of casualties, it is not unreasonable to assume that other hospitals and trained personnel, including particularly University of Pennsylvania and RMC specialists, will provide direct assistance. It may also be reasonably assumed that in the event of a hospital evacuation, trained personnel and some equipment would travel to the receiving hospital and provide assistance.

E-74. While the Board majority agrees that it would be prudent to make more formal arrangements with a third hospital, one less vulnerable to evacuation than Pottstown Memorial, and more accessible (closer) than the University of Pennsylvania, we decline to require such an arrangement. It is our view that the probability of Pottstown Memorial being unavailable is remote, that there are nineteen other hospitals in the three-county area with claimed capability for handling "contaminated injured" on an *ad hoc* basis in an emergency and the Pottstown Memorial Staff, RMC and University of Pennsylvania specialists can provide assistance to each other and other participating entities during an emergency. We also note that for the most severe emergency action level (a General Emergency), evacuation is not automatically recommended; sheltering is the first option and may be the preferred action. NUREG-0654, Appendix 1, at 1-16. These considerations militate against imposing any additional requirements. Applicant has met the requirements of Planning Standard (b)(12) in 10 C.F.R. § 50.47.

c. Partial Dissent of Judge Brenner

E-75. I respectfully disagree with my colleagues that there is no need for the emergency plans to include arrangements for the treatment of contaminated injured persons at a backup hospital to Pottstown Memorial which is closer than the Hospital of the University of Pennsylvania (HUP), in the event Pottstown Memorial has to be evacuated due to an accident at the Limerick facility. As noted above, Pottstown Memorial is located within the plume exposure EPZ less than 2 miles from the Limerick nuclear plant.

E-76. I readily grant that evacuation of Pottstown Memorial is improbable, perhaps even less probable than the evacuation of the area around it, for, as the Applicant's witness says, evacuation of a hospital is a serious matter. Tr. 9941 (Linnemann). Nonetheless, the possibility, remote though it is, of life-threatening releases from nuclear power plants is assumed by the NRC's regulations and guidance on emergency planning. Thus, the regulations and guidance envision the possibility of evacuation of an area up to about 10 miles in radius. Planning for medical care for even a small number of contaminated injured persons up to about twenty-five (per *San Onofre, supra,* ALAB-680, 16 NRC at 137 and CLI-83-10, 17 NRC at 532) should be consistent with this possibility.

E-77. Thus, the main issue under this contention becomes whether there are adequate arrangements for the care of the contaminated injured in a radiological emergency which requires the evacuation of Pottstown

Memorial. I think there are not. As the Applicant itself says, HUP can provide backup for Pottstown Memorial only when the trauma victim can withstand the delay caused by going to HUP. *See* Tr. 9906-07 (Linnemann).[17] Moreover, although JCHA accreditation may guarantee that any of the hospitals between HUP and Pottstown Memorial would accept contaminated injured victims, there is no reasonable assurance, due to the total absence of planning, that any of those hospitals is well prepared to treat such victims, especially if there were to be more than one or two victims. If JCHA accreditation were sufficient to guarantee adequate care for the contaminated injured, there would be no need to provide Pottstown Memorial with special training and equipment.

E-78. Even the Applicant's chief witness, whom I found to be knowledgeable and forthright, agrees that it would be prudent to have at least skeletal arrangements with a hospital between PMMC and HUP. Tr. 9914-15 (Linnemann). Even this has not been done. Moreover, I think that prudence suggests more than merely skeletal arrangements with a third hospital. I therefore conclude that the Applicant should assure that there is an emergency backup to Pottstown Memorial in addition to, but closer than the large resources available at HUP. I note that my view is consistent with the uncontradicted testimony of the Applicant and Staff, and the views of all parties, that it is prudent and proper medical practice that a hospital being relied upon for treatment of traumatic injury, contaminated or not, be reasonably close (accessible) to the plant. *See* Finding E-69, above.

E-79. Accordingly, I would have required, as a condition for the full power operation of Limerick, that the Applicant make arrangements with an additional hospital in the Limerick area, similar to the ones it has made with Pottstown Memorial for the care of the contaminated injured, e.g., similar arrangements for training, equipment, and NRC/FEMA-reviewed drills and exercises. Other than the obvious, namely that the third hospital should be less vulnerable to evacuation, and significantly more accessible than HUP, I can set out no simple rule for choosing this third hospital. It is not even required that the third hospital be outside the plume EPZ. Much depends on what hospitals the Applicant has to choose from, how accessible each is, and no doubt other factors which, on the record before us, I am in no position at this time to judge. As the majority notes, there are many candidate hospitals from which the Applicant could easily choose a satisfactory one with which to

[17] As noted above, and discussed under LEA Contention VIII-12(b), helicopter availability cannot be relied upon for med-evac purposes given the arrangements made by the Applicant.

engage in such planning. I would have further directed the parties to discuss such arrangements after they were proposed, and advise the Board whether any important material issues remained in dispute. There would be no reason to require such further arrangements prior to issuance of a low power operating license, since the concern over emergencies which may cause offsite consequences and necessitate evacuation does not arise for power levels up to 5%. *See* 10 C.F.R. § 50.47(d).

E-80. In conclusion, I note that I believe it appropriate for decisionmakers to put themselves in the place of one of the potentially affected persons — in this instance a contaminated injured worker at the Limerick Generating Station — when deciding whether proper and required emergency planning is being accomplished. In this instance, I believe proper and required emergency planning is not being accomplished, but readily could be by a utility presumably concerned for its nuclear power plant employees.

10. LEA Contention VIII-12(b): Adequacy of Transportation for the Contaminated Injured

E-81. This is yet another contention on the adequacy of the Applicant's arrangements with Goodwill Ambulance Unit. *See* our discussions of Contentions VIII-6(a) and VIII-10(a). Evaluation Criterion L.4 of NUREG-0654, ch. II says, "[e]ach organization shall arrange for transporting victims of radiological accidents to medical support facilities." LEA contends that the Applicant's arrangements with Goodwill Ambulance do not assure adequate transportation from the plant site for those who are both traumatically injured and contaminated, and that the Applicant has not arranged for any adequate backup for Goodwill. We find that the arrangements with Goodwill are adequate for possible onsite needs, but that the possibility of competing offsite uses for the ambulances will have to be considered during the review of the offsite plans.

E-82. Goodwill has five ambulances. Tr. 9847 (Kankus). Each is designed to carry two and could carry more in an emergency. Boyer *et al.*, ff. Tr. 9772, at 10-11. Thus, if in an emergency Goodwill's only responsibility was to transport contaminated injured persons from the plant site, there could be little question that the arrangements with Goodwill were adequate. The person responsible for establishing the Applicant's emergency medical program testified that, during his 15 years of experience in establishing similar programs at about twenty-five nuclear power plants, there had never been at any one time more than two contaminated injured victims who required transportation to a local hospital (Tr. 9806 (Linnemann)), and that it was reasonable to expect the same number in the future, since not even a melted core would increase the number of traumatic, nonradiation, injuries (Tr. 9806-07 (Linnemann)). Goodwill's five ambulances clearly could deal with a much larger number of contaminated injured than the one or two expected.

E-83. However, Goodwill may also have offsite responsibilities. One of the Applicant's witnesses testified that current drafts of the offsite plans assign to Goodwill some responsibility for providing special assistance to persons in various townships — twenty-four persons in Pottstown Township alone. Tr. 9936 (Kankus). The letter of agreement with Goodwill shows that Goodwill has agreed to furnish transportation for contaminated injured site personnel only "within the limits of [its] resources." Plan, Appendix A. The Applicant claims that it "would expect its call [to Goodwill] to take priority over another request, which would be assigned to one of the backup ambulances at the county level" (Tr. 9848-49 (Boyer)), but we have nothing more than the Applicant's expectation to support a finding that Goodwill would give priority to onsite needs. Thus, if the current offsite pla provision concerning Goodwill becomes final, it is possible that in an emergency Goodwill's offsite responsibilities would keep it from its onsite responsibilities.

E-84. Moreover, it appears that in such a situation the Applicant would be able to find only limited substitutes for Goodwill's services. Goodwill is the only ambulance company with which the Applicant has an agreement for the transportation of the contaminated injured. At the time of the hearing in April 1984, the Applicant was negotiating an agreement with a second company and expected to complete the agreement within a week (Tr. 9872-73 (Kankus)); but, apparently, even now, the agreement is not complete. The Applicant claims that there would be adequate backup ambulances at the county level, since if all of Goodwill's ambulances were occupied, "the Goodwill dispatcher would notify the county immediately and arrange for another ambulance to be dispatched for Limerick." Tr. 9937 (Boyer). It is not clear that this account is consistent with the Applicant's claim, noted in the preceding paragraph, that Goodwill would give priority to requests from Limerick. At any rate, we have too little evidence about the county dispatching system to conclude that in an emergency, backup ambulances would be available if Goodwill were not.

E-85. The Applicant also claims that private vehicles on site would be available for transporting the contaminated injured, but the Applicant also notes that such vehicles could transport only those whose injuries did not require them to be transported in an ambulance. Boyer *et al.*, ff. Tr. 9772, at 11.

E-86. Finally, a helicopter could also be used to transport the injured. The Applicant has an agreement with Keystone Helicopter which includes medical evacuation among the services Keystone is to be ready to provide. *See* App. Ex. 41, ¶ 1. However, for the same reason that HUP would be of limited use for treating the contaminated injured, Keystone would be of limited use for transporting them. As was noted in our discussion of LEA Contention VIII-12(a), HUP is a 45-minute drive from Limerick. Keystone has agreed to provide a helicopter on 2 hours notice, if one is available, or 1 hour, if Radiation Management Corporation, who entered into the agreement with Keystone on the Applicant's behalf, pays to have a helicopter on 24-hour standby. App. Ex. 41, ¶¶ 4-5. The treatment of some traumatic injuries probably should not be put off for 45 minutes to 2 hours.

E-87. Thus, for transportation of the contaminated injured, the Applicant has to rely mainly on Goodwill. Yet Goodwill may have competing duties off site. However, a determination by us about whether Goodwill could perform all the duties which the plans may finally assign it would be premature. To make such a determination, we would have to judge on the basis of speculation about the final state of the offsite plans. We think it preferable for us to judge on the basis of what we know: Considered apart from the final version of the offsite plans, the Applicant's agreement with Goodwill is adequate for onsite needs. Whether Goodwill can perform both its onsite duties and whatever offsite ones it may be assigned will be best determined at the time for consideration of the offsite plans, whether it be in a hearing as an issue in controversy or by authorities reviewing the offsite plans, for it will then be ascertainable on the basis of the final versions of both onsite and offsite plans.

11. LEA Contention VIII-14(c): Calculating and Monitoring Offsite Doses

E-88. The first part of this contention alleges a deficiency in the Applicant's way of calculating potential offsite doses. The second part alleges a deficiency in the Applicant's way of monitoring actual offsite doses. We rule against LEA on both parts.

E-89. The first part of the contention relies on a contention we have already ruled against. LEA alleges that both the Applicant's computerized dose projection system — the Radiological and Meteorological System (RMMS) — and its manual backup system are deficient because some of the meteorological data they rely on come from a monitoring station, the Applicant's Met-Tower 1, whose proximity to the cooling towers can cause distortions in its data. LEA Contention VIII-3 was based on the Staff's continuing concern with the impact on emergency planning of Met-Tower 1's location. In our discussion of VIII-3, we ruled that since the state of the record put us in the position of merely reviewing the Staff's work, rather than adjudicating competing claims on which the Staff's work had bearing, the Staff, not the Board, was the proper body to determine whether data from Met-Tower 1 could be relied on in an emergency. Thus, we are not in a position to find that the RMMS and its manual backup are deficient because they rely on data from Met-Tower 1.

E-90. The second part of the contention misunderstands the purpose of the monitoring system it alleges is deficient. The system consists of forty-eight thermoluminescent dosimeter (TLD) stations, forty of which are arranged in two rings. The other eight are variously located, but three of them are located where atmospheric dispersion analysis indicates that annual concentrations of radioactive releases to the air are likely to be the greatest. Tr. 10,202, 10,204 (Daebeler). None of the forty-eight TLD stations is more than 5.5 miles from the plant site. Tr. 10,202 (Daebeler). The Applicant claims that the layout of the system conforms to the guidelines in Regulatory Guide 4.8. Tr. 10,203 (Daebeler).

E-91. LEA argues that the system may underestimate radiation dose in an emergency, because the TLD stations are located so that there is no assurance that any one of them would record the maximum concentration of radioactivity released in an emergency: The three stations which are located to record maximums are meant to record annual maximums only, and in fact do not necessarily record actual annual maximums at all, but only the doses at their locations, which may, or may not, be maximums, depending on the accuracy of the dispersion analysis. Moreover, the maximum dose may occur beyond 5.5 miles, for, although it is, on the average, true that the greater the distance from the plant, the less the concentration, unusual atmospheric conditions can cause greater concentrations at greater distances. *See* Tr. 10,201 (Murphy).

E-92. All that LEA says here is true, but LEA misconstrues the purpose of the TLD array. Its primary purpose is to provide routine monitoring which will determine annual doses to the environment. Tr. 10,208 (Daebeler). Thus, it aims for annual maximums instead of a one-time maximum, and can afford to overlook the occasional high concentration at a great distance, since such a concentration would have little effect on average dispersion patterns.

E-93. Of course, in an emergency, the actual maximum is more important than the average one, but it is also less easy to predict. Thus, it is not possible to post a few monitoring stations to lie in wait for it. The maximum can be caught only by a perhaps imprudently dense and extensive array of stations, or by a few mobile units. The Applicant will rely on field survey teams. Tr. 10,211 (Dubiel).

12. LEA Contention VIII-14(e): Continuing Accident Assessment Capabilities

E-94. In Contention VIII-3, LEA alleged that three of the Applicant's onsite monitoring systems were inadequate for use in initiating emergency measures. Here, in Contention VIII-14(e), LEA alleges that for the reasons set out in the earlier contention, the same systems are also inadequate for use in continuing assessment throughout the course of an accident. In our discussion of the earlier contention, we found no deficiencies in one of the systems and ruled that, given the record, the Staff was the appropriate body to determine whether there were deficiencies in the other two systems. Thus, we cannot make a finding that any of the three systems is inadequate for use in continuing accident assessment.

13. LEA Contention VIII-14(h): Methodologies for Projecting Dose When Instrumentation Is Inoperable

E-95. Evaluation Criterion I.6 of NUREG-0654, ch. II, calls for the Applicant to establish methods of projecting doses when the instrumentation used for assessment is offscale or inoperable. The methods are described in Boyer *et al.*, ff. Tr. 9772, at 23. LEA contends that insofar as the methods rely on meteorological data from Met-Tower 1, whose proximity to the cooling towers can cause distortion in its data (*see* our discussion of Contention VIII-3), the methods are deficient. For the reason below, we rule against LEA.

E-96. Contention VIII-14(c) makes the same argument about the RMMS system and its backup. We ruled against LEA on Contention VIII-14(c) because we had decided earlier that given the state of the record, the Staff was the appropriate body to determine whether the location of Met-Tower 1 could have an adverse impact on emergency response. The same reasoning applies here.

14. LEA Contention VIII-15(b): Monitoring of Site Evacuees

E-97. Evaluation Criterion J.3 in NUREG-0654, ch. II, says, "[e]ach licensee shall provide for radiological monitoring of people evacuated from the site." Though as admitted, this contention raised a number of issues, foremost among them then, and among the two issues LEA now puts before us for decision, is whether the time which might be required to monitor the evacuees for contamination would pose a threat to their health. We conclude that it would not.

E-98. We first describe how the monitoring would take place. Under the Applicant's onsite emergency plan, plant personnel not essential to operation of the plant would evacuate to offsite assembly areas, where any needed decontamination would take place. Implementing procedure document EP-305, Rev. 0 (App. Ex. 33) and Rev. 1, names two possible assembly areas. *Id.* at 3. The direction of the wind would determine which was used. *Id.*

E-99. However, to speed up the process of identifying personnel who needed to be decontaminated, and yet not slow down the evacuation, the Plan calls for evacuees to exit the site through portal monitors. These will sound alarms whenever contaminated persons walk through them. Tr. 10,238 (Dubiel). Any person who set off an alarm would be instructed to report to health physics personnel when he arrived at the offsite assembly area. EP-110, Rev. 2, at 5.

E-100. LEA's concern in this contention is about the procedures which would be followed if the portal monitors were not to work. The Applicant says that all evacuees would be monitored at the offsite assembly area unless they had all passed through functioning portal monitors. Tr. 10,227, 10,255 (Dubiel). LEA makes two claims about this alternate procedure. The first is that the Applicant's implementing procedures, which do not say that all site evacuees would be monitored at the assembly area, ought to, even though it may be "normal practice in health physics procedures" to monitor all the evacuees. Tr. 10,228 (Dubiel). The issue raised in this claim has been made moot by yet another revision of the implementing procedures which apparently has escaped the notice of the parties. *See* our discussion of LEA Contention VIII-6(c). EP-254, Rev. 2, in bold letters says that personnel monitoring at the assembly area must be completed before any vehicle monitoring is performed. *Id.* at 4. Sections 9.1.3.8 and 9.2.1.1 speak respectively of monitoring "each individual," and "all personnel." *Id.*

E-101. The second claim LEA makes about the procedures the Applicant would follow if the portal monitors were not to work is that those procedures would take too long. Monitoring at the assembly areas would have to be done with hand-held survey instruments which require up to

2 minutes to monitor one person. Tr. 10,267-68 (Dubiel). LEA claims that the Applicant's procedures provide only one or two technicians to perform this monitoring at the offsite assembly areas. LEA PF 122 (*citing* Tr. 10,231 (Dubiel)). Thus, if, as would happen in a worst case, 3000 plant personnel and construction workers evacuated to the offsite assembly area, one technician taking 2 minutes to monitor each of 3000 personnel would take 100 hours to monitor them all. Moreover, each evacuee would have to stay at the assembly area until he had been monitored, even if the Commonwealth had ordered the evacuation of the plume exposure pathway emergency planning zone. Tr. 10,236 (Kankus).

E-102. LEA's figure of 100 hours is highly improbable. Perhaps it should be recalled at this point that the conditional assumption that enough portal monitors would fail, so as to prevent monitoring of all personnel as they leave the site, makes improbable that there would be a need for monitoring at the assembly areas. But there are reasons why 100 hours is especially improbable. First, it is not at all likely that 3000 people would show up at an offsite assembly area. For one thing, there would be 3000 *on* site only at a peak: The day shift of the operating personnel would number about 400 to 500, and the greatest number of construction personnel working on Unit 2 is expected to be about 2500. Tr. 10,230 (Boyer). Whatever number of construction workers there may be on site, they are to be evacuated at the Alert level of emergency response, before site evacuation, and therefore before they can be contaminated. Tr. 10,238 (Dubiel). Thus, they would not be sent to an offsite assembly area for monitoring and decontamination. Of the 400 to 500 operating personnel, LEA, relying on testimony by the Applicant, estimates that 100 or 200 might evacuate, the rest remaining on site as emergency workers. LEA PF 143. According to these probabilities and estimates, one can reasonably predict that only 100 to 200 plant personnel would reassemble off site for monitoring. Thus, LEA's figure of 100 hours is reduced by a factor between 15 and 30.

E-103. That figure can be reduced even further. Section 9.1.2.1 of EP-254, Rev. 2 requires that at least two technicians be sent to the offsite assembly areas to do the monitoring. Two technicians would take 200 minutes to monitor 200 evacuees. Three would take a little over an hour to monitor 100. *Cf.* Tr. 10,262 (Dubiel). The Applicant plans to get some idea of how many technicians would be needed by randomly monitoring evacuees as they exit the site. Tr. 10,257 (Dubiel). The Applicant could, though it would not expect to have to, assemble as many as thirty technicians at an offsite assembly area. Tr. 10,261 (Dubiel).

Finally, we note that choosing the assembly area according to the direction of the wind considerably reduces any health risk posed by holding evacuees at the area until they are monitored.

15. LEA Contentions VIII-15(d) and 16(g): Decontamination of Site Evacuees

E-104. As admitted, VIII-15(d) and VIII-16(g) were distinct contentions which raised a number of issues. LEA now raises a single issue but retains both numbers. LEA alleges that the Applicant should provide for the contingency that offsite decontamination of site evacuees would require showering or bathing facilities. We do not agree.

E-105. As we explained in our discussion of Contention VIII-15(b), site evacuees would be monitored for contamination either at a site exit point or at an offsite assembly area. As the Plan now stands, decontamination at the assembly areas would rely on simple methods: removing contaminated clothing, washing exposed areas of the skin with a damp washcloth, and cutting off contaminated parts of the hair. The Applicant claims that showering or bathing, which are available for personnel who remain on site, would be required for site evacuees only if the simple methods failed, and that the simple methods would not be likely to fail, since if the site evacuees encountered any contamination, it would very likely only be contamination of the clothing by the short-lived daughter products of some of the gases that would appear in a plume. Tr. 10,243 (Dubiel).

E-106. LEA says that the Applicant should plan for the contingency that the simple methods would not be enough by arranging for transporting site evacuees who need showers and baths to facilities which have them.

E-107. LEA does not dispute the Applicant's judgment that site evacuees are not likely to have to be decontaminated by showering and bathing. As we have said before in our discussions of the emergency planning contentions (*see, e.g.*, LEA Contention VIII-11), probabilities should be kept in mind, and the lesser of them should receive less attention in planning than the greater, especially when, as here, the more remote possibility is of the sort which, if it comes about, can be dealt with through *ad hoc* arrangements.

16. *LEA Contention VIII-15(e): Applicant's Ability to Account for Personnel*

E-108. Again we must struggle with the implementing procedures. Evaluation Criterion J.5 of NUREG-0654, ch. II, says, "each licensee shall provide for a capability to account for all individuals on site at the time of an emergency and ascertain the names of missing individuals within thirty minutes of the start of an emergency." LEA argues three reasons for concluding that the Applicant's implementing procedures do not conform to this Criterion. None of the three reasons are more than minimally argued, and we find them unpersuasive.

E-109. LEA's first reason is that since EP-110, Rev. 3, the implementing procedure document which covers personnel accountability, does not apply to Bechtel and subcontractor personnel, in particular Unit 2 construction workers (*see id.*, § 1.0), and since the Applicant apparently is not familiar with Bechtel's accountability procedures, the Applicant cannot show that it can account, in the language of Criterion J.5, for "*all* individuals on site" within 30 minutes of the start of an emergency. (Emphasis supplied.)

E-110. The Applicant does not bear the burden of proving the adequacy of Bechtel's procedures, for LEA has proffered no basis for thinking that those procedures might be inadequate in some respects. Such a basis is especially needed here, for, on its face, the division of responsibility between the Applicant and Bechtel makes sense, since one would expect that Bechtel would know more about the deployment of the construction force than would the Applicant, and therefore would be in a better position to devise accountability procedures for that force.

E-111. We note also that the Staff, whose opinion on the interpretation of NUREG-0654 is to be accorded some weight, apparently does not read the "all" in Criterion J.5 to be as inclusive as LEA thinks it is, for the Staff raises no objection to the division of responsibility between the Applicant and Bechtel. *See* Staff PF 81-82. The Evaluation Criteria can be explicit when they want to include construction personnel in their provisions. *See* Criterion J.1.[18]

E-112. The second reason LEA puts forward for concluding that the Applicant does not conform to the 30-minute limit called for in J.5 is

[18] The Applicant's argument against this first reason of LEA's cannot be squared with the text of the implementing procedures. The Applicant argues that construction personnel would be evacuated before accountability procedures would be put into effect. Applicant's Reply Findings at 18. However, the relevant implementing procedure document, according to its own terms, "should" be implemented whenever an Alert or higher response level is declared, and can be implemented even at the Unusual Event level. EP-110, Rev. 2, § 7.0. The same document explicitly calls for informing the Security Team Leader of any unaccounted-for Bechtel personnel. *Id.*, § 9.1.5.1.F. Besides, Bechtel *does* have accountability procedures.

that, according to LEA, the Applicant measures the 30 minutes from too late a moment. EP-110, Rev. 2 measures 30 minutes from the time of the evacuation or assembly announcement (*id.*, § 9.1.5.1.E), not from the "start of an emergency," as J.5 calls for. But LEA argues that an assembly announcement could come as much as an hour after the start of an emergency, because verification of the emergency classification must precede an assembly announcement (*see, e.g.*, EP-103, Rev. 3, at 2, 4), and verification could take up to an hour. Thus, an accounting for the locations of all personnel, if not completed until 30 minutes after an assembly announcement, could come as much as an hour and a half after the start of an emergency.

E-113. This claim that the Applicant measures the 30 minutes from too late a moment has the same form as the claim in LEA Contention VIII-6(c) that the Applicant measures the time to notification of offsite authorities from too late a moment, and it has one of that earlier contention's weaknesses too: The argument that verification could take up to an hour is without basis. *See* our discussion of LEA Contention VIII-6(c). We note also that the Staff speaks of the start of an emergency and the moment assembly is announced as if there were no significant difference between the two times. *See* Staff PF 81-82. We see no basis for assuming a significant difference, if any.

E-114. LEA's third and last reason for concluding that the Applicant cannot conform to the 30-minute limit in J.5 is that, according to LEA, during a site evacuation, there is no assurance that everything which must be accomplished before all personnel are accounted for can be accomplished in 30 minutes. First, the Emergency Director would have to perform not merely verification, but seven tasks before he announced assembly and evacuation. *See* EP-305, Rev. 1, at 2-4. Second, evacuees might have to be randomly monitored if the portal monitors were inoperable as they left the site, and, as we noted in our discussion of Contention VIII-15(b), the instrument which would be used in such random monitoring requires up to 2 minutes for monitoring one person. Third, the Personnel Security Group, using a master list of badge numbers, might have to check off by hand the numbers of all the badges evacuees are to deposit in buckets at the exit points. *See* EP-110, Rev. 2, § 9.1.4.2.D. Fourth, in order to compile a list of unaccounted-for plant personnel, the Personnel Accountability Group would have to compile a similar list of personnel remaining on site and then compare that list with the evacuee list prepared by the Security Group. *Id.*, § 9.1.5.1.C. and D. Fifth and last, before it could compile a list of all those not accounted for — both operating personnel and construction workers — the Accountability Group would have to find out from Bechtel which of

Bechtel's personnel were not accounted for. *Id.*, § 9.1.5.1.F. If the evacuation were to take place during the day shift and at a period in the construction of Unit 2 when the construction force was at its predicted peak, as many as 2700 persons might be evacuating from the site. *See* our discussion of Contention VIII-15(b).

E-115. We think that any appearance of great length LEA's list may have is created largely by the explicitness inherent in implementing procedures, and not by the length of time the tasks in the list would require. The seven tasks which the Emergency Director must perform before he announces assembly and evacuation are simple tasks such as notifications by telephone. *See* EP-305, Rev. 1, at 2-4. The random monitoring of evacuees is random precisely so that monitoring will not interfere with evacuation. Tr. 10,257-58 (Dubiel). Checking off a number on a list does not take long, and the checking would probably begin when the first evacuees passed through an exit point. Finally, though it might require precision drill work to move 2700 people through a single door in 30 minutes, a glance through EP-305, Rev. 1 shows that there would be more than one exit in a site evacuation.

E-116. In its approach to site evacuation, LEA has done little more than say that the Applicant would have a lot to do in 30 minutes. But to make a strong case, LEA would have had to show that, in light of the goals of rapid evacuation, rapid deployment of onsite emergency workers, and exact accounting of personnel, a significant part of what the Applicant was planning to do was unnecessary, or ill-timed, or best replaced. LEA having made no such case, we think it should be left to the emergency preparedness exercises to determine whether the Applicant can evacuate the site and account for all personnel in 30 minutes. *See* Sears, ff. Tr. 9772, at 22.

17. LEA Contention VIII-16(c): Information on Radiation Risks for Emergency Workers

E-117. Originally concerned with all emergency workers who might be on site at some point in an emergency, whether they be employees of the Applicant or not, this contention is now concerned solely with workers who are employees of offsite organizations which would provide support on site. LEA alleges four deficiencies in the information on radiation risks which is given to such workers. We find no such deficiencies.

E-118. The first deficiency LEA alleges is that workers from offsite organizations which would provide support on site are not given information about the acute effects of high doses of radiation. It is true that they are not. Tr. 10,024 (Dubiel). The reason is simply that their tasks on site will not expose them to high levels of radiation. Tr. 10,048 (Dubiel). Table 6-1 of the Plan sets out dose limits no emergency worker would be allowed to exceed without specific authorization from the Emergency Director. Such authorization would be given only to those who had the appropriate training. Tr. 10,056 (Dubiel). But that particular training is available only to employees of the Applicant. *Id.* Therefore, no employee of an offsite support organization would be given permission to exceed those limits. *Id.* We note that such workers are told a great deal about the risks posed by the radiation levels they would encounter, including the increased probability of injury, illness, or death due to radiation, the latent effects, including genetic, of low levels of radiation, and even the risks posed by doses which are below regulatory levels. *See* Tr. 10,019-29 (Dubiel). Such information should be enough to enable these workers to make sober, informed decisions.

E-119. The second deficiency LEA alleges is that although the Applicant's witness on this subject testified that the minimum training program for these workers required that the information in Regulatory Guide 8.13 be presented them, the witness was so vague as to make it impossible to determine just what information will be provided. To support the allegation, LEA claims that the witness "could not testify whether particular information actually in Reg. Guide 8.13 [was] specifically presented." LEA PF 151 (*citing* Tr. 10,036-38 (Dubiel)).

E-120. LEA misconstrues the witness' response. The "particular information" LEA refers to was the information in Regulatory Guide 8.13 on the risks radiation poses to pregnant women. The Applicant's witness could not say how detailed the coverage of that information might be without knowing the composition of the group to which it was being presented. Only if the group contained women, would the presentation of the information on the risks for pregnant women be detailed. Tr. 10,037 (Dubiel). We do not find this response vague, but rather, pedagogically sensible, since it shows that trainers will be emphasizing for each group what it most needs to know. The same pedagogy appears to be behind the emphasis in the training of these workers on the effects of low-level radiation.

E-121. The third deficiency LEA alleges, and alleges as the most "disturbing" (LEA PF 152), is that the U.S. EPA Protective Action Guides (PAGs) are not explained to these workers. LEA PF 152 (*citing* 10,041 (Dubiel)). Thus, LEA alleges, "the workers will not know when 'permissible' doses are exceeded." *Id.*

E-122. LEA's allegation is factually incorrect. What the testimony LEA cites says is that the workers in question will not be informed about the PAGs *specifically*. Tr. 10,041 (Dubiel). They will, however, be informed about them indirectly, for they will be informed about the dose limits under which they would operate, and these limits, set out in Table 6-1 of App. Ex. 32 (Plan), are consistent with the PAGs. Evaluation Criterion K.1 of NUREG-0654 requires the Applicant to establish such guidelines. Thus, the workers would have a standard by which to judge whether they had exceeded regulatory doses.

E-123. The last deficiency LEA alleges is that for such workers, there are no methods of determining whether the worker has comprehended the training. LEA PF 153 (*citing* Tr. 10,052 (Dubiel)).

E-124. The cited testimony is in fact not so broad. The witness said that there was no formal examination required of fire department personnel. *Id.* The testimony does not preclude more informal ways sensible people teaching and studying about risks to their health may have for assuring that what is being taught is being learned. We note that the Evaluation Criteria in § 0 of NUREG-0654, ch. II, set out with specificity means the Applicant is to use to assure that *on*site personnel are properly trained (*see* Criterion 0.2) but the same criteria say nothing similar about the training for the workers which are the object of this contention. LEA has not tried to argue that those workers should be trained to the depth onsite ones are. Nor do we see any basis for such a viewpoint.

18. LEA Contention VIII-18: Training of Offsite Support Personnel

E-125. Here LEA alleges that the deficiencies which Contention VIII-16(c) alleges exist in the program for informing offsite personnel about radiation risks show that the Applicant has not met the requirement in Planning Standard (b)(15) in 10 C.F.R. § 50.47 that adequate training be given those who may be called on to assist in an emergency. We did not agree that there were deficiencies in the program, and therefore rule against LEA on this last contention.

F. NEPA Severe Accident Risk Contentions: LEA Contentions DES-1, 2, 3 and 4

1. Summary

F-1. LEA's four contentions considered in this section allege that the risks of severe accidents have not been considered properly under the National Environmental Policy Act (NEPA). The first contention discussed, DES-4, argues that the NRC Staff's Final Environmental Statement (FES) (which superseded the draft statement (DES) to which the contentions were originally directed) fails to adequately disclose or consider certain nonfatal latent health effects, the interdiction (denial of consumption or access) of cropland, milk and the population in such land areas, and the cost of medical treatment. Part B of this contention alleges that the FES format obscures the estimated total impact of severe accidents at Limerick. In general, the Board finds that it would have been helpful to lay members of the public if the FES had contained more complete disclosure and explicit consideration of the matters set forth in LEA's Contention DES-4A. However, we also find that the conclusions of the FES as to total risk are unchanged by the explicit consideration now provided by the evidence and decision in this case. The Board also finds that the FES did emphasize the dominant contributors to total risk and did disclose the means by which a professional could estimate the other forms of risk (although in some cases this would have required resort to extensive references). Therefore, no further relief is required on the merits of the contentions. We find part B of the contention to be vague as litigated, and in any event we find the format of the FES adequate and proper given the state of the art of severe accident risk assessments.

F-2. LEA Contentions DES-3, 1 and 2 are discussed in that order after DES-4. They involve allegations that certain assumptions made about evacuation actions in the estimates of severe accident risks are not valid, i.e., that people will obey instructions to evacuate (DES-3), that people in certain areas beyond a 10-mile-radius zone can be relocated (DES-1), and that there will be only about a 2-hour delay from the time of the accident before people begin to evacuate (DES-2). As to each of these, the Board finds that the actual assumptions made in the severe accident analyses are not unreasonable. The Board also finds that, in any event, notwithstanding the large uncertainties in the way actual emergency actions would occur, sensitivity estimates of the effect of reasonable changes in the evacuation assumptions show the lack of significant effect of such changes on the risk estimates.

F-3. In a separate section after the decision on LEA's severe accident risk contentions, the Board explains why it rejects both LEA's and the City of Philadelphia's conclusions of law as applied to the severe accident risk contentions.

2. *LEA-DES-4*

F-4. This contention, as admitted, states:

A. The DES Supplement fails to adequately disclose or consider:

1. Total latent health effects due to both initial and chronic radiation exposure, other than those resulting in fatalities, including genetic effects, nonfatal cancers, spontaneous abortions, and sterility (*see, e.g.*, BEIR I-III);

2. The total land area in which crops will be interdicted;

3. The total land area in which milk will be interdicted;

4. The quantification of the cost of medical treatment of health effects.

5. The population within the land areas to be interdicted.

B. By treating some environmental costs in a CCDF format and treating other quantifiable costs in a nonquantitative, subjective manner, the DES format obscures the total impact of severe accidents at Limerick.

F-5. Both parts of this contention are directed to alleged deficiencies in the Supplement to the Draft Environmental Statement (DES) prepared (as required by NEPA) by the Staff. This document, NUREG-0974, Supplement No. 1, was issued in December 1983. The Final Environmental Statement, NUREG-0974, was issued by the Staff in April 1984. Staff Ex. 29. Both the Staff and Applicant presented testimony on this contention, LEA did not.

F-6. LEA would have us find that the Staff's Final Environmental Statement (FES) does not comply with the National Environmental Policy Act of 1969 (NEPA), with respect to the risk of severe accidents at the Limerick facility, largely due to alleged numerous material nondisclosures of environmental impacts, including health effects. LEA Proposed Findings (PF) at 1 (July 26, 1984). Moreover, LEA believes that any disclosure defects in the FES cannot be cured by discussion of such defects in this decision. In its view, publication of the decision is no substitute for the full circulation and comment requirements of NEPA and 40 C.F.R. Parts 1502 and 1503. *Id.* With respect to the alleged deficiencies, we discuss them in the context of the individual contentions. With respect to the disclosure and public comment matter, we note the following. Even though an FES may be inadequate in certain respects, ultimate NEPA judgments with respect to any facility are to be made *on the basis of the entire record* before the adjudicatory tribunal. *Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-262, 1 NRC 163, 197 n.54 (1975) (emphasis added). *See also Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 NRC 14, 39 (1979). Since findings of the licensing tribunal are deemed to amend the FES, amendment and recirculation of the FES is not *ipso facto* necessary where findings of a licensing board differ from those of the FES, particularly where the hearing will provide the public ventilation that recirculation of an amended FES would otherwise provide. *Limerick,* ALAB-262, *supra,* 1 NRC at 197 n.54. Thus, modification of the FES by Staff testimony or the licensing board's decision does not normally require recirculation of the FES, *Niagara Mohawk Power Corp.* (Nine Mile Point Nuclear Station, Unit 2), ALAB-264, 1 NRC 347, 371-72 (1975), unless the modifications are truly substantial. *Allied-General Nuclear Services* (Barnwell Nuclear Fuel Plant Separations Facility), ALAB-296, 2 NRC 671, 680 (1975). As we find below, the basic conclusions of the FES are unchanged by our findings. The modifications to the FES made by the record and decision in this case create no reason to recirculate the FES for further comments.

F-7. Two Courts of Appeals have approved the Commission's rule that the FES is deemed modified by subsequent NRC (AEC) administrative adjudications. *Citizens for Safe Power v. NRC,* 524 F.2d 1291, 1294 n.5 (D.C. Cir. 1975); *Ecology Action v. AEC,* 492 F.2d 998, 1001-02 (2d Cir. 1974). *See also Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-78-1, 7 NRC 1, 29 n.43 (1978).

F-8. More recently, the NRC has adopted an amendment to 10 C.F.R. Part 51, Licensing and Regulatory Policy and Procedures for Environmental Protection, which provides that

> [w]hen a hearing is held on the proposed action under the regulations in Subpart G of Part 2 of this Chapter or when the action can only be taken by the Commissioners acting as a collegial body, the initial decision of the presiding officer or the final decision of the Atomic Safety and Licensing Appeal Board or the final decision of the Commissioners acting as a collegial body will constitute the record of decision.

10 C.F.R. § 51.102(c).

F-9. A second general complaint of LEA is that the FES discusses the environmental impact of severe accidents in terms of the risk of one reactor operating for 1 year rather than two reactors operating for the lifetimes of the reactors. LEA could not conclude that the lay reader would discern without instructions in the FES, that the total risk over the operating life of the entire facility could be obtained by multiplication. LEA PF at 2-3. We need not speculate on what the lay reader might discern from the FES. The record is clear that the risk of both units is essentially double the risk from one unit. Tr. 11,194-96 (Acharya). Contrary to LEA's conclusion, one Staff witness did not reject this approach until corrected, but was somewhat ambiguous in maintaining th position that the risks from the two reactors would not be identical agreed that the accident frequencies at Limerick 1 would be app ately equal to the frequencies at Unit 2, but explained that the accide initiators would be different at the two units. Tr. 11,194-95 (Hulman). In any event, the importance of the units used for expressing risk is in the consistency with which comparisons are made. Tr. 11,456 (Levine). Thus, to compare the risks of the Limerick Station over its lifetime, one should compare the risks of the reactor(s) when operating with the risks to which the public is otherwise exposed during such reactor operation.

a. Latent Health Effects (DES-4A-1)

F-10. The Staff asserts that the FES does disclose and consider total latent health effects in that it has assumed a dose-effect relationship for projection of radiation-induced genetic effects; i.e., it has assumed 2.6×10^{-4} genetic effects cases per person-rem. Hulman and Acharya, ff. Tr. 11,148, at 5. This value is equal to the sum of the geometric means of all forms of genetic effects and the risk of effects with complex etiology, and is consistent with values given in the BEIR I (1972),[19] WASH-1400,[20] and BEIR III (1980)[21] reports. *Id.* at 5-6.

F-11. Using the Staff estimate for the risk of total population exposure from Limerick accidents and the risk estimator for genetic effects, one can obtain the estimated risk of genetic effects as

$$1000 \text{ person-rem/reactor-year} \times 2.6 \times 10^{-4} = 0.26 \text{ case of genetic effects/reactor-year}$$

A complementary cumulative distribution function (CCDF) curve for genetic effects can be obtained from the CCDF[22] for total person-rem (Figure 5.4c of the FES) by multiplying the consequence magnitudes (on the *x*-axis) by 2.6×10^{-4}. *Id.* at 6.

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[19] National Academy of Sciences/National Research Council, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Committee on the Biological Effects of Ionizing Radiations (BEIR I), November 1972.

[20] NUREG-75-014, "Reactor Safety Study — An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1975.

[21] National Academy of Sciences/National Research Council. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation," Committee on the Biological Effects of Ionizing Radiations (BEIR III), July 1980.

[22] In probabilistic risk assessments for nuclear plants, CCDF curves usually display in a log-log plot the probability per reactor-year of exceeding a certain consequence versus the magnitude of that consequence (e.g., number of early fatalities).

F-12. The Staff did admit that the risks of certain consequences of accidents at Limerick were not explicitly listed or displayed in the FES. These included genetic effects, spontaneous abortions, and sterility. Tr. 11,200-01 (Acharya, Hulman). The Staff asserts, however, as follows: The fact that genetic effects are not shown (explicitly) does not mean that the Staff did not allude to or make a statement that genetic effects could be a consequence from the reactor accidents, since it is stated that the genetic effect can be scaled from the population exposure and the population exposure and the conversion factor are given. Tr. 11,200 (Acharya). The (risk of) spontaneous abortions is not in the FES, but it is stated in the FES that such effects can be scaled from the population exposure. Most of the health consequences that were considered important are included. Tr. 11,201 (Acharya). Some of the ones . . . not mentioned, such as spontaneous abortion or sterility, . . . (the Staff) would have estimates for but they were not considered as important as those discussed in the FES. The Staff noted that sterility would be temporary and that spontaneous abortions would occur among a large number of normally occurring spontaneous abortions. Staff-referenced documents, principally WASH-1400, were stated to indeed contain the various other types of health consequences. Tr. 11,203-04 (Acharya). The Staff believes there are so many different categories of consequences and so many different probabilities, it tried to strike a balance in the FES, providing as much information as it thought important to the assessment. It did not provide it all. Tr. 11,205 (Hulman).

F-13. The Staff also agreed that the dose-effect relationship for genetic effects (2.6×10^{-4}) could be 4 to 5 times greater and still be consistent with the range of values given in the BEIR I, WASH-1400, and BEIR III reports. Tr. 11,212-13 (Acharya). Constructing a CCDF curve for genetic effects from the CCDF curve for total population exposure would not indicate that the curve might be 4 to 5 times too low, but the statement of the range of uncertainty would say so. Tr. 11,216 (Acharya).

F-14. With respect to the risk from genetic effects, 0.26 case per reactor-year, it is in fact (numerically) greater than any other health effect analyzed (listed in Table 5.11h) in the FES. Tr. 11,211-12 (Acharya). With respect to nonfatal cancers, the Staff agreed that this risk also is (numerically) greater than any other health effect analyzed in the FES and is the highest risk. Tr. 11,248 (Hulman). The Staff agrees that if a reader knew nothing more than what is explicit in the FES he wouldn't know that there is a risk of benign thyroid nodules, but that, indirectly, the references to the FES provide that level of information. The Staff believes the informed reader of the FES should also consult the references. Tr. 11,250 (Hulman). The Staff recognizes that the state of the art for the precise quantification of the uncertainty (in its risk calculations) is not well developed. Tr. 11,286 (Acharya). The uncertainty assessment is based on three components, probability, source term, and consequences. Tr. 11,290 (Hulman). Thus, its risk estimate could be too low by a factor of 40 or too high by a factor of 400. Tr. 11,286 (Acharya).

F-15. Spontaneous abortions in women exposed to radiation is a possible risk of severe accidents at Limerick, but this risk was not included in the risk estimator for genetic effects. Tr. 11,252 (Acharya, Hulman). The Staff explained that the majority (whether 90% or just more than 50%) of spontaneous abortions would lead to loss of fetus during the first trimester. Genetic effects in live births are included in the Staff risk estimator for genetic effects in succeeding generations. Spontaneous abortion is estimated as 58% of the total genetic effects. Tr. 11,253 (Acharya). The Staff's estimate per reactor-year of spontaneous abortions is 0.15, which is higher than any health effect risk estimated in (Table 5.11h of) the FES, but less than the estimated risk (0.26 per reactor-year) of genetic effects based on live births. Tr. 11,258 (Acharya).

F-16. With respect to temporary sterility for males, the Staff estimate is 0.16 per reactor-year (0.03 for females), which also is higher than any health effect risk estimated in (Table 5.11h of) the FES. Tr. 11,261 (Branagan). The estimated risk from genetic effects is higher than this, however. Tr. 11,261 (Acharya). No cases of permanent sterility would be expected, because doses necessary to induce permanent sterility would be accompanied by lethal doses to other organs. Temporary sterility is less serious than other early radiation illnesses. Hulman and Acharya, ff. Tr. 11,148, at 10.

F-17. The risk with respect to benign thyroid modules is 15 times higher than that of thyroid cancer fatalities. (Tr. 11,261 (Acharya).) Thus, this risk (0.15 per reactor-year) also is higher than any other listed in (Table 5.11h of) the FES. Tr. 11,262 (Hulman).

F-18. Hypothyroidism — a decrease in activity of the thyroid — is a possible consequence of irradiation. Medical treatment, administration of thyroid hormones or removal of the thyroid, would not impair the activity of a person in a measured way. Tr. 11,262 (Branagan, Acharya).

F-19. In addition to the health effects considered in the FES and in addition to benign thyroid nodules and hypothyroidism, other forms of health consequences not already accounted for in the FES or in this contention could be the early fatality dose to the exposure of the embryo and *in utero* exposures. The early fatality of such exposure could be within 5 to 10% of the early fatalities already reported. Also, there could be an early health effect due to excessive exposure of the thyroid organ, called thyroid ablation, in which case the thyroid could be destroyed. The number of such is very small compared to early fatality. Tr. 11,263 (Acharya).

F-20. With respect to impairment of or defects in the development of children due to *in utero* exposure of embryos and fetuses — e.g., microcephaly, mental retardation, growth retardation, blindness, cleft palate, spina bifida — the Staff did not explicitly calculate their risks. The Staff believes, however, that the bases for its estimates of early injuries are more conservative than the WASH-1400 basis and therefore provide a bounding calculation, including all other small impairment risks. Tr. 11,264-72 (Acharya, Hulman, Branagan). The Staff did not think that all of the health impacts that could be associated with reactor accidents were not important, but it did not feel that it was necessary to describe, in great detail, every single one of them in the FES. It thought that what it did was an adequate representation of and the more important types (of impacts). Tr. 11,274 (Hulman). The Staff could have listed the health effects not considered explicitly in the FES, and stated that they were subsumed by the other effects that were analyzed in some detail. However, that would not have changed any of the numbers in the CCDFs or the table expressions of risk that are present in the FES. Tr. 11,282 (Hulman). In its final judgment on whether the risks were low, the Staff did consider the health effects explicitly neglected and also did consider the fact that the risks from the neglected effects were a small percentage of the kinds of risk that were described. Tr. 11,281 (Hulman).

F-21. For perspective, the Staff compared the calculated risk of genetic effects resulting from severe accidents at Limerick to the natural incidence of genetic effects. The accident risk to the first generation of descendants of people irradiated was 0.05 genetic effects per reactor-year of operation. For a population of 8.1 million people, and a natural incidence fraction of approximately 11%, approximately 880,000 genetic effects would occur in the first generation of descendants Tr. 11,278 (Branagan).

F-22. As stated earlier, the specific section of this contention that we are discussing, DES-4A-1, is limited to the adequacy of the Staff's FES with respect to disclosure and discussion of total nonfatal latent health effects resulting from severe accidents at Limerick. The Applicant, however, also submitted testimony on this matter which we find helpful in reaching our conclusion. Although the public impacts presented in the FES are somewhat higher than those presented in the Applicant's Severe Accident Risk Assessment (SARA) report, the differences are within the range of uncertainties of such analyses. Daebeler *et al.*, ff. Tr. 11,114, at 1. *See also* Tr. 11,458-59 (Hulman, Levine). Thus, the Applicant agrees that potential accident risks from Limerick are expected to be a small fraction of the risks the general public incurs from other sources. Daebeler *et al.*, ff. Tr. 11,114, at 1.

F-23. The Applicant notes that, except for cancer fatalities, latent health effects (including nonfatal concerns, genetic effects, spontaneous abortions and temporary or permanent sterility) are generally not included in the numerical results of risk assessments, but that they can be estimated from available information. Tr. 11,329-31 (Levine). The Applicant's estimates of the public risk of latent health effects may be summarized as follows:

> Latent cancer fatalities excluding thyroid cancers — 0.033 per reactor-year.
>
> Thyroid cancer fatalities — 0.0064 per reactor-year.
>
> Total cancer fatalities — 0.04 per reactor-year.
>
> (Applicant estimates, for comparison, the expected number of cancer fatalities per year from all causes in the population around Limerick out to 50 miles to be approximately 20,000 per year.)
>
> Nonfatal latent cancers (including thyroid cancers) — 0.091 per reactor-year.
>
> Genetic defects in the population surrounding Limerick — 0.13 per year (compared to 6000 per year from other causes, in the population out to 50 miles). Using the most recent genetic risk estimator (i.e., dose conversion factor) of 45 per 150 million man-rem, the equilibrium damage (i.e., steady-state rate of occurrence) was calculated to be 0.067 per reactor-year.
>
> Spontaneous abortions are estimated to be on the order of 33 to 76% of total genetic effects for live births (i.e., less than 0.10 per year).
>
> Sterility consequence effects are viewed as subordinate to more serious radiation effects, such as acute fatality or early radiation illnesses. In general, doses either produce temporary sterility, or if large enough, mortality.

Daebeler *et al.*, ff. Tr. 11,114, at 29-34.

F-24. The Applicant, based on its calculations of estimated risks, made some approximate comparisons of risks predicted for Limerick severe accidents and risks to the various population areas around Limerick from all other causes. The individual risks at 1 mile from the reactors of early fatality from Limerick accidents is 10^{-5} of those that already exist from other causes. At 10 miles it is 10^{-7}. For cancer fatality risks within 50 miles of the reactor, the ratio of those predicted from Limerick (accidents) to those which exist within 50 miles to the general population from all (other) causes is 10^{-6}. In the Applicant's view, the (Limerick accident) risks are, in fact, vanishingly small compared to other risks, and are trivial. Further, Applicant believes that to take the worst possible (value for a) parameter or condition in each of the various choices and combining (these to) get a very, very (worst) possible case as a measure of the disclosure of risk to the population would be an irrational procedure. Applicant's witness believed that the chance of all these parameters, be they weather, be they reactor accident scenarios, whatever . . . all happening, in the very worst way, at the same time . . . is an irrational combination. The probabilities of such things happening are even smaller than the vanishingly small probabilities already discussed. Inclusion of factors that might affect these values by (up to) a factor of 2 or 3 is not going to change (the conclusions). Tr. 11,442-45 (Levine).

F-25. With respect to such comparison, the Staff noted that it estimates approximately 700 person-rems per year of operation of "the Limerick reactor." It estimates the natural background radiation that the population receives within 50 miles of the (Limerick) site as 800,000 person-rem per year. The Staff concludes that the ratio 700 to 800,000 (i.e., approximately 10^{-3}) is small. The Staff agrees with the general conclusion of the Applicant. Tr. 11,450-52 (Acharya).

F-26. We turn now to the merits of this specific contention, i.e., whether the FES has failed to disclose or consider adequately the total latent health effects of severe accidents at Limerick.

F-27. The record is clear that not *all* latent health effects of severe accidents at Limerick were explicitly disclosed in the FES. Among those not explicitly disclosed were those identified in the contention, i.e., genetic effects, nonfatal cancers, spontaneous abortions, and sterility, due to both initial and chronic radiation exposure, other than those resulting in fatalities. The reasons the Staff did not include explicit disclosure of these and other latent health effects also are evident. First, the Staff believed that such disclosure was implicit by citing authoritative references which treat these matters in detail, e.g., BEIR I, BEIR III, UNSCEAR, NUREG-75/014 (formerly WASH-1400). Second, the Staff considered that for the purposes of the FES it was not necessary to disclose explicitly those latent health effects that it believed to be relatively unimportant in its best-estimate calculations of the risks of potential reactor accidents at Limerick. This approach, i.e., characterizing reactor accident health risks by reference to early fatalities, latent cancer fatalities and man-rem, although not complete, appears not to be inconsistent with both industry practice and Commission policy. Tr. 11,329-30 (Levine). We do believe an explicit discussion of all the health effects in the DES and FES would better permit the public (as opposed to an informed professional) to understand all factors considered in the risk assessment. We find, however, that the nonfatal latent health effects have been adequately disclosed and considered in this proceeding. This explicit consideration has not changed the basic conclusions of the FES regarding the radiological risk associated with operation of the Limerick Station.

b. Crop, Milk and Population Interdiction (DES-4A-2, 3 and 8)

F-28. The FES does include disclosure and consideration of land interdiction, but land areas for which crops alone, or milk alone would be interdicted (i.e., consumption or access denied), and the population in such land areas, are not explicit. Staff Ex. 29 (FES), at p. 5-93, Fig. 5.4h, Table 5.11g. The Staff described its interdiction model as consisting of four successively increasing areas, based on successively decreasing levels of radionuclide concentration. The first area (most highly contaminated) would require interdiction for more than 30 years. The second area (which would include the first) would require decontamination. The third area (which would include the first two) would require crop impoundment. The fourth area (which would include the first three) would require milk impoundment. Hulman and Acharya, ff. Tr. 11,148, at 12-13 and attached figure. Estimates of the risks of interdiction of the various areas were calculated for the FES analysis using the CRAC (*C*alculation of *R*eactor *A*ccident *C*onsequences) computer program. The CRAC Code was developed for the Reactor Safety Study, WASH-1400 (NUREG-75/014), and generates CCDFs taking into account changing weather conditions and chronic pathways for radionuclides. The results, in terms of square meters per reactor-year interdicted (for the four different levels of contamination), are presented in Table 1 of the Staff's direct testimony. *Id.* at Table 1. The corresponding probability distributions (CCDFs) are defined by values listed in Tables 2 and 3 of that testimony.

F-29. The Applicant notes that both the CRAC and CRAC 2 computer programs are capable of estimating the different areas affected by contamination, and are routinely used to estimate associated costs. Daebeler *et al.*, ff. Tr. 11,114, at 35. The predicted frequency with which areas of various sizes would be contaminated above the levels set for crop interdiction was calculated by the Applicant using CRAC 2 and is shown in Applicant's Table 5. *Id.* at 61. Applicant states that the total land area within which crops are interdicted is generally not explicitly presented because the principal contributor to economic risk is the cost of decontaminating land, and crop interdiction is expected to last (only) 1 year. *Id.* at 38.

F-30. The predicted frequency with which areas of various sizes will be contaminated above the levels set for milk interdiction was calculated by the Applicant using CRAC 2; the results are tabulated in Applicant's Table 6. *Id.* at 38, 63. The time for milk interdiction, i.e., loss in dairy output, is only 2 months. Staff Ex. 29 (FES), at p. 5-106. Applicant finds that interdiction of milk products is not a dominant contributor to economic risks. Daebeler *et al.*, ff. Tr. 11,114, at 39, 59.

F-31. The Applicant also calculated the frequency with which various numbers of people would need to be relocated for long periods of time. Relocation costs also are found to be a relatively small contributor to total economic risk. *Id.* at 39, 59, 63.

F-32. Again, the Board finds that the FES did not *explicitly* disclose and consider the total land area in which crops would be interdicted, the total land area in which milk would be interdicted, or the population within the land areas to be interdicted. Here again, both Staff and Applicant appear to have done the societal risk analyses (in this case the estimation economic impacts) according to general industry and Commission practice, emphasizing the dominant, but not neglecting the lesser, contributions to risk (in some cases more conservatively than realistically). We again find that the FES would have been more helpful to the public (as opposed to the informed professional) had more complete disclosure and explicit consideration been given to the interdiction question. We conclude, however, based on the information provided by the Staff and corroborated by the Applicant in this proceeding that the conclusions of the FES with respect to interdiction are correct.

c. Cost of Medical Treatment (DES-4A-4)

F-33. The cost of medical treatment of health effects was not expressed quantitatively in the FES. Richter, ff. Tr. 11,148, at 6. The FES says only that the Staff has considered the health care costs resulting from hypothetical accidents in a generic model developed by Pacific Northwest Laboratory (Nieves, 1982) and that, based on this generic model, the Staff concluded that such costs may be a fraction of the offsite costs evaluated (in the FES), but that the model is not sufficiently constituted for application to a specific reactor site. Staff Ex. 29 (FES), at p. 5-102.

F-34. Staff witness Brian J. Richter testified that he estimated the health care costs of thirty-seven different accident sequences, as defined in Table 5.11d of the FES, obtaining direct, indirect and total costs. Richter, ff. Tr. 11,148, at 2. Actually, Table 5.11d of the FES lists the mean probabilities of thirty-seven release categories. Staff Ex. 29, at p. 5-77. He then calculated the risk on a reactor-year basis by multiplying the costs times the probabilities per reactor-year of accident sequences (presumably he meant release categories) occurring. Richter, ff. Tr. 11,148, at 2. His results are tabulated in Tables 1, 2 and 3 of his testimony. *Id.*, Attachments 1, 2 and 3. Table 1 lists the three types of costs resulting from twenty release categories initiated by internal causes, fires, and low-to-moderately-severe earthquakes. Table 2 lists the three types of costs resulting from seventeen release categories initiated by severe earthquakes. Table 3 lists the totals for the three types of costs per reactor-year. Direct costs are all costs associated with the treatment of the patient, e.g., physician fees, hospital charges, costs of medicines. Indirect costs are the losses due to the reduced productivity caused by disability or premature death. *Id.* at 2. The costs were estimated using the Health Effects Costs Model (HECOM), using the health effects data from CRAC calculations as input and using standard health economics cost of disease estimation techniques, along with some key assumptions in arriving at the cost estimates of acute radiation injuries and fatalities and latent cancers. The major assumptions used in deriving cost estimates using HECOM are described in the testimony. *Id.* at 3-4. The data provided in the testimony were not included in the FES because they give a likely magnitude of cost rather than precise estimates. Direct and indirect cost factors are based on national data, not specific to the area surrounding Limerick and several costs unique to the health costs of nuclear power plant accidents are not included in HECOM. *Id.* at 4. Some of the estimated health costs are large, i.e., over 2 billion dollars. The probabilities of the severe releases leading to such costs are so low, however, that the risk per reactor-year of such costs, expressed in dollars per year, is relatively insignificant. *Id.* at 5.

F-35. The Applicant estimates the offsite economic risk of health effects at $1900 per reactor-year, compared to its estimate of $6000 per reactor-year for the median economic risk due to other offsite economic risks from reactor operation. These estimates indicate that offsite economic risk is increased by approximately one-third if the cost of health effects is considered. Daebeler *et al.*, ff. Tr. 11,114, at 40. This conclusion is supported by the results of a recent study at the Sandia National Laboratories that estimates the ratio of the cost of health effects to total offsite cost varies from 5% to 25%. App. Ex. 149, at 12 and Table 11.

F-36. The Board notes that the estimates of health costs are uncertain, at best. Assumptions of the cost of human life vary widely. Predictions of applicable discount rates are arbitrary. Some costs, e.g., screening of potentially exposed persons, transportation, genetic effects, were not considered. National averages of costs rather than Limerick-specific costs were used. Tr. 114,000-08 (Richter).

F-37. In sum, the Board finds that a more complete discussion in the FES of the quantification of the cost of medical treatment of health effects may have been arguably helpful to the public (as opposed to the informed professional). The Board concludes, however, that the FES adequately considers the quantification of the cost of economic effects of severe accidents, since the addition of quantified costs of medical treatment is both so uncertain and so low when the probabilities of occurrence are factored in. In any event, the record and decision in this proceeding now adequately disclose such costs.

d. FES Format (DES-4B)

F-38. The FES, itself, provides some data in the complementary cumulative distribution function (CCDF) format and other data are expressed as a risk, e.g., cost per reactor-year. Reactor accident consequences are calculated using the CRAC computer program, which provides the CCDFs as output. No similar computer program exists for calculating health care costs and regional economic costs of accidents. These costs are expressed as average values and the risks are expressed on a per-reactor-year basis, using the CRAC-generated data as input. While the FES did not express health care costs quantitatively, Staff testimony relating to LEA Contention DES-4A-6 explains the analysis that was performed. Additional economic impacts that were quantified in the FES or the Staff testimony include health effects, regional industrial impacts, decontamination and replacement power. Richter, ff. Tr. 11,148, at 6.

F-39. The Applicant asserts that while not all aspects of the analysis of costs and risks are currently amenable to a fully rigorous probabilistic treatment, both the Staff and the Applicant have treated them using the current state-of-the-art in risk assessment to provide full disclosure. The Applicant believes that we must look at the entire discussion, both its quantitative and qualitative aspects, to understand the risks associated with the operation of Limerick. Daebeler *et al.*, ff. Tr. 11,114, at 41-42.

F-40. Since LEA provided no testimony or witness on this contention, it is difficult to understand exactly what LEA means by the "format obscures the total impact of severe accidents at Limerick." Judging from LEA's Proposed Findings 110-117, it would appear that the concern is not with structure, but with content and manner of presenting results. We agree with the Applicant that to understand the risks associated with the operation of Limerick one must look at the entire discussion, both its quantitative and qualitative aspects. As we have concluded with respect to part A of this contention, so we conclude with respect to part B, that the FES and the record in this proceeding adequately disclose and consider the risk of severe accidents at Limerick. To the extent that LEA believes that the FES consideration of *total* impact of severe accidents at Limerick should include something in addition to what is already there, we find no basis for such a conclusion. We find this part of the contention, DES-4B, without merit.

3. LEA-DES-3: People Will Decline to Evacuate

F-41. This contention states:

> The DES' severe accident consequence modeling fails to account for the probability that a portion of the population will fail to take protective action despite planning and instructions, thus understating the actual consequences of a severe accident at Limerick.

F-42. LEA's basis for this contention was an EPA-sponsored study of evacuations. Hans and Sell, "Evacuation Risks — An Evaluation," EPA-520/6-74-002, U.S. Environmental Protection Agency (June 1974). LEA asserted in its basis that the Hans and Sell study showed that a percentage of the population ranging from 6% to 50% would not evacuate despite instructions to do so. Actually, as now apparently conceded in LEA's Findings (LEA PF 28, at 11), the referenced study stated that approximately 6% of the population refused to evacuate in the cases studied. The 50% figure was taken from a separate report quoted by Hans and Sell studying the response to Hurricane Carla in 1961. That report considered the evacuation behavior of people not only in the Texas county in which the hurricane came ashore, but also another Texas county, two cities located 100 miles to the northeast and a county in Louisiana located 200 miles from where the storm came ashore. Daebeler *et al.,* ff. Tr. 11,114, at 24-25. We agree with the testimony that the inclusion of people living great distances from the eye of the hurricane, and the fact that a majority of people in the affected area were not advised to evacuate, make the 50% nonevacuation figure invalid as a guide for a postulated evacuation at Limerick. *Id.* at 25.

F-43. In sum, there is no basis to assume that with the required emergency plan in place, including prompt notification systems and followups, that more than a small percentage of the population — perhaps, for all we know, about 5-6% — would initially fail to evacuate. It requires, however, further speculation to assume that such persons would continue to refuse to do so in the face of followup evacuation efforts by authorities and the evident evacuation of the rest of the population. *See* Hulman and Acharya, ff. Tr. 11,148, at 5; Tr. 11,513-14 (Hulman). The evidence that only a very small percentage of the population in the plume exposure Emergency Planning Zone (EPZ) would fail to evacuate was buttressed by the report of an evacuation that took place in 1982 in the vicinity of the Waterford Steam Electric Station in Louisiana. In that case, an area of approximately 60 square miles, with the reactor situated fairly close to the center, was evacuated as a result of a nonnuclear chemical plant accident. The emergency response took place in the context of the planning that had been done for the nuclear power plant. The nonevacuating fraction of the population was approximately 0.2%, or 50 people out of 16,000. Significantly, the authorities knew the names and addresses of all nonevacuating individuals shortly after the accident. Tr. 11,514-16 (Kaiser); Tr. 11,517 (Hulman).

F-44. The Board does not believe it is clear that persons who, in the exercise of their individual liberty refuse to evacuate, even after followup efforts, should be considered as part of the total societal risk of a severe accident. Nevertheless, the record also discloses the effect on the risk estimates if a small percentage of the population refuses to evacuate. The Applicant's assumed base case protective actions, for its risk calculations in SARA, are those of evacuation of the entire population within 10 miles of the Limerick plant, and normal activities for 12 hours after plume passage with subsequent relocation for people between 10 and 25 miles from the plant. It modified this computer run for this base case to assume that 6% of the population would not take those evacuation and relocation actions. Daebeler *et al.*, ff. Tr. 11,114, at 27.

F-45. The Applicant's sensitivity analysis assumed that the 6% nonparticipating fraction of the population was uniformly dispersed throughout the area. Tr. 11,503-04 (Kaiser). The Board believes that this is probably conservative, since persons closer to the accident are more likely to heed the advice of authorities to evacuate (or take other recommended protective actions). The nonparticipating 6% were assumed to remain outdoors for 24 hours after the declaration of an emergency, and then to rapidly relocate. This assumption is the equivalent of exposures that would be accumulated in 2 to 3 days of normal activities following plume passage. Daebeler *et al.*, ff. Tr. 11,114, at 27-28; Tr. 11,504-06 (Kaiser). We find the sensitivity analysis to reasonably bound the speculative element of a nonparticipating percentage of the population. We find no basis to accept LEA's unsupported view (LEA PF 32-34), that even a much smaller percentage of the population, let alone 6%, would continue to fail to follow the advice of authorities to leave the area after 2 to 3 days.

F-46. The results of Applicant's sensitivity analysis increased the predicted public risk of early fatalities by 49%. We agree with the testimony of the Applicant and the view of the NRC Staff (Staff PF 36), that this 49% increase is relatively small for calculations of this type. Other uncertainties in the assessment of severe accidents, such as uncertainties in source terms, are much more significant. Daebeler *et al.*, ff Tr. 11,114, at 28. The uncertainties in the results of a PRA are large. It is stated in the FES that the risk estimates could be "too low by a factor of 40 or too high by a factor of 400." Tr. 11,286-90 (Acharya, Hulman). Typically, the area under the upper-estimate CCDFs in SARA are on the order of a factor of 100 greater than the area under the lower-estimate CCDFs. Any comparison of the results of sensitivity studies, or of other PRAs must be made with this large range of uncertainty in mind. If the uncertainty ranges of two estimates are large and overlap to a large extent, then the two results cannot be regarded as being significantly different. Thus, for instance, changes of a factor of 2 in estimates of public risk are insignificant in view of the large range of uncertainty. Daebeler *et al.*, ff. Tr. 11,114, at 9. *See also id.* at 8, and Staff Ex. 29 (FES), at 5-91 and 5-108 to 5-115.

F-47. There is no basis for LEA's assumption (LEA PF 38-39), that persons would remain in "hot spots" for 7 days so as to receive high (200-rem) bone marrow ground doses, thereby increasing the 49% increase calculated by the Applicant. Our findings above are to the contrary; again we believe the assumption of a 2- to 3-day period of failure for 6% of the population to take protective action to be more than reasonable — it is likely quite conservative.

F-48. The NRC Staff's base case in the FES, as will be further discussed in our findings below on other NEPA severe accident contentions, assumed a 100% evacuation of a 10-mile plume exposure pathway EPZ, after an average delay time of 2 hours and an average evacuation speed of 2.5 miles per hour (mph). The Staff, consistent with our own view above, believes the vast majority of people would heed instructions to evacuate. Hulman and Acharya, ff. Tr. 11,148, at 4. However, the FES (Staff Ex. 29), also presents an alternative analysis in Appendix M, using a postulated "Early Reloc" model of emergency response. The Staff did not perform this alternative analysis in response to this contention. Therefore, LEA's criticism that the Staff's alternative analysis is not a direct sensitivity analysis varying the factor of nonparticipation of the population is superficially valid. *See* LEA PF 35-37. However, LEA misses the point that, rather than studying the effects of small variations around the average values of all the different evacuation parameters, the "Early Reloc" model was used to reasonably bound the effect of different levels of effectiveness of offsite emergency response. Hulman and Acharya, ff. Tr. 11,148, at 4; Tr. 11,519-20 (Acharya). Staff Ex. 29 (FES), at p. 5-100.

F-49. In the "Early Reloc" alternative Staff model, it was assumed that all people in areas contaminated within the plume within a 10-mile EPZ would not evacuate until 6 hours after passage of the plume. Beyond the 10-mile EPZ, just as in the Staff's base case, people were assumed to relocate 12 hours after plume passage if they are in highly contaminated "hot spot" areas (projected 7-day ground dose of 200 rems to the bone marrow); if not, persons beyond the 10-mile EPZ were assumed to relocate after 7 days. Staff Ex. 29 (FES), at p. 5-80 and p. 5-82. Tr. 11,511, 11,534 (Acharya). Therefore, this model assumes that *all* people in the 10-mile EPZ receive a ground dose for 6 hours in addition to the plume dose (and for larger periods for people outside the assumed 10-mile EPZ). Tr. 11,521 (Acharya). For this reason, even though a percentage of nonevacuating people was not one of the varied parameters, the results of the Staff's alternative analysis bounds the results of the Applicant's sensitivity analysis, which we have already found to be reasonable. Tr. 11,529-34 (Hulman).

F-50. For the reasons stated, the FES adequately presents a range of consequences in the event 6% of the population declines to participate in an evacuation for the first 2 to 3 days after being advised to evacuate. This is further supported and made more explicit by the Applicant's analysis and our findings in this proceeding.

4. LEA-DES-1: Relocation of People Beyond 10 Miles Implausible

F-51. DES-1 states:

> The DES' severe accident consequence modeling assumes the relocation of the public from contaminated areas beyond the 10-mile plume exposure EPZ. (DES, Supp. 1, pp. 5-21 to 5-22). Such an assumption in Limerick's case is implausible and without foundation in fact.

F-52. LEA asserts, as basis, that no planning exists or is presently contemplated for such a "relocation." It notes that NRC planning guidance contemplates the possibility of *ad hoc* response beyond the approximate 10-mile plume exposure EPZ, but believes in the case of Limerick such an *ad hoc* relocation beyond the 10-mile radius is impractical, particularly in the SE and SSE sectors (towards Philadelphia) in which the year 2000 population between 10 and 25 miles will be 680,330 and 505,011, respectively. LEA states that no precedent exists for the *ad hoc* "relocation" of such numbers of people.

F-53. The Staff's severe accident modeling does, in fact, assume that those persons whose projected 7-day dose to the bone marrow would be more than 200 rems, would be relocated. Hulman and Acharya, ff. Tr. 11,525, at 4. Such potential evacuation is not considered in isolation, however. Rather, the Staff, using the CRAC computer program, calculated the complementary cumulative distribution function values for the number of people to be relocated under this criterion. *Id.* From this calculation it can be determined that for relocation from the hot spots outside the 10-mile EPZ the probability that 5000 or more persons would be affected is approximately 10^{-6} per reactor-year, the probability that 50,000 or more persons would be affected is approximately 10^{-7} per reactor-year and the probability that 300,000 or more persons would be affected is approximately 10^{-}[illegible] per reactor-year. Finally, the probability that 500,000 or more persons would be affected is approximately 2×10^{-11} per reactor-year. These estimates include the probabilities of accidents, the probabilities of the weather sequences and the probabilities of the wind blowing toward the various population sectors. *Id.* at 4-5.

F-54. The basis for assuming that *ad hoc* relocation of individuals outside of the 10-mile EPZ is discussed in NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," App. Ex. 139, which states on page 16 that for distances exceeding 10 miles, "actions could be taken on an *ad hoc* basis using the same considerations that went into the initial action determinations." Also, NUREG-0654, "Criteria for Preparation of Emergency Response Plan and Preparedness in Support of Nuclear Power Plants," App. Ex. 140, states on page 12 that "detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary." Daebeler *et al.*, ff. Tr. 11,114, at 10-11.

F-55. The Applicant carried out a series of sensitivity studies to determine the effects of alternative modeling assumptions concerning shielding and relocation of individuals outside of the 10-mile EPZ. *Id.* at

14-16. From these studies it is concluded that the results are insensitive (within a factor of 2 or less) to a variety of assumptions. *Id.* at 14. The risk of early fatality to individuals between 10 and 25 miles ranges from 4.5×10^{-5} to 9.3×10^{-5}. *Id.*, Table 1.

F-56. Evacuation of large numbers of people have in fact taken place expeditiously. Baton Rouge, Louisiana, population 150,000, was almost totally evacuated in 2 hours after a decision was made to evacuate the city following an accident involving a chlorine barge. Wilkes Barre, Pennsylvania, population 75,000, was effectively evacuated to a level of 96% in 1 hour because of a flood warning. Downtown Portland, Oregon, with a population of 100,000 was evacuated in 1 hour during a civil defense test exercise. One of the largest recent public evacuations occurred in Canada. Late in the evening of November 10, 1979, a freight train transporting both flammable and toxic materials derailed in downtown Mississauga, Ontario, Canada's ninth largest city. During the next 24 hours, 216,000 people were evacuated from homes and hospitals in a 50-square-mile area around the accident site. *Id.* at 16-17.

F-57. The contention is therefore incorrect in its assertion that there is no precedent for the *ad hoc* relocation of large numbers of people.

5. LEA-DES-2

F-58. This contention states:

> The DES' severe accident consequence modeling uses an assumption of a uniform two hour evacuation delay time in its emergency response model. (DES, Supp. 1, pp. 5-21 to 5-22). This assumption understates the likely delay time for a high population density site such as Limerick. This understatement of delay time results in an understatement of Limerick's risk, because accident sequence calculations are sensitive to evacuation time delay assumptions.

F-59. The FES considers three types of response to severe accidents at Limerick. Only the first type assumes evacuation. This response, identified as Evac-Reloc (evacuation of the plume exposure pathway emergency planning zone (EPZ) followed, if necessary, by relocation of persons outside of this zone), assumes an evacuation distance of 10 miles, a delay time of 2 hours, an effective evacuation speed of 2.5 mph and a 15-mile path length for each evacuee over which radiation exposure is calculated. Staff Ex. 29 (FES), at p. 5-81. Risk calculations may, in some cases, be sensitive to evacuation time estimates, which depend not only on the assumed delay time, but on the evacuation speed and effective downwind distance to be traversed. Hulman and Acharya, ff. Tr. 11,525, at 5-6, 9. For some accidents there would be sufficient warning time to allow the public to evacuate before the plume could reach them, even if the evacuation time were relatively long. For others, the warning time could be short and many persons in the (plume exposure pathway) EPZ could not evacuate before being overtaken by the plume (even if the evacuation time were relatively short). The FES considers a range of risk assuming a 2-hour delay time before evacuation to no evacuation at all. *Id.* at 6.

F-60. The Staff's basis for a 2-hour delay time does assume that there is a well-established emergency response plan, periodic testing of the notification system and procedures, and exercises and drills to maintain the plan in readiness. Hulman and Acharya, ff. Tr. 11,525, at 6. Such assumptions are not unreasonable, given that these actions are required by the Commission's regulations. 10 C.F.R. § 50.47, and Appendix E to 10 C.F.R. Part 50.

F-61. The 2-hour delay time is assumed to result from three time increments; 15 minutes (from the reactor operator's warning) for the authorities to interpret the plant data and decide to promptly notify people to evacuate, 15 minutes to notify most of the people in the 10-mile EPZ to evacuate, and 90 minutes for people to prepare to evacuate and to get under way. *Id.* at 7. There would likely be variations in the delay time around the 2 hours in either direction, but the impact of these variations on risk estimates would not be expected to be substantial. *Id.* at 6.

F-62. The 2-hour delay time assumed for Limerick is the same as that assumed for the Indian Point site, which was based on two evacuation time studies — one prepared for the Indian Point licensees and one prepared for the Federal Emergency Management Agency (FEMA), by different contractors. This delay time was characterized by the Indian Point Atomic Safety and Licensing Board (ASLB) as reasonable. *Consolidated Edison Co. of New York* (Indian Point, Unit No. 2), LBP-83-68, 18 NRC 811, 888 (1983). Because the population within the 10-mile EPZ at Indian Point (0.25 million people projected in 1990) is larger than the population within the 10-mile EPZ at Limerick (0.16 million people projected in 2000), the Staff considers the 2-hour delay time at Limerick as reasonable. *Id.* at 7-8. The evidence additionally indicated that this delay time is appropriate even for moderately adverse site conditions such as light snow, ice, and moderately severe hurricanes and earthquakes. *Id.* at 6-7.

F-63. LEA, in its basis for this contention, concludes that a more appropriate delay time would be in excess of 3 hours, based on the evacuation model developed at Sandia National Laboratories. App. Ex. 138. This model, based on historical data on experience with unplanned or impromptu evacuation following transportation accidents, derived values of 1 hour, 3 hours, and 5 hours for 15%, mean, and 85% likely delay times. Instead of 2.5 mph, however, 10 mph or higher evacuation speeds were assumed. The Staff does not consider an evacuation speed of 10 mph appropriate for Limerick, however, based on its estimate of required travel time to evacuate the 10-mile EPZ. *Id.* at 9.

F-64. Based on the 2-hour delay time and 2.5-mph evacuation speed, compared to the Sandia model using a 3-hour mean likely delay time and a 10-mph evacuation speed, the Staff believes that it should be inferred that the Staff's evacuation parameters have not resulted in understatement of Limerick risks. *Id.* at 10.

F-65. To examine the effects of changes in delay times and evacuation speeds on the final risk results, the Applicant performed sensitivity analyses using various models and various values for the delay time and evacuation speed parameters. These studies used the CRAC 2 computer code and the radioactivity release source terms developed by the Applicant in its Severe Accident Risk Assessment (SARA) study. The SARA evacuation model incorporates the results of the Sandia study (on delay times) explicitly with delay times weighted as follows: 1 hour — 30%, 3 hours — 40%, and 5 hours — 30%. The Applicant found that the FES risk estimates do not differ greatly from those in the Sandia model, even though the delay times and evacuation speeds are different in the two models. Daebeler *et al.*, ff. Tr. 11,114, at 22-23, 58. Applicant's sensitivity studies included variation of evacuation clear times from 4 to 13 hours and delay times of 1, 3 and 5 hours combined with a 2.5-mph evacuation speed. All of the results were within a factor of 3 of the result for the FES Evac-Reloc Model. The Applicant concludes that the Staff use of a 2-hour time in the FES does not lead to a significant understatement of Limerick's risk. Daebeler *et al.*, ff. Tr. 11,114, at 23.

F-66. LEA implies that a longer delay time for Limerick would be incurred because of its higher-than-average population density. To the contrary, the Hans and Sell report, upon which the Sandia Generic Study is based, contains examples of evacuation from areas with population densities greatly exceeding the 700 persons per square mile located within 10 miles of Limerick. Daebeler *et al.*, ff. Tr. 11,114, at 21.

F-67. Based on the record in this proceeding we find no basis for the assertion that the assumption of a 2-hour delay time for evacuation of the 10-mile EPZ at Limerick understates the likely delay time. It is clear that some people will evacuate earlier and some later, but the use of 2 hours versus, say, 3 or more hours is reasonable for the purposes of estimating risk provided the evacuation speed assumed also is reasonable. The assumption in the FES of a 2.5-mph, rather than a 10-mph, evacuation speed compensates, even though not completely, for the shorter delay time. Tr. 11,556 (Kaiser). Based on the uncertainties of postulating actual evacuation conditions, and the sensitivity analyses described above, we find that the FES assumption of a 2-hour delay time, together with the assumption of a 2.5-mph evacuation speed, does not result in any significant understatement of Limerick's risk, if indeed there is any understatement. Consequently, this contention is without merit.

6. *Conclusions of Law as Applied to LEA and City Severe Accident Contentions*

a. *LEA's Proposed Conclusions of Law*

F-68. LEA has summarized its position as to the defects in the FES in its proposed Conclusions of Law. Proposed Findings (July 26, 1984). It first cites *Baltimore Gas and Electric Co. v. Natural Resources Defense Council,* 76 L. Ed. 2d 437, 446-47, 452 (1983), to the effect that:

> The National Environmental Policy Act (NEPA) places upon an agency the obligation to consider every significant aspect of the environmental impact of a proposed action . . . and requires an EIS to disclose the significant health, socioeconomic and cumulative consequences of the environmental impact of a proposed action.

F-69. It then quotes from the NRC Statement of Interim Policy on Nuclear Power Plant Accident Considerations under the National Environmental Policy Act, 45 Fed. Reg. 40,101 (June 13, 1980), as follows:

> Environmental Impact Statements shall include a reasoned consideration of the environmental risks (impacts) attributable to accidents at the particular facility . . . within the scope of each such statement. 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.
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> * * *
>
> The environmental consequences of releases whose probability of occurrence has been estimated shall also be discussed in probabilistic terms. Such consequences shall be characterized in terms of potential radiological exposures to individuals, to population groups, and where applicable, to biota. Health and safety risks that may be associated with exposures to people shall be discussed in a manner that fairly reflects the current state of knowledge regarding such risks.

F-70. Finally, LEA concludes that the FES fails to comply with these mandates for eight reasons. We have already discussed the fact that compliance with NEPA need not be restricted to the content of the FES alone. Rather, our findings and conclusions, based on the entire record before us, are deemed to amend the FES.

F-71. Generally, with respect to *Baltimore Gas and Electric*, we note that the key word is "significant." As all parties agree, the estimates of environmental, including health, effects resulting from low-probability, high-consequence accidents are attended by large uncertainties. Where such estimates are clearly small, as they are here, compared to the risks to which the environment and the population are otherwise exposed, second-order effects cannot reasonably be considered significant. Further, whatever significance such second-order risks may have, they may reasonably be considered as enveloped by the uncertainty in the estimates of the dominant risks. Similarly, the precision of the estimates of the dominant risks is not important where the risks are clearly small — taking into account the uncertainty of the estimate — compared to the risks otherwise extant.

F-72. With respect to the first paragraph quoted from the Statement of Interim Policy, the Board certainly agrees that the FES and this decision should give equal attention to the probability of occurrence of releases and to the probability of the environmental consequences of those releases. This, we believe the Staff, the Applicant and we have done. With respect to the second paragraph, we believe Staff and Applicant testimony and our own familiarity with the subject supports the conclusion that the health and safety risks that may be associated with exposures to people have been discussed in the FES and on the record of this proceeding in a manner that fairly reflects the current state of knowledge regarding such risks.

F-73. Notwithstanding the above, we have found in a number of instances that the FES might have led to easier comprehension by the public (as opposed to the informed professional) had there been explicit discussion in the FES itself of the rationale for including some matters and excluding others. Perhaps this was a consequence of using state-of-the-art knowledge and methodology.

F-74. Based on the above, and the record before us, we find

(a) Certain health effects which may be caused by a severe accident at Limerick and their associated probabilities, including genetic effects, nonfatal cancers, child developmental impairment caused by *in-utero* radiation exposure, spontaneous abortions, sterility, benign thyroid nodules, and hypothyroidism, have been adequately disclosed.

(b) The total land area in which crops and milk will be interdicted and the probabilities associated with such interdiction, have been adequately considered and disclosed.

(c) The population in the areas to be interdicted, and the probabilities associated with such population interdiction due to severe accidents at Limerick, have been adequately considered and disclosed.

(d) The economic cost of medical treatment of all health effects of severe accidents at Limerick, and the probabilities associated with such costs, have been adequately considered and disclosed.

(e) The assumption used for population relocation beyond the plume exposure EPZ in the calculation of health effects is not inappropriate.

(f) The evacuation delay time used in the emergency response model for calculating health effects is not inappropriate.

(g) The probability that a portion of the population will fail to take protective action has been adequately taken into account, thus the risk of health effects of severe accidents has not been understated.

(h) The total risk of a two-unit facility over 30 years of operation is adequately disclosed by disclosing the risk per reactor-year of a single unit and the fact that the risk from two units is approximately twice that of one unit.

b. City's Proposed Conclusions of Law

F-75. The City does not propose specific conclusions of law with respect to its three admitted contentions. We have carefully considered each contention and have denied them for the reasons discussed in sections of this decision following this one. The City, however, concludes that further NEPA assessment in terms of weighing environmental costs versus benefits of the project is warranted for Unit No. 2, and a stay by the Nuclear Regulatory Commission of any determination of licensing of Unit No. 2, in terms of the acceptability of environmental impacts, is appropriate. City PF at 19-21 (July 26, 1984). We discuss the City's basis, as set out in its proposed conclusions of law.

1. The National Environmental Policy Act of 1970 ("NEPA") directed federal officials "to use all practicable means, consistent with other essential considerations of national policy," to protect the environment. 42 U.S.C.A. § 4331. Consistent with that mandate, the Nuclear Regulatory Commission, prior to issuance of an operating license for both Limerick units, must fully disclose the environmental impacts of the units' operation and must factor into its licensing decision consideration of NEPA's mandate.

F-76. We have found that the FES and the record in this proceeding fully disclose the environmental impacts of the operation of both units and we have taken NEPA's mandate into consideration in reaching our conclusions. The City, by its cross-examination, has not controverted the evidence of the Staff and the Applicant in this regard.

2. The informative uses of the environmental impact study are to provide information to the general public and public officials at all levels of government, 40 C.F.R. § 1500.1(b), and to provide the basis for an informed decision on the part of the NRC. *Sierra Club v. Froehlke*, 345 F. Supp. 440, 444 (W.D. Wis. 1972), *aff'd*, 486 F.2d 946 (7th Cir. 1973). On this count the study must be reasonably thorough and must take a "hard look" at the environmental consequences. *Kleppe v. Sierra Club*, 427 U.S. 390, 410, n.21 (1976).

F-77. Similarly, we find that the FES and the record in this proceeding provide information to the general public and public officials at all levels of government, and, together, are reasonably thorough and do take a "hard look" at the environmental consequences of severe accidents at Limerick. Neither has the City, by its cross-examination, controverted the evidence of the Staff and the Applicant in this regard.

3. NEPA does not mandate informational requirements only, however. NEPA injects environmental considerations into the decision making process itself. *Weinberger v. Catholic Action of Hawaii*, 454 U.S. 139, 143 (1981). An essential element of decision making is whether alternatives should be considered in light of any benefits of the action in relation to the measured environmental impacts of the action. 42 U.S.C.A. § 4332(2)(c)(iii).

F-78. The Commission, in its Statement of Consideration accompanying the change in 10 C.F.R. Part 51, relating to Need for Power and Alternative Energy Issues in Operating Licensing Proceedings (47 Fed. Reg. 12,940 (1982)) stated that it is not necessary, absent a showing of special circumstances, to consider the issues of need for power and alternative energy sources at the operating license stage of a licensing proceeding. (*See also* 10 C.F.R. § 51.53(c)). The City has not made a showing of special circumstances in this proceeding and therefore the issue is not a proper subject for review by this Board. Further, the City now raises essentially the same issue that was the subject of its Contention City 17. That contention was opposed by the Staff and the Applicant and was rejected by the Board. (Memorandum and Order Confirming Rulings and Schedules Made at Special Prehearing Conference on NEPA Severe Accident Contentions (April 20, 1984) (unpublished), slip op. at 4).

4. In keeping with the National Environmental Policy Act, 40 C.F.R. 1502.22(b) and the Commission's Environmental Protection Regulations, 49 Fed. Reg. 9352,

9357 (March 12, 1984), the Board has considered a full range of both the probabilities of various accident scenarios and their associated consequences. Given the developmental status of these types of analyses and their high degree of uncertainty, a reasoned approach is to review and consider this range, including the calculated uncertainty range. We have considered on this record a reasonable range of dose conversion factors, exposure levels (protective action effectiveness), bad weather, and the probability calculation uncertainty range. Although upper bound results were not portrayed here in every instance, we have compensated for that lacking by giving greater weight to the uncertainty range, especially the upper bounds.

F-79. It is inherent, perforce, that estimates of very low probability, severe consequence accident risk, for which there is no direct experience, will have large uncertainty. It is correct that we have considered the uncertainty range, but we find there is no basis for giving greater weight especially to the upper bounds. Rather, we maintain that in consideration of risk it is not only proper, but mandatory, to consider the combination of probability with the magnitude of the consequence.

5. Based on our consideration of this record in the above described framework and what has been thereby disclosed in terms of the environmental impacts of potential severe accidents and the uncertainty in measuring both the probabilities and consequences associated therewith, we conclude that further NEPA assessment in terms of weighing environmental costs versus benefits of the project is warranted for Unit No. 2. A stay by our Commission of any determination of licensing of Unit No. 2, in terms of the acceptability of environmental impacts, is appropriate for the following additional reasons:

(a) The pending availability, for NRC review, of the Pennsylvania Public Utility Commission's investigation results will precisely focus on and develop the economic issues associated with Unit No. 2's potential operation.

(b) Unit No. 2 is only partially completed, with in-service not scheduled until the 1990s. A stay of licensing now will not have the construction scheduling impact associated with such a stay for a nearly completed plant.

(c) There have been vastly changed circumstances since 1973, when this issue was last examined by the Commission in an adjudicatory context. These changes will affect the economics of the plant's operation. Also the partial nature of construction completion will affect the economic analysis when comparing Unit No. 2 to alternatives, in contrast to comparing the economics of a completed plant to the economics of alternatives.

(d) The lack of previous consideration at the construction stage of conservation, cogeneration, etc. as alternatives also compels reconsideration. Conservation, good management, cogeneration, and rate structures to promote efficient use of production are now an essential component of the Nation's energy policy. National Energy Act of 1978. They are no longer viewed as "remote and speculative" possibilities.

In conclusion, before doubling the potential for the public's exposure to these environmental impacts in such a high density population area, NEPA requires us, as federal officials charged with protecting the environment, to stay a decision on Unit No. 2 until the Pennsylvania Public Utility has completed its investigation.

F-80. City's reasons to stay a decision on Unit No. 2 simply will not wash. First, the fact that there are uncertainties in estimating (of course they cannot be "measured") both the probabilities and consequences of potential severe accidents in no way supports the conclusion that further NEPA assessments are required. The record is complete and adequate with respect to environmental costs. The benefits (a reconsideration of need for power and alternative energy sources) are not a proper subject for litigation before this Board. No special circumstances have been shown or are apparent to call into question at this late date the *environmental* judgments reached many years ago, at the construction permit stage, on the benefits of the proposed action. This is not affected by *economic* considerations of:[23]

(a) the pending availability of the Pennsylvania Public Utility Commission's investigation results of economic issues,

(b) a change in construction scheduling impact,

(c) possible changes in the economics of the plant's operation.

F-81. Finally, we do not accept the conclusion that the public's exposure to the environmental impacts of severe impacts has been doubled. Philadelphia Electric's application has been and is for operating licenses for two units at Limerick. The fact that risk estimates have been expressed in terms of reactor-years of operation certainly has not obscured the fact that risk will attend operation of both units.

F-82. City's proposed Conclusions of Law are rejected, for the reasons given above.

7. *City-14: Evacuation Speed, Backups and Bad Weather*

F-83. This contention, as admitted, alleges three reasons why the FES does not accurately reflect either the median or upper estimates of the radiological effects which would result from an accident at Limerick because several key input assumptions associated with human activity after a severe accident are not realistic: (a) incorrect assumption of evacuation speed, (b) failure to correctly consider backup of evacuees at Philadelphia's outskirts, and (c) failure to adequately consider bad weather scenarios. We discuss them in turn.

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[23] *See Consumers Power Co.* (Midland Plant, Units 1 and 2), ALAB-458, 7 NRC 155, 161-63 (1978) (economic cost of the proposed action is only material under NEPA when there are environmentally superior alternatives).

a. Evacuation Speed

a. The base case average evacuation time (speed) of 2.5 mph is based on a 1980 study which is now inaccurate. City, as part of this section of the contention, refers to the Statement of Issues of the Commonwealth of Pennsylvania with Respect to Offsite Emergency Planning, January 30, 1984.

F-84. In its Statement the Commonwealth asserted that the Applicant must prepare an updated evacuation time estimate study for the Limerick plume exposure pathway EPZ; the evacuation time study the Applicant has submitted to the NRC for approval is outdated and based on inaccurate information. Deficiencies in the study include, but are not necessarily limited to, reliance on out-of-date and inconsistent census data, use of incorrect evacuation routes, use of a concept of "maximum evacuation time" that does not accurately reflect the size of the plume EPZ, and failure to account for the notification system to be installed by the Applicant.

F-85. The Staff did derive the mean effective radial speed of 2.5 mph using an Applicant's consultant 1980 report estimate of 4 hours travel time to clear the 10-mile EPZ. This was not the only basis for this rate of travel. The Staff, in its risk analysis for the Indian Point site, derived an effective evacuation speed of 1.5 mph on the basis of a mean estimate of 6.7 hours of travel time to clear the 10-mile EPZ. This was based on two evacuation time studies made for Indian Point, as reviewed in NUREG/CR-1856, "An Analysis of Evacuation Time Estimates Around 52 Nuclear Power Plant Sites," Vol. 1, May 1981. This speed, equivalent to a slow walk, was considered reasonable by the Indian Point Licensing Board. Hulman and Acharya, ff. Tr. 11,525, at 12; *Indian Point*, LBP-83-68, *supra*, 18 NRC at 888. Because the population within the Limerick 10-mile EPZ (0.16 million projected for the year 2000) is considerably less than the population within the Indian Point 10-mile EPZ (0.25 million projected for the year 1990) the Staff judged the effective evacuation speed of 2.5 mph for Limerick to be consistent with the 1.5 mph for Indian Point. The Staff recognized there could be other factors, such as terrain differences, differences in capacities of road networks, etc., which could influence the effective evacuation speeds. Hulman and Acharya, ff. Tr. 11,525, at 12.

F-86. The Staff did not presuppose great accuracy in the 2.5-mph speed estimate or in other parameters used in the risk analysis. It asserts that a reasonable bounding of risk estimates due to minor perturbations in evacuation model parameters is provided by the use of the "Early Reloc" mode of emergency response discussed in an alternative risk analysis of Appendix M of the FES. Finally, the Staff notes that the risks of early fatality are dominated by Limerick reactor accidents initiated by severe earthquakes for which evacuation is unlikely, and only the "Late Reloc" mode of emergency response would apply. Hulman and Acharya, ff. Tr. 11,525, at 10-13.

F-87. To examine the effects of changes in delay times and evacuation speeds on the final risk results, the Applicant performed sensitivity analyses using various models and various values for the delay time and evacuation speed parameters. The results of these calculations were summarized as estimates of the public risk of early fatality, from which it was concluded that the predictions of public risk do not differ significantly when the evacuation speed is varied from 2.5 to 10 mph. Daebeler *et al.*, ff. Tr. 11,114, at 22-23, and Table 2.

F-88. The Board finds that the value of 2.5 mph for the average evacuation time may, indeed, not be accurate. We note, however, that comparison of the FES results with the results of an extreme case of a 3-hour delay time and a 1-mph effective evacuation speed would change the estimate of the predicted public risk of early fatality from 3.5×10^{-5} to 9.9×10^{-5}, a factor of less than 3, which is insignificant compared to the uncertainty of the estimate itself. *Id.*, Table 2. *See also* our Findings above on DES-2. This part of the contention (City 14a) is without merit.

b. Evacuee Backups at the Outskirts of Philadelphia

b. Not included in the base case is the known phenomenon that as evacuees approach the City outskirts, their speeds would reduce, backups would occur and consequences due to trapped evacuees would increase.

F-89. Philadelphia, at is nearest outskirts, is approximately 21 miles from the Limerick reactors. The Staff does not disagree with the City assertion, but concludes that there would be no appreciable changes in the results of the risk calculations, taking the backup phenomenon into account, for the following reasons. First, an accident would have to occur, of low probability, that would release a large amount of radioactivity to result in high radiological doses substantially beyond the 10-mile EPZ. Second, the wind blows toward Philadelphia only 27% of the time. Third, given the above, the atmospheric diffusion conditions would have to be poor to allow sufficient concentrations of radioactivity to remain in the plume. Fourth, evacuees would be advised that after crossing the 10-mile EPZ boundary they should travel in a crosswind direction. Fifth, in an actual situation, contrary to the CRAC Code assumptions, the plume direction would be variable, and the evacuees' directions of motion would be variable. Sixth, the Staff made additional calculations assuming that all the evacuees in the plume exposure pathway within the 10-mile EPZ and in the SE and SSE sectors (toward Philadelphia) would wind up in those sectors between 20-25 miles before the plume arrived and remain there during plume passage. The results of the latter calculations allow the comparison of the estimated societal risks originally calculated for the FES with those calculated in response to the City contention. These comparisons show no increase in early fatalities (assuming supportive medical treatment), a 5% increase in early injuries, a 4% increase in latent cancer fatalities (excluding thyroid), a 5% increase in latent thyroid cancer fatalities, and a 4% increase in total person-rems, for the calculations based on the stated assumptions. Hulman and Acharya, ff. Tr. 11,525, at 13-17 and Tables 2, 3 and 4.

F-90. Given the magnitude of the uncertainties inherent in the risk analysis calculus, and the conservatisms of the CRAC model cited above, such low percentage changes in the public risk caused by a backup phenomenon have no significance. This part of the contention (City 14b) has no merit.

c. *Bad Weather Scenarios*

> e. The DES does not separately portray the health consequences under bad weather scenarios. Many weather scenarios, including theoretically bad weather conditions, are averaged together.

F-91. The FES does not, in fact, provide a separate showing of the effects of bad weather scenarios on risks. The CCDFs in the FES implicitly portray the effects of bad weather, however, because these higher consequence situations (assuming large releases) have much lower probabilities than the better weather situations and show up in the tail ends of the CCDFs. The weather conditions, themselves, are not averaged. Rather, the consequence magnitudes associated with the ninety-one weather sequences are averaged to obtain the conditional mean value of the consequences. The Staff recognized, however, that bad weather scenarios might have an impact on evacuation. To provide a bounding calculation on the impacts of bad weather, the Staff provided, in Appendix M of the FES, an analysis of an alternative response mode, "Early Reloc," as an alternative calculation of public risk. Comparison of (a) the total societal risks within 50 miles of Limerick per reactor-year for the case of Early Reloc for accident causes other than severe earthquakes and Late Reloc for accidents caused by severe earthquakes (Table M.1a) with (b) the case of Evac Reloc for accident causes other than severe earthquakes and Late Reloc for accidents caused by severe earthquakes (Table L.1a), shows an increase in early fatalities with supportive medical treatment of 20%, an increase in early fatalities with minimal medical treatment of 25% and no change in early injuries, latent cancer fatalities excluding thyroid, latent thyroid cancer fatalities, or total person-rems. Hulman and Acharya, ff. Tr. 11,525, at 17-20.

F-92. While it is true that the FES does not separately portray the health consequences under bad weather scenarios, the worst (weather) cases are included in the calculations of the CCDFs (Tr. 11,672 (Kaiser)) and the bounded changes in public risk due to such conditions can be inferred from the results of the analyses presented. Moreover, such changes, while not a result of not considering bad weather, *per se*, but a result of assumed changes in emergency response, are found not to be significant compared to the uncertainties inherent in the risk analysis.

F-93. This part of the contention (City-14e) has no merit.

8. City-13: Dose-Distance Calculations for Philadelphia

F-94. The essence of this contention is that the FES does not explicitly provide curves of calculated radiation dose resulting from postulated severe accidents at Limerick, as a function of distance, specifically for distances including the City of Philadelphia (City). City asserts that the absence of this explicit data makes it impossible for the Commission to accurately ascertain the likelihood of the public receiving doses in excess of Protective Action Guide (PAG) levels, or in excess of some other unacceptable level of societal risk. In particular, City believes that the high-density population around the (Limerick) site should be taken into account and the probabilities of the occurrence of release and of occurrence of environmental consequences should be presented separately, to be separately understood and evaluated.

F-95. The Staff, in fact, did not separate out doses to individuals or population groups for presentation in the FES, since these were considered as only intermediate parameters in the assessment of the impacts of severe accidents at Limerick. What the Staff did present in the FES were curves of the risk of individual dose versus distance, the individual risk of early fatality versus distance, the individual risk of early injury versus distance, and the individual risk of latent cancer fatality versus distance. Staff Ex. 29, Figs. 5.4i, 5.4j, 5.4k and 5.4*l*.

F-96. The Staff also presented in the FES the results of its calculation of the probability distributions of the number of persons who would receive doses to the whole body, thyroid and bone marrow in excess of

25, 300 and 200 rems, respectively. Staff Ex. 29, Figs. 5.4b, L-1, L-2, L-3 and Table 5.11g. Included in the results were the people of Philadelphia who might be so affected. Calculation of the individual dose versus distance for each release category considered would have resulted in a substantial increase in the bulk of the FES without providing any additional perspective regarding the important health and economic impacts (resulting from severe accidents at Limerick). Acharya, ff. Tr. 11,525, at 22.

F-97. In response to the contention, however, the Staff made calculations of the conditional (i.e., assuming the occurrence of the low-probability severe accident) downwind individual whole-body dose from early exposure versus distance (using CRAC) for the release category II-T/WW, one of the worst consequence categories analyzed, whose probability of occurrence is calculated to be 2×10^{-6} per reactor-year. Given the occurrence of this release, the mean values of downwind individual whole-body dose from early exposure (inhalation dose integrated to 50 years) in the Philadelphia area would be:

Within 20-25 miles: 27 rems.
Within 25-30 miles: 16 rems.

The mean values of population exposures would be:

Within 20-30 miles in the SE sector: 18 million person-rems.
Within 20-30 miles in the ESE sector: 13 million person-rems.

The mean values of latent cancer fatality would be:

Within 20-30 miles in the SE direction: 1100.
Within 20-30 miles in the ESE direction: 800.

All of the above calculations assume the wind blowing toward the SE and ESE directions, which occurs 11 and 16% of the time, respectively. Based on the above, the probability of a II-T/WW type of release impacting people in the SE sector is 2×10^{-7} per reactor-year and is 3×10^{-7} per reactor-year for people in the ESE sector. The conditional person-rem estimates are higher and the conditional latent cancer fatalities are lower than those presented by the City in its contention. Hulman and Acharya, ff. Tr. 11,525, at 23-24.

F-98. The Applicant asserts that it is not necessary to prepare dose-distance curves to disclose environmental risk, since such curves do not consider the effect of the doses on the population. To respond to the contention, the Applicant nevertheless developed dose-distance curves for the two sectors (SE and ESE) which encompass Philadelphia. These are presented as Fig. 2, for whole-body dose, and Fig. 3, for thyroid dose, of the Applicant's testimony. Daebeler *et al.*, ff. Tr. 11,114, at 45.

F-99. The results of preliminary dose-distance consequence calculations by the City for the II-T/WW release with the wind blowing toward the SE sector indicated that the chance of citizens of Philadelphia receiving a whole-body dose of 5 rems at the City boundary (21 miles downwind from Limerick) would be 70%; the chance of a 30-rem dose would be 40%. At the eastern boundary of the City the chance of receiving a whole-body dose of 5 rems would be 55%; the chance of a 30-rem dose would be 15%. In 50% of such releases, given the wind direction toward Philadelphia, the total exposure within the SE sector in the 20–30-mile range could reach 10.5 million person-rems. This, according to the City's Contention 13, could result in as many as 8400 latent induced cancers including 4200 latent cancer fatalities.

F-100. While the Applicant did not check the City's results by independent CRAC 2 calculations, it does not find them unreasonable. It does not believe that presenting the results in this way gives useful insight, however. For more helpful perspective it, like the Staff, factored in the probability of release category II-T/WW and the probability of the wind blowing towards Philadelphia to calculate the predicted frequency with which various dose levels are exceeded, as follows:

Dose	**Distance**	**Predicted frequency with which dose level is exceeded per reactor-year**
5 rem	21 miles	one chance in 2½ million
30 rem	21 miles	one chance in 5 million
5 rem	30 miles	one chance in 3 million
30 rem	30 miles	one chance in 12 million

These doses would not lead to clinically detectable early effects. Daebeler *et al.*, ff. Tr. 11,114, at 46-47.

F-101. The Applicant also calculates a much smaller number of latent cancer fatalities. City's conversion of 10.5 million person-rem to 4200 such fatalities implies a dose-response relationship of approximately 400 fatalities per million man-rem. *Id.* at 48. The predicted number of latent cancer fatalities is uncertain in the range 10 to 500 cases per million man-rem, with a probable value of 150. Staff Ex. 29 (FES), at p.

5-67. CRAC 2 uses 168 cases per million man-rem, modified by the central estimate, which, generally speaking, reduces the predicted effectiveness of the dose by a factor of 5 for individual doses under 30 rem. App. Ex. 152, at 10-25. Thus, the 10.5 million person-rem would lead to approximately 400 fatalities. These would be spread out over approximately 30 years, at a rate of approximately thirteen per year. This compares with a death rate due to cancer from all causes of approximately 3000 per year for a city of the size of Philadelphia. Furthermore, the 400 latent fatalities must be associated with their frequency of occurrence, 2×10^{-6} (probability of source term) times 0.27 (wind direction) times 0.5 (accounts for the less favorable diffusion conditions) equals 3×10^{-7}, i.e., approximately 1 chance in 3 million. Applicant believes the predicted societal and individual risks within the City of Philadelphia (from severe accidents at Limerick) are very small indeed. Daebeler *et al.*, ff. Tr. 11,114, at 48-49.

F-102. Considerable cross-examination of the Applicant by the City related to the concept of "risk aversion." Specifically, the City asked whether the Applicant agreed that not all people weigh the consequences of accidents equally; that is, they do not give the same weight to an accident involving 10,000 deaths versus 1 death, assuming the same frequency. Applicant thought that people would weigh those things differently. It added that,

> [o]n the other hand, if the frequencies were very low, and here in connection with the kind of large consequences that are considered in probabilistic risk assessments (PRAs) the frequencies are so low as to be almost beneath comprehension of the average person, when you start talking about probabilities of one in a million or one in a billion per year, it's very hard to conceive of what the consequence means, certainly independent of the absolute probability or even with the absolute probability, it's sometimes difficult to conceive of it.

Tr. 11,787-88 (Levine). Asked whether it would be important to disclose those probabilities, separated from, but not isolated from the consequences, Applicant answered, "I don't think you can view them separately. I think you have to view probabilities and consequences jointly, whether it's with an 'and' or with a 'times'." Tr. 11,789-90 (Levine). Applicant agreed that certainly anyone who is rational would view that, at the same frequency, the larger consequence is a more serious event than the smaller consequence. Tr. 11,794 (Levine). *See* our discussion of risk aversion, at the end of this section.

F-103. To the extent that the adequacy of the FES might depend upon explicit disclosure of dose-distance relationships, particularly but not exclusively, for the population of Philadelphia, both the Applicant and the Staff have either provided such information in the record of this proceeding, or described how such information can be derived from the information available either in the FES or the record. In any event, we do not agree that such explicit data are necessary for the purpose of assessing the environmental impact of severe accidents at Limerick. That impact necessarily involves the total population surrounding Limerick, including that of Philadelphia. Average measures of environmental risks are obtained by combining the frequency (likelihood of occurrence) of accidents and their impacts (consequences).[24] Such averages are used as an aid to the comparison of radiological risks associated with the accident releases with risk associated with normal operational releases and with other forms of risk to which the public is exposed. A common way to combine the risk factors is simply to multiply the probabilities by the consequences (as done by both the Applicant and the Staff). The resultant risk is then expressed as a measure of consequences per unit time. Such a quantification of risk does not mean that there is universal agreement that peoples' attitudes about risks, or what constitutes an acceptable risk, can or should be governed solely by such a measure. It can be a contributing factor to a risk judgment, although not necessarily a decisive factor. Staff Ex. 29 (FES), at p. 5-98.

F-104. As an example of the kind of risk comparison made in the FES, it is noted that the largest risk in the entire region surrounding Limerick is associated with latent cancer fatalities (excluding thyroid persons) and is estimated to be 7×10^{-2} per reactor-year. Using the American Cancer Society value for background cancer mortality rate in the U.S., and the year 2000 population estimate within 50 miles of Limerick, it is estimated that there would be 10,000 background cancer fatalities in that year. FES at p. 5-99. Even if the FES estimate were low by a factor of 40 (Tr. 11,286 (Acharya)), and the latent cancer fatality rate were 2.8 per reactor-year, this would be only 2.8×10^{-4} (2.8/10,000) times the background rate. From comparisons like this, in the FES, it is concluded that the risk associated with severe accidents at Limerick is small compared to like risks to which the public is otherwise exposed. Staff Ex. 29 (FES), at pp. 5-98 to 5-99.

F-105. For the reasons discussed above, we find this contention (City-13) without merit.

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[24] This is in accord with the Commission's "Statement of Interim Policy" on severe accident risk analysis. 45 Fed. Reg. 40,101, 40,103, col. 1 (June 13, 1980). It requires that the NEPA analysis of the risks of severe accidents give equal attention "to the probability of occurrence of release and to the probability of occurrence of the environmental consequences of those releases."

a. Risk Aversion

F-106. In its findings and recommendations in the *Indian Point* proceeding, the majority of the Board recommended to the Commission that in assessing societal risk the Commission consider not only expected risks, defined as the arithmetical product of probability and consequences, but also the absolute value of the consequences. *Indian Point,* LBP-83-68, *supra,* 18 NRC at 891. It stated that "[b]y focusing on expected risk values only, we may overlook other important social and ethical considerations." *Id.* at 892. The majority then gave examples of one accident (sequence) with a probability of 1.5×10^{-5} of causing two fatalities (per reactor-year) and another accident (sequence) with a probability of 2×10^{-8} of causing 100,000 fatalities, for (presumably) Unit 1, and one accident (sequence) with a probability of 4×10^{-6} of causing two fatalities and another accident (sequence) with a probability of 10^{-8} of causing 100,000 fatalities, for Unit 2. The risks are 3×10^{-5} and 2×10^{-3} fatalities per reactor-year for Unit 1 and 0.8×10^{-5} and 10^{-3} fatalities per reactor-year for Unit 2. The ratios of the risks for high consequence to low consequence are 0.7×10^{2} and 1.2×10^{2}, respectively, for Units 1 and 2. On this basis the majority suggests that lower risk should be demanded as the potential consequences increase, analogously to insurance companies limiting their liability for very large accidents. Further, it specifically suggests that the Commission should not ignore the potential consequences of severe-consequence accidents by always multiplying those consequences by low-probability values.

F-107. Judge Gleason, in his dissent, referred to the Commission direction that any testimony on accident consequences for Indian Point must include a discussion of the probability of the accidents leading to the proposed consequences. (*See Consolidated Edison Co. of New York* (Indian Point, Unit 2), CLI-82-15, 16 NRC 27, 36-37 (1982).)

F-108. We observe the following: First, the *Indian Point* Hearing was a very special discretionary proceeding, in which the Commission provided specific guidance on the admission of contentions and the formulation of issues for hearing. *Indian Point,* CLI-82-15, *supra.* We do not find this guidance binding on us in consideration of severe accidents under NEPA in this proceeding. Rather, under NEPA and the guidance provided in the Commission's Statement of Interim Policy on Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969 (45 Fed. Reg. 40,101 (June 13, 1980)), we find, first, we must pay approximately equal attention to the probability of occurrence of releases and to the probability of occurrence of the environmental consequences of those releases. *Id.* at 40,103, col. 1. Second, while there may be some emotional appeal to attaching greater significance to the risks of high consequence, it is no less rational to argue that event probabilities of 10^{-8} per reactor-year are so small they may be ignored.

F-109. In any event, we believe the proper approach is to characterize the risk of potential accidents at Limerick as meaningfully as possible and to compare this predicted risk to the actual risk (based on extrapolation of actual experience) to which members of the public are otherwise exposed. Thus, we are led to the value judgment of whether or not a societal gain resulting from the proposed action is acceptable knowing the magnitude of the incremental increase in risk attendant to that action.

9. *City-15: Contamination of City's Water Supplies*

a. *Introduction and Summary*

F-110. As admitted, this contention states that:

> The DES does not adequately analyze the contamination that could occur to nearby liquid pathways, and the City's water supplies sources therefrom, as a result of precipitation after a release. A reasoned decision as to environmental impacts cannot be made without a site-specific analysis of such a scenario.
>
> The DES addresses at great length releases to groundwater (DES at p. 5-34 *et seq.*), but gives only a cursory and conclusory discussion of contamination of open water (DES at 5-33). This issue is of crucial concern here as the two major water bodies at and near the facility are the City's only water supplies. The City also has open reservoirs within its boundaries which could be contaminated through precipitation. For an issue of such great importance, insufficient consideration has been given here. The mandate of NEPA to take a hard look at environmental consequences has been ignored.

F-111. Evidentiary hearings on this contention were held in Philadelphia, Pennsylvania on June 19-20, 1984. Both the Applicant and the Staff provided qualified witnesses and written testimony. The City of Philadelphia (City) cross-examined the witnesses, but provided no witnesses of its own.

F-112. City's contention refers to the cursory and conclusory discussion of contamination of open water in the Staff's Draft Environmental Statement (DES). We note that the Final Environmental Statement (FES) expands the discussion of this subject somewhat, but, in fact, does not provide a site-specific analysis of the environmental impacts of contamination of open water for Limerick. Staff Ex. 29, at pp. 5-92 to 5-93. Both the Applicant and the Staff provide such analyses in their testimony. Bartram *et al.*, ff. Tr. 12,007; Acharya, ff. Tr. 12,141; Wescott and Fliegel, ff. Tr. 12,141; and Lehr, ff. Tr. 12,141. It is the results of these analyses that we examine to determine the adequacy of disclosure and the contribution to risk from this source, in the context of NEPA requirements.

F-113. While the FES discussion of the risk from potential contamination of the Philadelphia drinking water supply resulting from a severe accident at Limerick largely dismisses this risk as being of small importance compared to the risk from radioactive fallout on land (FES at p. 5-93), no site-specific analysis was reported in the FES. In response to the contention both the Staff and the Applicant presented the results of such analyses in testimony. Both parties used probabilistic risk assessment methodology to estimate the probabilities and quantities of release of fission products to the environment. Both parties also used versions of the same computer code to calculate the dispersion and deposition of radioactivity on the ground and open bodies of water below the traveling radioactive plume. The amount of deposition in the Delaware and Schuylkill watersheds was then determined. The concentrations of Sr-90, principally, in Philadelphia's water supply system were then calculated as a function of time. These concentrations (and also those for other nuclides of possible significance, i.e., Cs-137, Cs-134, I-131, I-133) were then compared to (a) Federal and State guidelines for consumption of contaminated drinking water, and (b) the health effects resulting from the airborne pathway for dispersion and deposition of radionuclides. Both the Staff and the Applicant conclude that the risk from the liquid pathway is small compared to the airborne pathway. We concur.

F-114. In addition, the record shows that there are a number of potential countermeasures that could be undertaken to reduce the risk from such a severe accident. These include interdiction and use of alternate sources and modification of water treatment processes to remove radioactivity.

b. Source of Potential Contamination

F-115. Both the Staff and Applicant used probabilistic risk assessment methodology to estimate the probabilities and quantities of release of fission products to the environment as a result of severe accidents at Limerick. For a detailed analysis of liquid pathway contamination, one would use all of the release categories developed in the probabilistic risk assessment. Acharya, ff. Tr. 12,141, at 3. The Staff, however, chose a much simpler and reasonably bounding type of analysis, by selecting only one release category. This category, II-T/WW, whose specifications are listed in the FES Table 5.11C, Staff Ex. 30, and is described in Appendix H of the FES, at H-13, was selected because the quantities of radionuclides in the atmospheric release associated with it are among the highest of all release categories considered in the FES. The probability of this release was artificially assigned as the sum of the probabilities of all release categories, i.e., 9×10^{-5} per reactor-year. Acharya, ff. Tr. 12,141, at 3-4; Tr. 12,147-48, 12,245-46 (Acharya). This accident sequence was selected because it provided the largest combination of probabilities and consequences. Other accidents might give more deposition, but would have a lower probability or, would be of higher probability, but would result in less deposition. Tr. 12,163-64 (Fliegel).

F-116. The Applicant used all of the accident sequences developed in its Severe Accident Risk Assessment ("SARA") to define the radioactive source terms. Bartram *et al.*, ff. 12,007, at 4-5.

c. *Transport of Radioactivity*

F-117. Both the Staff and the Applicant used versions of the CRAC computer code to calculate the dispersion and deposition of radioactivity following an atmospheric release from Limerick. The Staff used CRAC, which has the capability of calculating concentrations of radionuclides deposited on the ground and open water bodies below the traveling radioactive plume, in terms of curies per square meter (Ci/m^2) of the ground surface, due to the effects of dry and wet deposition processes. Acharya, ff. Tr. 12,141, at 4-5. Using actual site meteorological data and ninety-one different accident start times uniformly distributed throughout a 1-year period, the ground deposition of various radionuclides was calculated as a function of distance and direction from the plant site. Sixteen equal sectors and thirty-four spatial intervals extending up to 500 miles from the site were used. *Id.* at 5. The sampling scheme and meteorological data used are the same as used in the Limerick FES for probabilistic analysis of severe accidents. *Id.* at 6. Using the CRAC output and the location of the watersheds relative to the site, the amount of deposition on the watersheds for various wind directions and meteorologic dispersion conditions was determined. Wescott and Fliegel, ff. Tr. 12,141, at 5. The amount of area covered by free water was not considered specifically, because it is a very small percentage of the area of the watershed. Tr. 12,147 (Fliegel).

F-118. The model used for washoff of radionuclides into the Schuylkill and Delaware rivers consists of three terms. One term describes the initial washoff (within a month or two after deposition) as a fraction of the total radionuclide deposited. Another term describes the annual washoff (primarily due to erosion) as a constant fraction of the total radionuclide inventory available for transport during the year. A third term accounts for radionuclide losses such as from radioactive decay. The model is limited to determining radionuclide transport over a period of years. The total washoff, however, is relatively unaffected by changes in the initial washoff coefficient. *Id.* at 7.

F-119. Because of the slow rates of washoff, determined most reliably for the New York City water supply for nuclear weapons fallout, and correlation to the Schuylkill and Delaware River watersheds, only the long-lived isotopes of Strontium-90 and Cesium-137 would contribute significantly to population dose from drinking water. Based on the amount of Cesium-137 released, the appropriate washoff coefficients and dose conversion factors, Cesium-137 would contribute less than 10% to the total dose. Consequently, only Strontium-90 dose estimates were made. Calculations were made assuming no treatment or interdiction of the Philadelphia water supply. *Id.* at 9-10.

F-120. The Schuylkill watershed has an area of almost 1900 square miles at Philadelphia and an average flow of approximately 3000 ft^3/sec. The Delaware watershed has an area of almost 7781 square miles at Philadelphia and an average flow estimated to be more than 12,000 ft^3/sec. Wescott and Fliegel, ff. Tr. 12,141, at 3. The long axis of the Schuylkill Basin runs in a northwest to southeast direction with the farthest point of the watershed approximately 50 miles northwest of the Limerick site. The long axis of the Delaware Basin runs in a north-northeast to south-southwest direction with the farthest point in the watershed about 160 miles north-northeast of the site. Because of the difference in orientation of the watersheds, a wind direction that could cause a high deposition on one watershed generally would preclude a high deposition on the other. *Id.* at 4.

F-121. Each calculated deposition has a probability of occurrence associated with it. By ranking the deposition by magnitude, the Staff determined the probability of nonexceedance for a given deposition and constructed curves of cumulative probability distributions for deposition of Sr-90 on the Schuylkill and Delaware watersheds. *Id.* at 5-6 and Attachment 1. From these curves the Staff determined that there is a 99% chance that less than 160,000 Ci of Sr-90 would be deposited in the Schuylkill watershed and less than 140,000 Ci in the Delaware watershed. *Id.* at 6.

F-122. The Applicant used CRAC 2 to calculate the amount of radioactive material deposited in the Schuylkill and Delaware watersheds for each combination of fission product source term, weather sequence and wind direction. Like the Staff, the Applicant found that Strontium and

Cesium dominated the long-term contamination of ingestion pathways, because of their potentially large release quantities, relatively long half-lives, and recognized radiotoxicity. In consideration of population doses arising from drinking of contaminated water in the short term (e.g., 1 month), other radionuclides, such as Iodine-133 and -131 were included. Bartram *et al.*, ff. Tr. 12,007, at 3-4. The results, expressed as Ci/m^2, together with information on the plume width as a function of distance downwind, are used in the computer code LIQPATH to calculate the total amount of Strontium and Cesium deposited in the two watersheds, including that deposited directly in the rivers. *Id.* at 5. LIQPATH also predicts the subsequent temporal variation of the concentration of each radionuclide. Physical phenomena modeled include radioactive decay, runoff, erosion, groundwater transport, sediment scavenging, and possible removal of radionuclides by water treatment systems. *Id.* at 5-6.

d. Potential Consequences

(1) STAFF ANALYSIS

F-123. To estimate the potential consequences of a II-T/WW release to the Philadelphia water supply and potential health effects, the Staff made a number of calculations, assumptions and observations. First, they constructed curves of the concentration of Sr-90 in the Schuylkill and Delaware rivers for the first year after the release as a function of nonexceedance probability. Wescott and Fliegel, ff. Tr. 12,141, at 10 and Attachment 3. From these curves, and the maximum permissible concentration (MPC) of Sr-90 permitted to be discharged to unrestricted areas, 300 picocuries per liter (pCi/l) (10 C.F.R. Part 20, Appendix B, Table II), it is determined that the Schuylkill River is likely to be highly contaminated. There is only a 2% chance that the Delaware would be above the MPC, a 38% chance of no Sr-90 and a 50% chance of less than 15 pCi/l of Sr-90. Thus, it is highly probable that the Delaware would remain a safe drinking water source after the release. *Id.* at 10-11. With respect to the Schuylkill, the Staff constructed curves of the cumulative probability distribution of time after the release for the Schuylkill River to reach the MPC and 1/3 MPC. *Id.* at 11 and Attachment 4. From these curves it was determined that there is a 50% probability that the Sr-90 concentration would be reduced to the MPC in 1 to 2 months. For the most severe cases, it could take as long as 20 years to reach MPC and 53 years to reach 1/3 MPC. *Id.* at 11.

F-124. The radiation dose to the population using the Philadelphia drinking water system would depend upon the concentration limit for

Sr-90 chosen for permitting consumption. For illustration, the Staff calculated the annual dose to people ingesting water at MPC, 1/3 MPC and at 8 pCi/l. The results were as follows:

	MPC	1/3 MPC	8 pCi/l
Person-rems (whole body)	1.6×10^5	6.4×10^4	5×10^3
Person-rems (bone)	7.2×10^5	2.4×10^5	1.9×10^4

F-125. Similarly, the Staff calculated the long-term residual doses to people from ingesting water after it has receded to the same concentrations, as follows:

	MPC	1/3 MPC	8 pCi/l
Person-rems (whole body)	5.4×10^6	1.8×10^6	1.4×10^5
Person-rems (bone)	2.2×10^7	7.2×10^6	6×10^5

Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 C.F.R. Part 50, Appendix I," Rev. 1, October 1977, was used in making these calculations. *Id.* at 11-14.

F-126. Deposition of radionuclides on open water bodies could result in immediate contamination, but the total amount of radioactivity entering the water supply in this manner would be very small compared to that entering the water supply as washoff from the upstream watersheds. Since Philadelphia is located such that a heavy deposition on the reservoirs within the City is not likely to coincide with high concentrations in the Schuylkill or Delaware Rivers, the replacement of contaminated reservoir water with relatively clean water prior to residential distribution would be expected. *Id.* at 15.

F-127. With respect to consequences for time periods less than 1 year, the Staff did a worst-case analysis for only the Schuylkill River, since its flow is lower than the Delaware and concentrations of Sr-90 would therefore be higher. The deposition of 162,000 Ci of Sr-90 was assumed, although there is a probability of less than 1% that all of this would be deposited within the basin (there is a 50% probability that less than half of this quantity would be deposited in the basin). The Staff also considered a number of additional cases. First it considered situations with average Schuylkill River flow and 2% Sr-90 runoff. This runoff is consistent with measured data as a result of fallout from atmospheric weapons testing. Runoff was considered to occur in time periods of a day, a week and a month. The resulting concentrations ranged from less than 15,000 pCi/l for runoff in a month to about 440,000 pCi/l for runoff in a day. *Id.* at 15-18. For time periods less than a day, the entire Schuylkill drainage system would not have time to transmit flow and contaminants downstream to the point of interest. The high-runoff scenario would flush a relatively large fraction of the radionuclides from the river system during a short period of time when, almost certainly, drinking water would not be withdrawn from the river. Since a smaller percentage of radionuclides would remain — after high runoff — the total long-term population dose would be reduced. *Id.* at 18-20.

F-128. The Staff conservatively estimated the risk of population exposure from contaminated Philadelphia drinking water by multiplying the probability of all release categories (9×10^{-5} per reactor-year) times the consequences of residual population exposures for all time following the reduction of Sr-90 concentrations to 8 pCi/l (it being assumed that no consumption of water above this level would be permitted). Radiation doses associated with drinking water for a year at this contamination level would not result in early health effects. The risk of latent cancer fatalities over all time was estimated to be eight cases, excluding bone cancer, or at a rate of about 7×10^{-4} per reactor-year. The risk of bone cancer fatalities was estimated to be four cases, or at a rate of about 5×10^{-4} per reactor-year. This total rate of 1.2×10^{-3} latent cancer fatalities per reactor-year was considered small compared to the estimate of 9×10^{-3} latent cancer fatalities per reactor-year resulting from the air and ground pathway results derived from Fig. 5.4*l* of the FES. Acharya, ff. Tr. 12,141, at 13-14.

(2) APPLICANT ANALYSIS

F-129. The Applicant's analysis of the consequences of contamination of the Philadelphia water supply considered the potential health effects by developing a complementary cumulative distribution function for whole-body dose resulting from contamination of the drinking water supply by Cesium-134, Cesium-137, Strontium-89, Strontium-90 and Iodine-131. The bases for its analyses included the following: Doses to the population resulting from water used outside the body were not considered since they would make a very small contribution to total exposure; time-dependent calculations of the concentrations of Cesium and Strontium nuclides in the river water were used; the population was assumed to consume the river water for 50 years; population doses were calculated using the methods of NRC Regulatory Guide 1.109, as implemented in the LADTAP computer code (App. Ex. 167; App. Ex. 168), with one exception; more recent dose conversion factors recommended by the ICRP were used, to be consistent with the analysis of ingestion pathways used in Applicant's SARA. Bartram *et al.*, ff. Tr. 12,007, at 11-12.

F-130. Specific calculations were made for both the Schuylkill and Delaware Rivers, since the proportions of radionuclides would differ and because the Schuylkill would likely be more heavily contaminated than the Delaware. It was assumed that, in an emergency, 93% of the City's population would be served by the Delaware and 7% by the Schuylkill. According to the City, the Baxter plant, which takes water from the Delaware, could supply all of the City's needs except for the Roxborough High Service District, which constitutes approximately 7% of the needs. *Id.* at 12.

F-131. The calculations made on the basis of Strontium and Cesium contamination lead to the estimates of chronic or long-term contributions to population dose. To take into account more short-lived radionuclides, such as radioiodine, a simplified, bounding calculation was made. For each source term, weather sequence and wind direction, the isotopes of Iodine deposited on the watersheds were assumed to pass into the rivers immediately, at a rate approximately 50 times that for Strontium (2% of the Strontium is expected to pass directly into the rivers). The resulting increment in population dose was included in the CCDF for population dose. *Id.* at 13. A further contribution to the total CCDF for population dose was calculated for the potential contamination of the City's raw and finished water basins (reservoirs) even though in practice, much of this water could be disposed of. *Id.* at 13-14.

F-132. The area under the overall CCDF curve provides an estimate of radiation risk from drinking water contamination of 0.67 person-rem per reactor-year. The three contributors are 0.49, 0.16 and 0.02 person-rem per reactor-year from iodine deposited on the watershed, Strontium and Cesium deposited on the watershed and direct deposition into the system, respectively. This contribution to radiation risk, 0.67 person-rem per reactor-year, may be compared to the radiation risk, 70 person-rem per reactor-year, estimated by the Applicant in its SARA for the airborne pathway. Whereas airborne pathway analyses routinely assume protective actions such as interdiction of milk and crops and decontamination of land, the Applicant did not consider some possible countermeasures with respect to the drinking water pathway (discussed below) in the above comparison. *Id.* at 14-15.

F-133. To assess the significance of the person-rem per reactor-year estimates, it would be possible (as the Staff did), on the basis of these results, to estimate early and late health effects. Also (as both the Staff and Applicant did), one may compare the estimated concentrations of nuclides with Federal and State guidelines for consumption of contaminated drinking water. The applicable guides (regulation, in the case of 10 C.F.R. Part 20) are listed in Table 1, below.

TABLE 1
Protective Action Guides for Drinking Water Concentrations (pCi/l)

	Sr-90	Cs-137	Cs-134	I-131	I-133
10 C.F.R. Part 20, Appendix B, Table II	3×10^2	2×10^4	9×10^3	3×10^2	1×10^3
PEMA[25] — uncontrolled discharges to surface water and in circumstances where the water supply is influenced by contaminated runoff and fallout — exposure time not to exceed 1 year	9.6×10	2.4×10^3	2.4×10^5	3.6×10	1.2×10^2
PEMA — acute crisis conditions where no other water supply is available — exposure time not to exceed 30 days	8×10^3	2×10^5	2×10^7	3×10^3	1×10^4

Bartram *et al.*, ff. Tr. 12,007, Table 1.

F-134. The Commonwealth of Pennsylvania Emergency Management Agency (PEMA) Protective Action Guides (PAGs) are based on the U.S. Environmental Protection Agency (EPA) National Interim Drinking Water Regulations, EPA-570/9-76-003, Appendix B. The NRC regulation, 10 C.F.R. Part 20, Appendix B, Table II, applies to the maximum permissible concentrations in effluents to unrestricted areas. 10 C.F.R. § 20.106(a). The PEMA PAG for uncontrolled discharges to surface water, and in circumstances where the water supply is influenced

[25] Pennsylvania Emergency Management Agency.

by contaminated runoff and fallout, the U.S. EPA Appendix B concentrations multiplied by 12 will apply — assuming that the exposure time will not exceed 1 year. The associated dose commitment to any organ is 50 millirem. Bartram *et al.*, ff. Tr. 12,007, at 16. For the acute crisis conditions, where no other water supply is available and the duration is less than 30 days, the average concentration may reach 1000 times the U.S. EPA Appendix B concentrations. The associated dose commitment to any organ is 330 millirem. *Id.* at 16-17.

F-135. The probability that the PAGs would be exceeded may be determined by use of the Applicant's CCDF curves. For example, considering Sr-90 as the principal contributor to the long-term accumulation of radiation dose and the PAG for circumstances in which the water supply is influenced by contaminated runoff and fallout, i.e., 96 pCi/l averaged over 12 months, the probability of exceedance in the Schuylkill is 1 in 300,000 per reactor-year. *Id.* at 17 and Fig. 4(a). The corresponding probability for the Delaware is 1 in 7 million per reactor-year. *Id.* at 17 and Fig. 5(a). Similarly, it may be determined, for the same circumstances, that the probability of exceeding the radiocesium PAG is less than 1 chance in a billion per reactor-year. *Id.* at 18.

F-136. For the short term, the 1-month PAG for Sr-90, 8×10^3 pCi/l, would apply. Considering Sr-90 alone, the probability of exceedance is approximately 1 in 3 million per reactor-year in the Schuylkill and less than 1 in 1 billion per reactor-year in the Delaware. For the short term, however, other radionuclides, such as I-131 cannot be neglected. Using the simplified, bounding calculation for Iodine deposition described above, the probability of exceedance would be approximately 1 in 100,000 per reactor-year in the Schuylkill and approximately 1 in 150,000 per reactor-year in the Delaware. *Id.* at 19.

F-137. None of the above estimates take into account the possibility of countermeasures, except for the assumption that the use of the Delaware River was maximized to supply the water needs of the City of Philadelphia.

e. Potential Countermeasures

F-138. Following potential contamination of the Philadelphia water supply, a number of potential countermeasures could be undertaken to reduce the risks presented by such an accident. Such countermeasures, depending on the nature and level of contamination, location and timing, could include interdiction (e.g., by bypassing a reservoir and using alternative sources), modification of water treatment processes (e.g., use of activated charcoal to reduce iodine content, use of a lime-soda softening process to remove strontium). Bartram *et al.*, ff. Tr. 12,007, at 21-25; Lehr, ff. Tr. 12,141, at 13. In this decision we do not discuss possible countermeasures from the point of view of offsite emergency planning, or in the detail necessary for that subject. That matter is a subject for future hearings. Our discussion here is simply to provide some perspective on the potential to reduce the risk from contaminated drinking water in the event of a low-probability, severe accident at Limerick. Whether the potential is realized could depend on emergency preparedness measures.

F-139. Approximately half of the City's water requirement is supplied by the Delaware River and half by the Schuylkill River. All water withdrawn by the City from the Delaware is treated at the Samuel S. Baxter Plant. Water withdrawn from the Schuylkill is treated either at the Queen Lane Plant or the Belmont Plant. The Queen Lane Plant is located on the east side of the Schuylkill and the Belmont Plant is located on the west side of the river. All withdrawal locations are within the city limits. Lehr, ff. Tr. 12,141, at 3. The City Water Department distributed an average of approximately 345 million gal/day to 1.69 million people and to industry within the City Limits in 1982. An additional 11 million gal/day were distributed for use in lower Bucks County. *Id.* at 3-4. The total filtered water storage capacity of the system was approximately 1.1 billion gallons in 1982. Plant retention capacity of untreated and in-process water in 1982 was 86 million gallons at the Belmont Plant, 201 million gallons at Queen Lane Plant and 216 million gallons at the Baxter Plant, for a total of 503 million gallons. *Id.* at 4.

F-140. The Baxter Plant normally provides water to the area of the City east of Broad Street (and east of the Schuylkill). The Queen Lane Plant normally serves the area west of Broad Street and east of the Schuylkill. The Belmont Plant serves the area of the City west of the Schuylkill. Flexibility exists in the system such that the entire City area, except for an area west of the Schuylkill known as the "Belmont High Service District," may be served by the Baxter Plant (Delaware River water), provided it is fully available, based on average daily demand. The demand of the Belmont High Service District is about 12 million gal/day (i.e., approximately 3% of total daily demand). *Id.* at 4-5.

F-141. To adjust the valve line-ups from the normal situation to use the full capacity of the Baxter Plant could be done in 24 hours. Tr. 12,113 (Guarino). The water system has covered filtered water storage facilities with approximately 2 days supply of water (at normal usage rate). Bartram *et al.*, ff. Tr. 12,007, at 22. The City has the authority to limit the use of water in its system and in an emergency situation should be able to cut water consumption by more than 50% and would have the ability to make sure that the industries that use a tremendous amount of water would be shut down. Tr. 12,113-13 (Guarino).

F-142. Trucking of drinking water is an option for an alternate source (e.g., to the Belmont High Service District). Assuming a need for approximately a gallon per day per person for 100,000 people would require approximately fifty truckloads, which is not a large number. Tr. 12,126-27 (Schmidt).

F-143. The decontamination factor provided by current drinking water treatment processes can be anticipated to be no more than 2 (i.e., 50% removal) for total radioactivity, and less than that for dissolved Strontium, Cesium and Iodine. The addition of activated carbon prior to flocculation would give a decontamination factor for iodine of from 4 to 5. Adding a layer of activated carbon to the surfaces of the sand filters would provide an additional factor of 2, for a total decontamination factor of from 8 to 10. Bartram *et al.*, ff. Tr. 12,007, at 23-24. Decontamination factors for Strontium of from 5 to 10 can be obtained by co-precipitation with dosages of soda ash (sodium carbonate). Additional decontamination could be achieved by repeating the process, albeit reducing the throughput, in the absence of construction of a major plant addition. *Id.* at 24-25. *See also* Lehr, ff. Tr. 12.141, at 8-13.

f. Conclusion

F-144. We do not conclude that specific countermeasures would or could be implemented, nor what quantitative reductions in risk could be achieved. We do conclude that a number of alternatives to consumption of contaminated drinking water could be considered should the City of Philadelphia water supply become contaminated. These alternatives include water rationing, use of stored or bottled water, construction of temporary or permanent pipelines from the points of use to a safe and adequate supply, dilution by a known safe water supply, delivery of safe water by auxiliary means (e.g., tank truck) or use of special decontamination equipment or procedures. Lehr, ff. Tr. 12,141, at 13.

F-145. We do conclude that the record before us, which supplements the FES, does adequately consider and analyze the contamination that could occur to nearby liquid pathways and the City's water supply sources therefrom, as a result of precipitation after a release (from a severe accident at Limerick). This includes consideration of the City's only two water supplies (the Delaware and the Schuylkill) and the open reservoirs within the City boundaries.

F-146. For the reasons given above, this contention requires no further relief.

III. CONCLUSIONS OF LAW

In reaching this decision, the Board has considered all the evidence submitted by the parties and the entire record of this proceeding. That record consists of the Commission's Notice of Hearing, the pleadings filed by the parties, the transcripts of the hearing, and the exhibits received into evidence. All issues, arguments, or proposed findings presented by the parties, but not addressed in this decision, have been found to be without merit or unnecessary to this decision. Based upon the foregoing Findings which are supported by reliable, probative, and substantial evidence as required by the Administrative Procedure Act and the Commission's Rules of Practice, and upon consideration of the entire evidentiary record in this proceeding, the Board, with respect to the issues in controversy before us;

CONCLUDES that the Applicant, Philadelphia Electric Company, has fully met its burden of proof on each of the contentions decided in this P.I.D. As to these issues, there is reasonable assurance that the Limerick Generating Station, Units 1 and 2, can be operated without endangering the health and safety of the public, and further that all requirements applicable to these issues under the National Environmental Policy Act have been met.

IV. ORDER

WHEREFORE, in accordance with the Atomic Energy Act of 1954, as amended, and the rules of the Commission, and based on the foregoing Findings of Fact and Conclusions of Law, IT IS ORDERED THAT:

The Director of Nuclear Reactor Regulation is authorized, upon making the findings on all applicable matters specified in 10 C.F.R. § 50.57(a), as to each respective reactor unit, to issue to the Applicant, Philadelphia Electric Company, a license or licenses to authorize low power testing (up to 5% of rated power of each unit) of the Limerick Generating Station, Units 1 and 2.

Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this Partial Initial Decision shall become effective immediately. It will constitute the final decision of the Commission forty-five (45) days from the date of issuance, unless an appeal is taken in accordance with 10 C.F.R. § 2.762 or the Commission directs otherwise. *See also* 10 C.F.R. §§ 2.764, 2.785 and 2.786.

Any party may take an appeal from this decision by filing a Notice of Appeal within ten (10) days after service of this Partial Initial Decision. Each appellant must file a brief supporting its position on appeal within thirty (30) days after filing its Notice of Appeal (forty (40) days if the Staff is the appellant). Within thirty (30) days after the period has expired for the filing and service of the briefs of all appellants (forty (40) days in the case of the Staff), a party who is not an appellant may file a brief in support of or in opposition to the appeal of any other party. A responding party shall file a single, responsive brief *only* regardless of the number of appellants' briefs filed. (*See* 10 C.F.R. § 2.762).

IT IS SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

Lawrence Brenner, Chairman
ADMINISTRATIVE JUDGE

Dr. Richard F. Cole
ADMINISTRATIVE JUDGE

Dr. Peter A. Morris
ADMINISTRATIVE JUDGE

Bethesda, Maryland
August 29, 1984

[Appendices A and B have been omitted from this publication, but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Peter B. Bloch, Chairman
Dr. Oscar H. Paris
Frederick J. Shon

In the Matter of **Docket No. 50-155-OLA**
(ASLBP No. 79-432-11-LA)

CONSUMERS POWER COMPANY
(Big Rock Point Plant) **August 29, 1984**

In this Initial Decision, the Licensing Board authorizes the Licensee to add three more spent fuel racks to its spent fuel pool, expanding its capacity from 193 spent fuel assemblies to 441 assemblies upon the condition that the plant will not be operated should the heat load from the fuel and the temperature of the nearby lake prevent the Licensee from assuring that the makeup line to its pool will be able to keep the bulk pool temperature below 150°F. The Board also requires that there be a human factors analysis of the meter for the noble gas stack monitor and that Licensee advise emergency planning authorities to consider practicable means of improving emergency evacuation at time of a major event at the Castle Farms site.

TECHNICAL ISSUES DISCUSSED

Spent Fuel Pool Inside Containment (Makeup Water Line)
Temperature Analysis of Spent Fuel Pool
Zircaloy/Steam Reaction in Spent Fuel Pool

Concrete Integrity in Spent Fuel Pool
Radiation Exposure from Spent Fuel
Aircrash Risks
Seismic Stability of Gantry Crane
Emergency Planning, Size of EPZ
Radiation Monitoring
Emergency Planning, Summer/Winter Conditions
Emergency Planning, Children and Pregnant Women.

APPEARANCES

Joseph Gallo, Esq., and **Rob Fitzgibbons, Esq.**, Isham, Lincoln and Beale, Washington, D.C.; **Phillip Steptoe, Esq.**, **Peter Thorton, Esq.**, and **Victor Copeland, Esq.**, Isham, Lincoln and Beale, Chicago, Illinois, for Consumers Power Company.

Judd Bacon, Esq., for Consumers Power Company.

Ms. Joanne Bier, *pro se,* Intervenor.

Ms. Christa-Maria, *pro se,* Intervenor.

Mr. Jim Mills, *pro se,* Intervenor.

Mr. John O'Neill, *pro se,* Intervenor.

Spence Perry, Esq., for the Federal Emergency Management Agency.

Richard Goddard, Esq., and **Richard Bachman, Esq.**, for the United States Nuclear Regulatory Commission Staff.

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INITIAL DECISION
(On All Remaining Issues)

The Big Rock Point Plant, which may expand its spent fuel pool storage capacity as the result of this decision, was constructed and licensed to operate under regulations that were less stringent than those now in effect. This change in regulatory requirements caused the parties to inquire in depth into issues that would not be relevant were a more modern plant to seek to enlarge its spent fuel pool.

In granting the license amendment, subject to a few appropriate conditions, the Board is grateful to the intervenors for their volunteer efforts, which improved the safety of the plant by: (1) contributing to a more thoughtful offsite emergency plan for the plant, (2) assisting Applicant and Staff to decide to install a reliable makeup line that will prevent the heating up of the pool should there be a TMI-2-type accident,[1] (3) requiring a seismic analysis of the gantry crane, which was consequently modified to reduce the risk that the crane may fall into the pool during a seismic event, and (4) emphasizing to military authorities the need to avoid overflights of the plant in order to reduce the risk of aircrash.

Procedural History

This proceeding began on July 23, 1979 with publication of a notice in the *Federal Register* (44 Fed. Reg. 43,126) of the NRC Staff's proposal

[1] The spent fuel pool at the Big Rock Point Plant is located inside the containment building, so that an accident that prevents access to containment would prevent maintenance personnel from attending to a malfunctioning spent fuel pool circulating pump.

to issue an amendment to the operating license for the Big Rock Point Plant. The proposed amendment would permit the owner of Big Rock Point, Consumers Power Company ("Licensee"), to expand the capacity of the Plant's spent fuel pool, which is presently licensed to store 193 spent fuel assemblies. The Licensee would be able to store 441 assemblies under the proposed action. The additional capacity would be accomplished by adding three more spent fuel racks to the pool.

The July 23, 1979 notice provided an opportunity for interested members of the public to request a hearing. Three petitions to intervene were received. After the completion of the procedural steps required by NRC regulations, a special prehearing conference was held on December 5, 1979. Upon consideration of the legal arguments presented at the prehearing conference, the presiding Atomic Safety and Licensing Board ("Licensing Board") issued an order on January 17, 1980 (LBP-80-4, 11 NRC 117), which accepted two and rejected one of the petitions.

Ms. Christa-Maria, Ms. Joanne Bier and Mr. Jim Mills ("Christa-Maria") were admitted as a single-party Intervenor. Mr. John O'Neill ("O'Neill") was admitted as a separate party. Several contentions of both parties were admitted as issues in controversy, except that some contentions, e.g., Christa-Maria Contention 9 concerning emergency planning was admitted for discovery purposes only. The intervention petition of Mr. John A. Leithauser was denied for his failure, despite ample opportunity provided by the Licensing Board, to perfect the petition as required by NRC regulations. LBP-80-4, *supra*.

Thereafter, the proceeding lagged because NRC Staff engineering resources had other priorities, resulting from the accident at Three Mile Island. For this reason the technical review of Licensee's application to expand the Big Rock spent fuel pool was held in abeyance during most of 1980. The NRC Staff ultimately issued its Safety Evaluation Report and Environmental Impact Appraisal on May 15, 1981. Both documents supported the proposed expansion action.

The delay in the proceeding afforded the parties ample opportunity for discovery. The answers to interrogatories and documents furnished by Licensee and the NRC Staff to the Intervenors occupy more than 10 feet of shelf space. Additionally, the Intervenors answered interrogatories posed by the Licensee and NRC Staff.

On October 5, 1981, the Licensee and the NRC Staff filed motions for summary disposition on all but two of the Intervenors' contentions. Replies in opposition to the motions were filed by Christa-Maria and by O'Neill. The Licensing Board ruled on February 19, 1982, and granted the motions in part and denied them in part. LBP-82-8, 15 NRC 299 (1982).

The Licensing Board's February 19 Order reformulated some of the contentions and admitted genuine issues of fact that emerged from the summary disposition filings. The contentions and issues set forth for litigation were as follows:

1. Christa-Maria Contention 8 and O'Neill Contention III.E-2,

> The occurrence of an accident similar to TMI-2 which would prevent ingress to the cr ıtainment building for an extended period of time would render it impossible to m ntain the expanded spent fuel pool in a safe condition and would result in a signif. ·ntly greater risk to the public health and safety than would be the case if the increase storage were not allowed.

This con. ·lidated contention was limited by the Licensing Board's February 19 Order :o the following genuine issues of fact (15 NRC at 312):

> (1) How reliable is the remotely activated makeup water system which will be added to the spen' fuel pool? How reliable does it need to be? How many gallons per minute will it be able to make up?
>
> (2) How reliable are the spent fuel pool water level monitors which applicant is planning to install? Is applicant required to install and maintain these monitors?
>
> (3) Are motor-operated valves MO-7064 and 7068 necessary to control containment pressurization? Are they qualified for high temperature and high humidity?
>
> (4) Will Zircaloy react with steam in a fuel pool which is boiling because its cooling system has failed? Will the reaction become self-sustaining?
>
> (5) Is the concrete in the fuel pool strong enough to resist a temperature of 247°F and point loading from the storage racks?

2. Christa-Maria Contention 2:

> The increase in fuel stored in the Big Rock pool will result in an increase in the amount of radiation released to the environment at the south wall of the storage pool where there is less shielding, according to the licensee's Description and Safety Analysis. This increment in the level of radiation released to the environment enhances the risks to the health and safety of the public in the vicinity of the plant.

O'Neill Contention II.A:

> The routine releases of radioactivity during the installation of new racks, the loading of those racks, and storage of fuel in the racks will exceed the exposure of workers, as will the releases of radioactivity through the south wall of the pool exceed the limits imposed by Appendix I to 10 C.F.R Part 50 on exposure to the general public.

These contentions were limited by the Licensing Board's February 19 Order to the following genuine issues of fact (15 NRC at 321-22):

(1) What caused the discrepancy between staff and applicant statements about the relevant dimensions of the south wall of the spent fuel pool and what effect, if any, has this discrepancy had on radiation calculations?

(2) What is the combined radiation from the pool and filter sock tank?

(3) What point on the south wall was used as a reference point for calculating dose estimates?

(4) What is the reason that applicant stated that it used "mass absorption coefficients in radiation estimates when it apparently used linear absorption coefficients?"

(5) What was the location and reference level to which staff applied the inverse square rule to calculate offsite doses?

(6) What hiring, training and supervision methods and what health physics safeguards will be used during the installation of the new fuel rack?

(7) What has applicant done to correct alleged health physics deficiencies identified by the Institute of Nuclear Power Operations in its August 1981 report?

(8) To what extent will the radwaste demineralizer be employed on a continuing basis to attenuate radiation from the spent fuel pool?

3. O'Neill Contention II.D:

> The licensee has not adequately provided for the protection of the public against the increased release of radioactivity from the expanded fuel pool as a result of the breach of containment due to the crash of a B-52 bomber.

Summary disposition was denied with respect to this contention; the Board listed eleven areas of genuine dispute as reasons for the denial. (15 NRC at 327-29.)

4. O'Neill Contention II.C:

> Licensee's plan, which provides for makeup water to replace water being lost from the pool at rates of up to 200 gallons per minute, is deficient because it does not consider the impact of the lost water on health and safety or the environment.

The Board found that this contention did not raise any genuine issues of fact and granted a motion for summary disposition of the contention as worded. But the Board identified genuine issues of fact raised by information obtained by Intervenors in the course of discovery. Consequently it admitted under II.C the following reworded contention (15 NRC at 331):

> Is the spent fuel pool safe from a rupture which might be caused by a drop of a spent fuel transfer cask or of the overhead crane?

The Board explained this decision as follows:

> The genuine issues of fact under this contention are whether the overhead crane used for handling fuel assemblies and casks is seismically safe, whether the threading on the fire water system is seismically safe, and whether it is necessary for the safety of the enlarged spent fuel pool that 200 gallons per minute of makeup water

be available to protect the pool from the consequences of a drop of a spent fuel transfer cask.

(15 NRC at 332.)

5. O'Neill Contention II.E-3:

The application has not adequately analyzed the possibility of criticality occurring in the fuel pool because of the increased density of storage without a gross distortion of the racks.

Summary disposition of this contention was denied because of questions about the adequacy of the analyses performed by Consumers Power and the NRC Staff. LBP-82-7, 15 NRC 290 (1982). Further questions were raised subsequently because of an affidavit filed by an Intervenor. LBP-82-8, *supra*, 15 NRC at 332-33.

6. Summary disposition was granted with respect to Christa-Maria Contention 3 and O'Neill Contention I.B-3 subject to the submission by Licensee of a clarifying affidavit (15 NRC at 326). The affidavit of Mr. A. John Birkle was filed in response to this directive.

7. Motions for Summary Disposition were *not* filed with respect to O'Neill Contentions II.E-4 and II.G(a), which state respectively:

In the event of an accident which results in a substantial release of radioactivity from the expanded fuel pool, the containment building does not provide adequate shielding to protect the public health and safety.

Administrative controls proposed to prevent a cask drop over the pool are inadequate. These are mentioned on pages 4-9 of the application. Administrative controls have proved inadequate in the past in preventing incidents and are frequently violated at the plant.

8. Christa-Maria Contention 1 involved the adequacy of the NRC Staff's Environmental Impact Appraisal and its consideration of alternatives to the expansion of the Big Rock Point spent fuel pool.

9. Christa-Maria Contention 9, as admitted by the Licensing Board, raised several subcontentions concerning emergency planning. (*See* LBP-82-32, 15 NRC 874 (1982).)

Evidentiary hearings were held during June 1982 with the expectation that litigation of all issues would be completed and the hearing record closed. However, a continuing disagreement between the NRC Staff and the Licensee over the adequacy of Licensee's structural analysis of the concrete pool culminated with the withdrawal by the Licensee of its prefiled testimony on this issue, a genuine issue of fact under O'Neill Contention III.E-2. Licensee could provide no estimate at the time as to when new testimony on this issue would be submitted for the record.

This circumstance together with the realization that the time set aside for the June 1982 hearings was insufficient to litigate all of the issues resulted in the record being closed with respect to only a few of the issues. Thereafter the Licensing Board issued a number of partial initial decisions. Thus, some of the issues litigated in June 1982 have been decided, while others are pending decision and one is subject to a remand order.

The record on two of the five genuine issues of fact under Christa-Maria Contention 8 and O'Neill Contention III.E-2 was closed in June 1982. These two issues, concerning the reliability of the spent fuel pool water level monitors and motor-operated valves MO-7064 and MO-7068, and the remaining three issues — makeup line reliability, zircaloy/steam reaction and concrete integrity — are decided below.

The record on the cask drop issue under O'Neill Contention II.C was closed in June 1982. This issue and the remaining issue concerning the seismic stability of the overhead crane located inside containment are decided below.

A partial initial decision was issued on O'Neill Contention II.E-3. (*See* LBP-82-97, 16 NRC 1439 (1982).) This decision was adverse to the Licensee and an appeal was taken to the Atomic Safety and Licensing Appeal Board, which reversed and remanded our decision. (ALAB-725, 17 NRC 562 (1983).) The Appeal Board directed us to consider the adequacy of a makeup line to the fuel pool for the purpose of keeping the spent fuel pool full and thus avoiding criticality.

Partial initial decisions favorable to the Licensee were issued on O'Neill Contention II.G(a) and Ms. Christa-Maria's environmental contentions. (*See* LBP-82-77, 16 NRC 1096 (1982), and LBP-82-78, 16 NRC 1107 (1982).)

The Licensing Board's two partial initial decisions on Christa-Maria Contention 9, concerning emergency planning, required the submission of further evidence by the Licensee. Our decision of September 14, 1982 held that Licensee had failed to sustain the burden of proof with respect to subcontention 9(2) concerning the training of public officials, 9(4) concerning assistance to persons without vehicles, and 9(5) concerning a current list of invalids. Subcontention 9(7) was dismissed. In due course the Licensee provided substantial additional evidence which, after receipt of the views of the NRC Staff and the Intervenors, was carefully examined by the Board. The foregoing subcontentions were dismissed by our Supplementary Initial Decision, LBP-83-44, 18 NRC 201 (1983).

Finally, the Board's August 6, 1982 Initial Decision concerning the emergency planning pamphlet (subcontentions 9(2) and 9(3)) required

Licensee to provide additional evidence on the manner and method for notifying transients in the Big Rock Point plume exposure pathway emergency planning zone of the existence of the Big Rock emergency plan and its protective measures. LBP-82-60, 16 NRC 540 (1982). This information was submitted under affidavit by the Licensee on September 2, 1983. This matter is decided below.

Subsequent to the June 1982 hearings, the Licensee and the NRC Staff engaged in a lengthy dialogue concerning the issues involving the integrity of the concrete pool and the seismic stability of the overhead crane. These matters were ultimately concluded to the Staff's satisfaction by early Fall 1983. These issues and all other outstanding issues were litigated during a hearing beginning on October 25 and ending on November 4, 1983. The parties submitted findings of fact and conclusions of law as required by the Commission's regulations. Our decision follows.

I. CHRISTA-MARIA CONTENTION 8 AND O'NEILL CONTENTION III.E-2 – RELIABILITY OF MAKEUP WATER SYSTEM

A. Background

Christa-Maria Contention 8 and O'Neill Contention III.E-2 postulate the occurrence of an accident, similar to the TMI-2 accident, which would prevent entry to containment for an extended period. The Licensee's and NRC Staff's motions for summary disposition of this contention were granted in part. However, five genuine issues of fact were set for trial. We deal with two of those issues in this opinion. Our decision on these issues is determinative of the merits of the contention.

The primary issue concerns the reliability of Licensee's remotely activated makeup water system, which was designed and installed as a means of providing makeup water to the spent fuel pool without the need of entering the Big Rock Point Containment in the event of a TMI-2-type accident. A second issue is whether a zircaloy/steam reaction might occur as a result of a loss of pool cooling. (Memorandum and Order, February 19, 1982 (LBP-82-8, *supra*).)

These genuine issues of fact were originally litigated in June 1982. However, by letter of September 9, 1982, Licensee informed the Board and the parties that it intended to alter the design of the makeup pipe to provide more makeup water, thereby keeping the pool structure cooler and making it possible to demonstrate the adequacy of the concrete by a structural analysis. The reliability and zircaloy/steam reaction issues were therefore taken up again in October 1983, during the hearings in Petoskey, Michigan.

B. Applicable Law

The remotely actuated makeup line is an engineered safety feature which was installed at Big Rock Point to ensure the adequacy of fuel storage and to protect the health and safety of the public during postulated accident conditions. 10 C.F.R. Part 50, Appendix A, General Design Criteria 61 and 62. The makeup line system, if demonstrated to be reliable, would satisfy this objective by maintaining an average bulk pool temperature at or below 150°F, thereby ensuring, as explained below, the integrity of the concrete pool structure. In addition, the 150°F temperature limitation would maintain the normal inventory of water in the pool,[2] thereby providing adequate cooling for the spent fuel and avoiding a containment repressurization scenario or an unanalyzed criticality and zircaloy/steam reaction scenario.

Although it was promulgated after the Big Rock Point Plant was constructed and licensed to operate, we consider the reliability standard of Appendix A to 10 C.F.R. Part 50 as a useful guide against which the reliability of the makeup line may be assessed. We are, of course, referring to the "single failure" criterion set forth in the section of Appendix A entitled "Definitions and Explanations."

A single failure is an occurrence which results in the loss of the capability of a component to perform its intended safety function. (10 C.F.R. Part 50, Appendix A.) Fluid systems, such as the one designed to make up water to the Big Rock Point spent fuel pool, are considered to be designed against single failure if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function properly) results in the inability of the system to perform its safety functions. (*Id.*, Appendix A.) An active component is one that depends on moving parts for its proper operation, e.g., a pump, whereas a passive component is one that functions without moving parts, e.g., a pipe.

Specific criteria for design against passive failures in fluid systems are under development, and Appendix A does not provide specific insight on the treatment of such failures for design purposes. Nonetheless, and as provided in Appendix A, sound engineering practice dictates that mechanisms for passive failures be considered in the design of passive components in fluid systems. (Appendix A, n.2.) Consequently the

[2] *See* Criterion 61(5).

reliability of the makeup line will be evaluated against the "single failure criterion" and sound engineering practice to determine whether, as required by 10 C.F.R. § 50.57(a)(3)(i), operation with the expanded spent fuel pool capacity as proposed by the Licensee can be conducted without endangering the health and safety of the public.

C. Discussion

At the hearing, Licensee and Staff presented the testimony of several witnesses. Mr. David P. Blanchard, an engineer who helped to design the makeup water system, testified for Licensee regarding its reliability and its capacity to deliver a sufficient minimum flow rate to the spent fuel pool under a variety of accident conditions. (Finding A-2.)

Staff presented four witnesses. Fred Clemenson, a principal system analyst, and Richard L. Emch, Jr., the Big Rock Point project manager, testified to the makeup system's purpose, its capacity to deliver a sufficient rate of flow to the pool, and its reliability. Mark A. Caruso, a senior systems engineer, explained the basis for the Staff's conclusion that spent fuel pool water temperature can be maintained at or below 150°F by using the makeup system should normal pool cooling be lost. Finally, Dr. Pei-Ying Chen, a senior mechanical engineer, assessed the adequacy of the seismic design of the makeup pipe. (Finding A-5.)

Although the Intervenors withdrew their prefiled testimony (Tr. 4042) and did not present their own witnesses, they participated extensively in cross-examination of the Licensee and Staff witnesses. The Board also had many questions for the witnesses.

For purposes of evaluating the aspects of a postulated loss-of-coolant accident ("LOCA") pertinent to the contention, it is assumed that the spent fuel pool cooling system, which is located inside containment, becomes inoperable. This assumption is made because the spent fuel pool cooling system has not been qualified for a LOCA environment. If the cooling system fails, heat from the radioactive decay of the spent fuel will cause the water in the spent fuel pool to heat up and eventually boil and evaporate. Additional makeup water could be provided manually by entering containment. However, radiation resulting from the accident postulated in the contention may persist at unsafe levels for an extended period thereby precluding entry to containment. Consequently, the makeup water system is designed to provide makeup water by remote activation without personnel having to enter containment. (Findings A-6, A-7.)

Under the postulated accident scenario, the makeup water system will not begin operating immediately. Rather, between 4 and 24 hours after the onset of the accident, water which has collected at the bottom of containment will be drawn from containment at a minimum of 28 gallons per minute (gpm) by the core spray pump and routed through the makeup system to the spent fuel pool. (Finding A-9.) We do not anticipate any significant loss of water from the pool during this period prior to initiation of the makeup water system. Once this system is activated, the core spray pump recycles water from the containment floor through the core spray heat exchanger, where it is cooled to 100°F by heat exchange with water from the fire protection system; this cooled water is then directed to the spent fuel pool. (Finding A-8.)

When the reliability of the makeup system was first litigated at evidentiary hearings held during June 1983, the Licensee's evidence assumed that boiling of the spent fuel pool could occur if the normal cooling system were disabled under LOCA conditions. The possibility of pool boiling and water loss due to evaporation suggested several related safety concerns.

It is necessary to cool the spent fuel stored in the pool by keeping it covered with water. (Finding A-10.) If the possibility of the spent fuel becoming uncovered were credible, the potential for fuel melting and a zircaloy/steam reaction would require analysis. (Finding A-11.) A further unanalyzed scenario concerns Licensee's criticality analysis for the proposed expansion of the spent fuel pool. This analysis assumed a full pool of water in determining pool moderator conditions. A significant loss of water due to boiling might cause the neutron multiplication factor calculated by the Licensee to exceed the NRC Staff's guideline of 0.95. (Finding A-12.) Finally, pool boiling might cause containment repressurization due to the release of steam during the boiling process. (Finding A-13.)

All of the foregoing scenarios are, of course, avoided if pool boiling can be prevented. Licensee's redesign of the makeup system to maintain the pool temperature at or below 150°F would accomplish this objective. (Finding A-14.) Licensee committed to maintaining the 150°F temperature restriction because that is the highest temperature for which it can demonstrate the integrity of the concrete pool. (Finding A-15.) This is the temperature below which the American Concrete Institute Code indicates that loss of concrete strength is not significant. (Finding A-16.)

The 150°F temperature limitation is clearly the most restrictive of all the uses identified for the remotely actuated makeup system. It follows that maintaining the pool at or below this temperature ensures that all uses identified for the makeup system will be successfully fulfilled. We turn now to a discussion of the reliability of the makeup water system.

1. Reliability of the Makeup Water System

At the outset we note that during the hearing it was necessary for the Board to resolve a dispute between Licensee and the Staff as to the definition of the remotely activated makeup water system for purposes of this contention. Licensee took the position that the contention encompassed those portions of the core spray recirculation system on which the fuel pool relies for its source of makeup cooling water. This includes all piping and active components between the suction strainers in the bottom of containment and the fuel pool as well as the piping and active components associated with providing cooling water to the shell side of the core spray heat exchanger. (Finding A-17.) The NRC Staff, on the other hand, considered only the makeup pipe and its interfaces with the ECCS, which supplies cooled water for the makeup line, because the makeup pipe was the only hardware added to the existing plant. It did not reexamine the ECCS for reliability. (Clemenson, Emch, ff. Tr. 3979, at 5-6; Blanchard, ff. Tr. 3770, at 25-28.) (Tr. 3358.) The Board ruled that to the extent that the functions of the ECCS are identical with respect to the makeup line and to cooling the core, ECCS functions are licensed and need not be litigated. (*See Wisconsin Electric Power Co.* (Point Beach Nuclear Plant, Units 1 and 2), ALAB-739, 18 NRC 335, 338-39 (1983).) Those functions that are not entirely identical, however, such as additional flow and the temperature of water passing through the ECCS to the spent fuel pool, were considered litigable. (Tr. 3373, 3469-70.) With respect to seismic design, the makeup pipe and its connections to the ECCS must be considered.[3] (Tr. 3469.)

We adopt the single-failure criterion of 10 C.F.R. Part 50 for purposes of evaluating the reliability of the spent fuel pool makeup system in terms of the failure of active components. There are only three pairs of active components in the system feeding the makeup line, each of which is fully redundant: two core spray pumps, two fire pumps and two core spray heat exchanger valves (MO-7066 and VPI-5). (Finding A-18.)

Either of the two core spray pumps is sufficient to provide cooling water to the spent fuel pool without interfering with an adequate supply of water to cool the reactor core. Similarly, either of the two fire pumps is sufficient to provide cooling water to the shell of the core spray exchanger through either of the two core spray heat exchanger valves. (Finding A-20.) Moreover, these active components are located outside

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[3] In their proposed findings (Bier Finding 1.A), Intervenors have questioned the overall seismic capability of Big Rock Point structures and equipment, including the ECCS. We do not consider overall seismic capability of the plant to be within the scope of the contention or of a license amendment proceeding. We note, however, that an ongoing Staff Systematic Evaluation Program (SEP) is addressing the overall seismic capability of the Big Rock Point Plant structures and equipment.

containment and will not be required to operate in an accident environment. (Finding A-19.)

In addition, the power supply for the ECCS is adequate. The power for each core spray pump is supplied by a separate AC power bus, and while the normal power source for these buses is off site, either can be transferred to the emergency bus if offsite power is lost. (Finding A-21.) The emergency bus is powered by either of two onsite emergency diesel generators. (Finding A-22.) One fire pump is AC powered by the emergency bus and can be supplied by either of the emergency diesel generators if offsite power is lost. The second fire pump is diesel driven. (Finding A-23.) One of the core spray heat exchanger valves, MO-7066, is an AC-powered, motor-operated valve remotely actuated from the control room. In an emergency it can be powered by either of the two diesel generators via the emergency bus. This valve can be manually operated by a hand valve. The second core spray heat exchanger valve, VPI-5, is hand operated. (Finding A-24.)

Intervenors argue that a control room operator would not know whether the fire protection system had been activated because the control room at Big Rock Point is not equipped to indicate battery output current for the electrical fire pump or the diesel fire pump. (Bier Finding I.D.) The Staff testified, however, that there is flow instrumentation on the core spray system. (Emch, Tr. 3990, 4161.) This instrumentation would be used to determine whether the fire protection system had been activated and was operating. Hence the "missing" indicators for battery output current are not required.

The remaining components in the makeup system are passive, i.e., they need not operate to place the system in service; rather they merely provide a path for the core spray pumps to draw water from the containment and route it to the pool and core spray systems. These passive components include the suction and discharge of the core spray pumps, the core spray heat exchanger, the makeup pipe and valve to the spent fuel pool, and the piping between the fire pumps and the core spray heat exchanger shell.

Mr. Blanchard testified that, as suggested by Appendix A, sound engineering practice requires that consideration be given in the design of these passive components to avoid their failure. (Finding A-25.) For example, the majority of the components in the system feeding the makeup line are located outside of the containment, where there are no lines containing high-energy primary coolant. Therefore, these components are not vulnerable to pipe whip or steam impingement or to the hostile environmental conditions inside containment following an accident similar to TMI-2. Further, the makeup line is routed such that it is unlikely that a failure of the primary coolant system leading to a LOCA could simultaneously cause a failure of the pool makeup system. The makeup line and the system feeding it are also located so that the drop of a heavy object cannot simultaneously damage primary coolant system lines and components required for makeup to the fuel pool. (Finding A-26.) Mr. Blanchard testified that in his opinion no credible mechanisms could cause the failure of any passive components following a LOCA.[4] (Finding A-27.)

The makeup line itself is 190 feet long and consists of 115 feet of 2-inch-diameter piping and 75 feet of 1-inch-diameter piping. Given the pipe diameters, witnesses for both Licensee and the Staff agree that there is no credible possibility of pipe blockage by crud, scale, rust, or other foreign objects. Nevertheless, as an additional precaution, the pipe will be flushed each year with rust-inhibiting chromated water. (Finding A-28.)

The Board also has considered the possible adverse consequences of one surveillance test which is performed while the plant is at power and temporarily removes the core spray heat exchanger from service. During the time the heat exchanger is isolated for this test, the pumps of the core spray recirculation system are unable to pump water through the heat exchanger. This heat exchanger surveillance test, however, isolates the heat exchanger for no more than 4 hours once a month and thus is extremely unlikely to coincide with a LOCA. In any event, the valve which isolates the heat exchanger is located outside containment, and the operator is instructed to return the heat exchanger to service whenever a reactor trip occurs, regardless of its cause. (Finding A-29.)

We also note that administrative controls require that hand-operated valves routinely remain in positions necessary for the makeup water system to function. These valves are positioned otherwise only for testing and maintenance during reactor shutdowns, when a LOCA could not occur. Before the plant resumes operation the valves are returned to their correct positions in accordance with extensive surveillance procedures. The valves are locked into position after at least two individuals have confirmed that the valve is positioned correctly. Another check then verifies that the valves have been locked in the correct position. (Finding A-30.)

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[4] In their proposed findings (Bier Finding 1.B), the Intervenors question the condition of the yard piping, which is part of the fire protection system and provides water to the ECCS. Testimony provided by the Licensee (Blanchard, Tr. 2167-70) indicates that the functions of the fire protection system without the makeup line are identical to the function with the makeup line. In light of our ruling with respect to identical functions (Tr. 3469-70), Intervenors' proposed finding is outside the scope of this hearing. This matter lies within the Staff's responsibility to assure the safety of the plant.

Despite the redundancy built into the core spray pumps and their power supplies and despite the elaborate measures for ensuring proper positioning of valves, Licensee has taken further steps to increase the reliability of the makeup water system. Specifically, water can be routed directly from the fire pumps to the spent fuel pool through valve MO-7072. This valve can be manually operated from outside containment or remotely activated by a switch in the control room. (Finding A-31.)

The Intervenors argue, without supporting evidence, that long-term use of the fire protection system as a method of injecting water to the spent fuel pool could cause rupture of the containment (Bier Finding 1.C). It is extremely unlikely, however, that the fire protection system would be used to add water continuously to containment until the water level threatened the integrity of the containment. The water level in the containment would have to exceed 23 feet before there was danger of containment failure. For the 23-foot level to be exceeded, hundreds of thousands of gallons of water would have to be added. (Tr. 3783-87.) Because the fire protection system can deliver water at the rate of only 40 gpm, it would take a very long time for the system to add enough water to endanger the containment. (Tr. 3785-87.) If there were a threat to the containment from too much water, the water supply could be shut off manually from outside containment.

There are multiple layers of redundancy and backup capability built into the makeup water system. We find these features adequate to assure that a single failure of any active component of the makeup system will not result in the inability of the system to perform its intended safety function. Moreover, there are no credible mechanisms which could cause the failure of any passive components following a LOCA. Therefore, measured against the single-failure criterion of Appendix A to 10 C.F.R. Part 50 and sound engineering practice, we conclude that the remotely operated spent fuel pool makeup water system is reliable.[5]

2. *Flow Capacity of the Makeup Line*

The remaining outstanding question for ensuring the reliability of the makeup water system is whether it is capable of delivering sufficient flow to prevent the fuel pool temperature from rising above 150°F. The pool temperature is primarily a function of the decay heat generation

[5] The Intervenors proposed a finding with respect to the makeup line reliability based on an exemption to the ECCS single-failure criterion and the operability of the ECCS as a whole. But as we have noted, *supra*, issues concerning the ECCS alone are not litigable under this contention. Therefore, this finding must be rejected.

rate of the stored spent fuel. Mr. Blanchard, the only witness to testify on this subject, calculated a best-estimate decay heat rate of 176,000 watts, assuming that the expanded pool is filled to capacity, with 25 of the 441 assemblies having been discharged from the most recent refueling. (Blanchard, ff. Tr. 3770, at 8.) At this heat generation rate, a 24-gpm makeup rate is sufficient to prevent the bulk pool water temperature, and hence the concrete temperature, from exceeding 150°F. This result assumes that the temperature of the makeup water entering the pool is 100°F, which is the design outlet temperature of the core spray heat exchanger, and that activation of the makeup water system occurs no later than 2½ days following the loss of normal pool cooling. (Finding A-32.)

The Licensee recognizes that the best-estimate heat generation rate could be exceeded if more than twenty-five spent fuel assemblies were discharged into the pool from the most recent refueling, or if a refueling outage is less than 30 days. Also, seasonal variations in the temperature of lake water (the original source of the makeup line system water) could adversely affect the assumed heat removal capability of the makeup water entering the pool. (Blanchard, ff. Tr. 3770, at 28-29.) (Finding A-32.) As a result of these and other variables, Licensee has committed to institute a technical specification to assure that the heat removal capacity of the water entering the pool from the makeup line system is sufficient to prevent the pool temperature from exceeding 150°F before plant startup following any outage where spent fuel has been discharged into the pool. (Finding A-33.)

Such a commitment is formally submitted in § 7.3.2 of Licensee's Proposed Technical Specification Change, dated October 25, 1983, copies of which were distributed to the Board and parties at the hearing. (Tr. 2751-52.) In addition, flow testing of the makeup line will be performed before the startup after each refueling to assure that the line is free of obstructions. (Finding A-34.)

Licensee has used standard hydraulic analysis techniques to determine the flow rate through the makeup line under a variety of conditions. A mass and energy balance was performed to evaluate flow to the pool through the piping associated with the makeup system. Flow resistance was analyzed using a computer program called FLOWNET. FLOWNET was used by Licensee to establish the adequacy of core spray and enclosure spray flows following a postulated LOCA. FLOWNET was also used to design the makeup line to obtain adequate flow to the pool together with adequate core spray flow to the core. (Findings A-35, A-36.)

Two models that include the makeup line were analyzed with FLOWNET: the core spray system in a recirculation mode following a LOCA and the fire protection system with valve MO-7072 in position to supply water directly to the pool. Several cases were run assuming various configurations of the core spray and fire protection system, including single failures of components in the core spray system. In addition, a series of flow tests on the system were performed. In no cases were the calculated flows to the core spray lines or the spent fuel pool makeup line below the required minimum rates, even where the worst single active failure was assumed. (Findings A-37, A-38.)

Various issues that may potentially affect the plant ECCS, and thereby the makeup line, are under review in the Staff's SEP. These issues include overall plant seismic capability, as already noted, and susceptibility to tornado missiles. These issues are being evaluated to determine the need for plant improvements. At present the Staff has not identified any deviations from current requirements which require immediate action before the SEP is complete. (Finding A-39.)

We find that the makeup system is capable of delivering adequate flow to the spent fuel pool, even under single failure accident conditions. Moreover, a technical specification will prohibit startup of the plant following an outage in which fuel is discharged to the pool until procedures have been followed to assure that the makeup system can perform its intended function; flow testing will be performed before startup, as well, to make certain the system will deliver water to the pool as required.

3. Structural Adequacy of the Makeup Line

The structural adequacy of the makeup line was addressed by Arthur K. Smith, a senior engineer for Licensee, and Pei-Ying Chen, a senior engineer for the NRC Staff. (Findings A-4, A-5.) The makeup line (or pipe) is made of Schedule-80 carbon steel. A dropped wrench would not dent the pipe significantly, nor could the pipe be crushed by being stepped on. We note also that Licensee has administrative controls to prevent fuel elements from falling on or near the makeup line. (Finding A-40.)

The structural adequacy of the makeup line under seismic loading conditions was determined by computing potential pipe stresses using the ADLPIPE computer code and comparing these stresses to those allowable under applicable piping and support codes. The maximum pipe stress from seismic loading that would be expected during safe shutdown earthquake loading conditions is approximately 8800 psi, while the allowable stress is 36,000 psi. (Finding A-41.) In addition, the fact that the makeup line crosses expansion joints has no significant effect on its seismic capability; the motion of the expansion joints is very small and will not significantly affect pipe stress. Finally, all pipe supports were evaluated in accordance with the American Institute of Construction (AISC) Manual Code. (Finding A-42.) We find that the makeup line is adequate to withstand the maximum stress induced by seismic loading.

4. *Maximum Localized Temperatures in the Spent Fuel Pool During Operation of the Makeup System*

JAYCOR, an engineering and scientific research and development firm employed by the Licensee, performed a thermal hydraulic analysis of the Big Rock Point Plant spent fuel pool to determine whether, despite an average water temperature of 150°F, the pool walls and floors might experience higher temperatures in localized areas. (Finding A-43.) To determine the greatest temperature which could develop in the spent fuel pool, it was necessary for JAYCOR to calculate the circulation patterns which carry heat away from the fuel elements. (Finding A-44.) For this purpose, JAYCOR's EITACC-SFP computer program solves a set of equations that simulate buoyant flow in spent fuel pool geometries, taking into account the location of the inlet cooling water, the location of the exiting flow, and the geometric blockage and flow resistance of the spent fuel racks. The simulation generates detailed estimates of the temperature and flow quantities in cells representing every part of the pool. (Findings A-45 and A-46.)

The JAYCOR model assumed that the makeup system pours 100°F water onto the top of the northeast corner of the pool at a flow rate of 30 gpm and that the fuel generates 217,000 watts, with spent fuel assemblies that generate 62% of that heat rate being located in the northwest corner of the pool. In addition, no credit was taken for heat loss through the walls, floor, or pool surface. (Findings A-47, A-48.) Using these highly conservative assumptions, JAYCOR concluded that the highest temperature on the pool floor varied from the average temperature of 150°F by no more than 0.4°F, while the highest wall temperature was only 2.7°F greater than the average. (Finding A-49.)

The design basis for the makeup water system was initially conceived by Licensee to be 30 gpm as the maximum amount of flow that could be diverted from the core spray system under worst-case conditions. As a result of completing the actual design and testing of the system, 28 gpm was established as the maximum flow rate. (Finding A-50.) JAYCOR reexamined its analysis in view of the 2-gpm reduction in flow rate. Since, as recognized earlier in this Decision, the 150°F temperature limit can be maintained by simply altering the heat generation rate, JAYCOR used a 28-gpm flow rate with a heat rate of 205,000 watts (an estimate that is still conservative in light of Mr. Blanchard's best estimate of 176,000 watts). JAYCOR determined that under these conditions the general circulation patterns predicted at the higher flow and heat rates remained unchanged. The only difference was a drop of 0.1°F in the temperature at the warmest spot in the pool (2.7°F to 2.6°F). (Finding A-51.)

The EITACC-SFP computer code used in the thermal hydraulic analysis was verified under JAYCOR's quality assurance program. (Finding A-52.) Moreover, three experiments collectively provide data for validating the use of EITACC-SFP for assessing temperature conditions in the Big Rock spent fuel pool.

First, data recorded in cold-leg injection experiments performed by the Electric Power Research Institute have provided detailed information on the mixing of cold and warm streams in a variety of turbulent flow situations. The JAYCOR computer code was applied to this problem and its predictions were compared with the measured temperature fields. Despite the complexity of the phenomena, the JAYCOR calculations were generally within a degree of the measured values and always within the scatter of the experimental data. (Finding A-53.)

Second, an attempt was made by another company to measure the temperature and flow patterns in the Maine Yankee spent fuel pool during a refueling outage in 1982. In most locations within the pool, however, the measuring devices were not sufficiently sensitive to measure the convective flows that developed. The EITACC-SFP computer code correctly predicted that flow velocities in the Maine Yankee pool would often be beneath the range of the measuring devices and that pool temperatures would be within the range of experimental error values obtained. (Finding A-54; Stuhmiller and Sargis, ff. Tr. 3849, at 18-19.)

Finally, JAYCOR performed a scale model experiment to develop data on convective flow patterns in operating spent fuel pools. A color movie of the experiment was shown at the hearing. The scale model experiment was used to verify the EITACC-SFP computer code. Computer calculations were performed to correspond to the model tests, and these calculations were then compared with the actual average and local temperatures and with the observed circulation patterns. The EITACC-SFP simulation produced temperature estimates that generally were within half of a temperature degree of scale model results, and the maximum error was between 1 and 2 temperature degrees. (Finding A-55.)

During the hearings, Intervenors established that additional items are placed in the spent fuel pool besides spent fuel and spent fuel racks. Some of these items are stored in the pool on a permanent basis, including special racks containing control blades and a small amount of research and development equipment. Other, smaller, items are stored in the pool temporarily and then sent off site. Radioactive maintenance materials are stored temporarily in buckets on the floor of the pool for biological shielding purposes. Large casks are periodically stored in a designated area in the southwest corner of the pool. (Finding A-56.)

Intervenors argued that calculations had not taken into consideration objects other than racks and fuel rods that are stored in the fuel pool and that there are no administrative controls for items being placed in the pool. (Christa-Maria Finding on Reliability of Makeup Water System.) Therefore, they urge us to find that the presence of these additional objects in the pool invalidates the JAYCOR computer analysis.

Such a finding would be contrary to the evidence. Mr. Blanchard testified that all equipment of significant size that was permanently stored in the spent fuel pool had been taken into account in the JAYCOR computer model. Indeed, as a conservatism JAYCOR included in its calculations more racks than are presently in the pool. (Finding A-57.) Dr. Stuhmiller testified for JAYCOR that the placement of additional objects on the floor of the spent fuel pool could, under certain circumstances, block or divert flow patterns and influence local temperatures. He further testified, however, that local temperatures will not be affected as long as important flow patterns are not blocked. For the Big Rock Point spent fuel pool, the important flow pattern is through the space between rack B and the east wall of the pool. This space, if not blocked, will provide the necessary cooling path, and local temperatures will remain consistent with the JAYCOR analysis. (Finding A-58.) To that end, Licensee will promulgate written administrative procedures prohibiting the storage of any materials in the area between rack B and the east wall of the pool. (Finding A-59.)

Mr. Caruso of the NRC Staff reviewed the JAYCOR analysis. From that review and from the Licensee's proposed technical specification verifying cooling capacity, Mr. Caruso concluded that the pool water will be well mixed by natural circulation and that a bulk pool water temperature of 150°F will be maintained with a maximum localized water temperature of less than 153°F. (Finding A-59a.)

The JAYCOR thermal hydraulic analysis is sophisticated and thorough. The analysis demonstrates that the temperature distribution within the Big Rock Point spent fuel pool during operation of the makeup water system will generally be at or below 150°F, with the limited exception that it may reach 153°F in one portion of the pool.

5. *Zircaloy/Steam Reaction*

The second major issue for consideration concerns the possibility that zircaloy will react with steam in the spent fuel pool. Based on our previous findings this issue is easily resolved. We note first that the fuel cladding at Big Rock Point is made of zircaloy, which can react with steam at high temperatures. However, the reaction rate becomes significant only at or above temperatures of approximately 2200°F, and the fuel cladding could approach this temperature only if water in the pool evaporated and the spent fuel became uncovered. (Findings A-60, A-61.) Since we have already concluded that the makeup water system will maintain a full pool water level at an average water temperature not to exceed 150°F, and localized temperatures no more than 3°F greater, we conclude that the makeup system will prevent zircaloy/steam reaction from occurring in the pool.

D. Conclusion

The record establishes that should an accident prevent entry into containment and cause the normal pool cooling loops to fail, Licensee's remotely activated makeup water system is adequate to keep the spent fuel pool full of water, the average temperature of which will not exceed 150°F. The reliability of the makeup system has been established based on the single-failure criterion of Appendix A to 10 C.F.R. Part 50 and sound engineering practice. The makeup line itself is structurally sound and fully sufficient to withstand the maximum stress induced by seismic loadings. Finally, there is no realistic possibility that zircaloy cladding of the spent fuel will be exposed to steam in the spent fuel pool.

Consequently, we conclude that, consistent with 10 C.F.R. Part 50, Appendix A, and specifically GDC 61, Licensee's fuel storage and handling systems have been designed to provide adequate safety under normal and postulated accident conditions; in particular, they have been designed to prevent any significant reduction in fuel storage coolant inventory under accident conditions. The genuine issues of fact regarding the adequacy and the reliability of the makeup systems, and the potential for zircaloy/steam reaction in the spent fuel pool, are therefore dismissed.

Necessarily, then, and in accordance with the Appeal Board's decision in ALAB-725, *supra* (*see* 17 NRC at 572), we also find that there is no credible potential for the occurrence of a criticality accident due to a LOCA; therefore, the genuine issue of fact concerning criticality remanded to us by the Appeal Board and arising under Christa-Maria Contention 8 and O'Neill Contention III.E-2 is also dismissed.

II. CHRISTA-MARIA CONTENTION 8 AND O'NEILL CONTENTION III.E-2 — INTEGRITY OF THE CONCRETE POOL STRUCTURE

A. Background

Christa-Maria Contention 8 and O'Neill Contention III.E-2 postulate the occurrence of an accident, similar to the TMI-2 accident, that would prevent entry into the containment building for an extended period. A genuine issue of fact admitted under this contention at summary disposition asks whether the postulated accident could cause the water in the spent fuel pool to boil and the pool's concrete to fail. Also admitted as an issue was whether point loading, resulting from the weight of the spent fuel and the storage racks being applied to the pool floor through the rack legs, could cause failure of the concrete. (Finding B-1.)

The Licensee filed testimony in May 1982 presenting an analysis of the pool concrete under the accident conditions. The NRC Staff also filed testimony, which expressed uncertainty about the assumed strength properties of concrete at the elevated temperatures that would prevail if the pool were to boil. Licensee withdrew its testimony at the hearings held in June 1982. Subsequently, Licensee committed to install a modified remotely activated makeup water line that will maintain 150°F as the maximum bulk pool temperature. (Finding B-2.) Since Licensee's makeup line must assure that maximum temperature, the safety of the concrete need only be assured at that temperature. It is, therefore, no longer necessary for Licensee to demonstrate the safety of its concrete at boiling temperature.

On September 30, 1983, Licensee submitted testimony presenting a structural analysis of the concrete pool under dead, hydrostatic and thermal loadings. The analysis assumes that the bulk pool temperature will not exceed 150°F and demonstrates that the pool structure is adequate to withstand such loads. In response to a Licensing Board question regarding assurance that current concrete code standards used in this analysis apply to a pool built 20 years ago, Licensee submitted testimony demonstrating that the analysis applies to the as-built structure. (Finding B-3.)

The NRC Staff submitted testimony which concluded that the Licensee's analysis adequately assured the integrity of the concrete structure under the assumed accident conditions. The Staff also agreed that the analysis was applicable to the structure as built. (Finding B-4.) This issue was fully litigated at the hearings held in Petoskey, Michigan, in October and November of 1983.

B. Applicable Law

Appendix A to 10 C.F.R. Part 50 sets forth a series of General Design Criteria for nuclear power plants. Although it was promulgated after the Big Rock Point Plant was constructed and licensed, Appendix A provides a useful guide against which the plant's structures and systems may be compared. General Design Criterion ("GDC") 61 provides in pertinent part:

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions.

In the context of this issue we take GDC 61 to require that the Licensee demonstrate the integrity of the reinforced concrete pool structure under the assumed accident conditions.

General Design Criteria 1, 2 and 4 provide generally that structures, systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that they be designed to withstand the effects of postulated accidents and resulting environmental conditions as well as the effects of normal operating conditions. Regulatory Guide 1.142, Rev. 1 (October 1981) and Standard Review Plan, § 3.8.4, Rev. 1 (July 1981) provide guidance with respect to these criteria. They indicate that the procedures and requirements described in the American Concrete Institute ("ACI") Code 349, "Code Requirements for Nuclear Safety Related Concrete Structures," provide an adequate basis for complying with the Commission's regulations with regard to the design and construction of safety-related concrete structures. The current version of this Code is ACI 349-80 (1980). In particular, Appendix D to SRP § 3.8.4, Rev. 0, "Technical Position on Spent Fuel Pool Racks" (July 1981), finds the analysis procedures of ACI 349 acceptable for spent fuel pool structures. The ACI Code specifies applicable concrete strength capacities and analytical procedures to be followed in assessing the impact of thermal loads on concrete structures. It also establishes criteria to assure the adequacy of the reinforcing bars. The associated ACI Building Code, ACI 318-77 (1977), provides for the application of a structural analysis to an existing reinforced concrete structure.

C. Discussion

The Big Rock Point spent fuel pool is a rectangular reinforced concrete structure, the cavity of which is lined with 3/16-inch stainless steel. The walls vary in thickness from 2 feet to 6 feet 9 inches. On three sides the pool structure is supported by walls which lie below the pool walls. On the fourth side it is supported from below by a shear key, a reinforced concrete member which protrudes from the reactor cavity concrete. (Finding B-5.)

Under the postulated accident conditions, thermal loads, in the form of temperature gradients, are imposed on the walls and floor of the pool as the pool water heats up relative to the surrounding air. Concrete expands as it is heated. The inner surfaces of the pool walls and floor will heat first and will tend to expand more than the cooler outer portions. Because the walls and floor are connected, they cannot independently bend to accommodate this growth, and internal forces are created in the concrete. These forces, termed shear forces and bending moments, resist the tendency of sections of the concrete to shear (i.e, slide relative to one another) and to bend. (Finding B-6.)

1. The NUS Structural Analysis

Dr. Howard J. Eckert and Dr. Madarapalli K. Prabakhara, structural engineers employed by NUS Corporation, presented the results of a finite element analysis that they performed to determine the integrity of the structure. (Finding B-3.) They initially determined the thermal loads caused by the assumed accident conditions. They also calculated the loads imposed by the weight of the structure itself and its contents. Using these loads and a mathematical model of the structure, they calculated parameters, such as moment and shear, which portray the structure's behavior. To determine the adequacy of the structure, they then compared these shear forces and bending moments to the strength capacities of the concrete and the adequacy of the steel reinforcing bars imbedded in it. (Finding B-7.)

2. *Loading Conditions*

The witnesses assumed a water heatup rate of approximately 1°F per hour from the operating temperature of 101°F to a maximum bulk temperature of 150°F. Because the stainless steel pool liner expands faster than the concrete, they also considered the load this differential thermal expansion would impose. In addition, they considered the hydrostatic pressure applied to the walls and floor by the pool water and the deadweight loading of the water, the racks, the fuel, the floor slab and miscellaneous equipment. In determining the strength capacities of the support walls they also considered the weight of the pool walls. (Finding B-8.)

3. *Structural Analysis*

Drs. Eckert and Prabakhara performed a finite element analysis, idealizing the structure as an assemblage of discrete blocks, for each of which shear forces and bending moments were calculated. Because the inner surfaces of the walls and floor will tend to expand more, the inner portions of the structure will be in compression while its outer portions are in tension. (Finding B-9.) Concrete is relatively weak in tension; hence the need for steel reinforcing bars. When the tensile stress becomes great enough, a crack is formed, and as load increases the crack progresses, thereby reducing the flexural stiffness of the section and relieving the stress, affecting the distribution of load as load application continues. To reflect this behavior the witnesses performed a nonlinear analysis, increasing the load in increments, after each of which the stiffness was reduced, until the maximum gradients were reached. (Finding B-10.)

This procedure is approximate because it assumes that the maximum gradients for each wall and the floor occur at the same time. In reality, because of the differing thickness of these elements, they would heat at different rates. Because the NRC Staff questioned whether this method of applying load was conservative, and because of an error in the application of the computer code, Drs. Eckert and Prabakhara repeated the analysis, this time applying the maximum gradients at the time they actually occurred and correcting the error in computer code application.

The NRC Staff also questioned the ability of the structure to resist forces generated by differential expansion of the steel liner and pool concrete; this factor had been omitted in the January 10, 1983 analysis. Therefore the witnesses also performed a study of the effect of the differential thermal expansion of the stainless steel liner on the pool concrete in conjunction with the reanalysis of the January 10, 1983 submittal.

4. Strength Capacities

The witnesses calculated strength capacities at various cross-sections of the structure in accordance with the ACI Code. The capacities are a function of the yield strength of the steel reinforcing bar, the compressive strength of the concrete and the dimensions of the section. (Finding B-12.) The Code indicates that the strength properties of concrete are not degraded at a temperature of 150°F, and it allows temperatures of up to 200°F in local areas. (Finding B-13.) The Code also specifies required development lengths for the reinforcing bar, i.e., the depth of embedment necessary to assure that the bar can be stressed to the yield point. Splicing of the bars is normally accomplished by overlapping, and required lap splice lengths are also specified by the ACI Code. The analysis showed that in one location a lap splice was not sufficient to meet the Code criterion. The witnesses testified, however, that they used information contained in a technical paper[6] to recalculate the required splice length taking into account the strength provided by the 6 inches of concrete covering the splice; this calculation showed that the splice was adequate. (Finding B-14.) Neither the Intervenors nor the NRC Staff challenged this recalculation, which we find acceptable.

5. Conclusions

The final step in the Licensee's analysis was to compare the strength capacity of the structure to the calculated forces. To quantify this comparison, the witnesses computed ratios of the shear and moment capacities to the calculated values of shear forces and bending moments. They also computed ratios of the length of the reinforcing bars and overlaps to those required to develop the moment capacities. Values of these ratios, or margins, greater than 1 indicate that there is a margin of safety, i.e., excess capacity, or strength. (Finding B-15.)

The thermal-hydraulic analysis of the Big Rock Point spent fuel pool performed by JAYCOR and discussed in § E of this opinion showed a localized area in one corner of the pool in which the pool water temperature reaches 152.7°F, i.e., 2.7°F greater than the bulk pool temperature. Such a localized temperature is acceptable with respect to concrete strength properties, since the ACI Code allows temperatures of up to 200°F locally. Also, the strength margins at this location are sufficient to accommodate the effects of this small localized increase in pool water temperature. (Finding B-17.)

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[6] Orangun, Jirsa, and Breen, *A Re-evaluation of Test Data on Development Length and Splices*, 17 ACI Journal Proc. 114-22 (March 1977) (ff. Tr. 4056, at 12, and References).

For the pool floor and walls, all *average* shear, local shear, and development length (with the inclusion of the recalculation discussed above) factors of safety are greater than unity, as is the moment margin for the pool floor. These calculations make the reasonable assumption that there are minimal gaps between the pool and the liner because there is not a perfect fit. In one location in one wall, the moment margin was less than unity. Exceeding the allowable moment locally, however, is acceptable provided the surrounding material can carry the additional load and no collapse mechanism develops. The witnesses examined the margins surrounding the region where moment capacity is exceeded and concluded that the surrounding material is capable of carrying the additional load and that there would not be a collapse. Thus, when load distribution in the walls is factored into the analysis, the support walls have margins greater than unity with respect to all applicable parameters. (Finding B-18.)

6. *Additional Analyses*

In response to an NRC Staff question, the Licensee analyzed the shear key located on the west wall and determined that it was adequate to support all calculated loads. (Finding B-19.) Subsequently, however, Drs. Eckert and Prabakhara reduced the margins calculated by Licensee after the NRC Staff pointed out that the weight of the wall over the shear key had not been included in the calculation. The reanalysis showed that some of the local shear margins were less than 1. The witnesses found this to be of no significance, however, because the north and south support walls act in parallel with the shear key to carry the loads. Even if the shear key were eliminated, the other two walls would be more than sufficient to carry the load. (Finding B-20.)

Drs. Eckert and Prabakhara also considered the weight of the 120,000-pound shipping cask, the heaviest object that can be set in the spent fuel pool The effect of this additional weight would be to further reduce the margin for the shear key. Again, however, the support walls would take the added load; indeed, the shear key is not needed at all to support the fuel pool structure. The witnesses also evaluated the load imposed by the cask on the corner of the pool floor where it would rest. They concluded that the margins were more than adequate to withstand this local pressure. (Finding B-21.)

With regard to point loading from the storage racks, Drs. Eckert and Prabakhara reviewed and adopted the analysis contained in the Licensee's Consolidated Application. The analysis considered bearing stress, resulting from the weight of the rack and fuel applied through the rack leg, and punching shear stress, the local loading condition under the rack leg which could punch a hole through the pool floor. The analysis determined that margins were greater than 1 in all instances. (Finding B-22.)

In addition, Mr. Gary Pratt of Consumers Power Company performed an analysis showing that when the containment atmosphere temperature rises rapidly during a LOCA, so that the outside of the pool structure is heated more than the inside, the loads imposed on the structure are less severe than those analyzed in detail by Drs. Eckert and Prabakhara. (Finding B-23.)

On the basis of all these analyses, Drs. Eckert and Prabakhara concluded that the spent fuel pool structure is adequate to resist the effects of a temperature of 150°F and point loading from the storage racks. (Finding B-25.)

At the hearing the Board raised a question about the scenario analyzed by Mr. Pratt. The Board asked whether, when the containment sprays were activated following the LOCA, cold water from the sprays would impinge upon the pool walls, possibly reducing the temperature of the outer portions below the ambient temperature used in the NUS analysis, thus creating gradients larger than those considered in that analysis. (Tr. 4207 ff.) In response, Mr. Pratt testified that three of the pool walls are shielded from the sprays. Moreover, the containment sprays are located high above the pool and the nozzles put out a very fine spray. By the time the spray reaches the pool walls, it will have absorbed all the heat it is capable of and will be at the ambient temperature. The temperature profiles used in the NUS analysis therefore remain valid. (Finding B-24.)

7. *Staff Review*

Mr. Drew Persinko, who reviewed the Licensee's analysis for the NRC Staff and helped prepare the Staff's Supplemental Safety Evaluation Report (SSER) on the pool concrete, testified on this issue for the Staff. (Finding B-4.) Mr. Persinko thoroughly reviewed the structural analysis, and requested Drs. Eckert and Prabakhara to perform several reanalyses to assure him of the accuracy of certain details in the modeling. Based on his review, Mr. Persinko concluded that the spent fuel pool structure is adequate to withstand the increased load resulting from the proposed pool expansion for pool water temperatures up to 150°F. (Finding B-26.) The Staff's conclusions were based upon the reanalyses of the structure, which included the effects of differential expansion of the liner and concrete. (Persinko, ff. Tr. 4169, at 3; SSER, ff. Tr. 3988, at 5.)

Mr. Mark A. Caruso also reviewed the thermal portions of the analysis for the Staff and helped prepare the Staff's SSER. (Finding B-4.) He testified that the thermal analysis methods used by NUS to calculate temperature distributions in the concrete were appropriate. He also testified that based on the uniformity of pool water temperature shown in the JAYCOR thermal-hydraulic analysis, the calculated temperature distribution appeared reasonable. (Finding B-27.)

8. *Applicability of the Analysis to the Structure as Built*

The Licensing Board raised questions about the applicability of the NUS structural analysis to the spent fuel pool structure as built. The Board asked whether the American Concrete Institute (ACI) Code standards in effect when the pool was built would make it appropriate to apply to the existing structure concrete strength capacities derived from the current Code. The Board also questioned the basis for assurance that the structure as built complied substantially with the Code in effect at the time. (Tr. 4060-67, 4076-78, 4115-16, 4131-33, 4187-96.) In response Licensee submitted the testimony of Professor Mete A. Sozen, a nationally recognized authority on reinforced concrete structures and a member of the ACI Building Code committee, and the testimony of Jerome D. Lescoe, Licensee's construction superintendent during construction of the Big Rock Point Plant. (Finding B-3.)

(a) Code Criteria

Professor Sozen testified that the acceptance criteria used by NUS could validly be applied to Big Rock Point. The criteria assumed by NUS related mainly to flexural, shear and bond strengths and were derived by the procedures specified in the current ACI Concrete Code and the associated ACI Building Code. The Big Rock Point Plant was built according to the 1958 Uniform Building Code, whose provisions pertaining to reinforced concrete were based on the 1951 ACI Building Code. (Finding B-28.) Although the current ACI Building Code contains some conservatisms not present in the 1951 Code, most of the fundamental criteria contained in the current Code (ACI 318-77 (1977)) are essentially the same as those in the 1951 version. The design requirements on which strength capacities are based have not changed substantially since 1951. Although the 1951 version was based on working stress and design, while current methods are based on ultimate strength design, essentially the same sizes and sections would result for a given load. (Finding B-29.)

Under the criterion used by NUS to determine flexural strength, or the structure's capacity to tolerate bending moments, the strength of a lightly reinforced structure is insensitive to variations in the compressive strength of the concrete; rather, it depends on the strength of the reinforcing steel. The current allowances for steel strength have remained unchanged for over 20 years. (Finding B-30.) Shear strength is calculated under the current Building Code by a method different from that of the Code under which Big Rock Point was built. The current shear strength limit used by NUS, however, is conservative, assuring that the plant also meets the criteria that would have been used at the time of design. (Finding B-31.) The current Code's criterion for bond strength, which controls the required development length of the rebar, is not directly comparable to that of the earlier Code because of calculational and design changes. Particular comparisons, however, show that the current requirement used by NUS is more conservative (Finding B-32.) Professor Sozen also examined the analysis of the shear key and concluded that it was extremely conservative in that actual strength of the shear key would be 3 or 4 times what was assumed. He also concluded that there will be adequate support of the pool along the west edge. (Finding B-33.)

(b) Relationship of Existing Structure to Code Criteria

The basic parameters of the NUS structural analysis are the pool dimensions, the concrete strength and the amount, arrangement and strength of the steel reinforcement. The current ACI Building Code provides for the application of such an analysis to an existing structure. In such a case the Code requires a thorough field investigation of dimensions, properties of materials and other pertinent conditions. Such an investigation has been undertaken at Big Rock Point. (Finding B-34.)

Dr. Eckert took actual measurements of the spent fuel pool walls and found them to conform to the values indicated in the structural drawings. (Finding B-35.) The NUS assumption of a concrete compressive strength of 3000 psi has been verified by documentation as well as field investigation. Recorded compression tests of cylinders made from the Big Rock Point spent fuel pool concrete indicated a mean compressive strength of 3686 psi, and no cylinder was below 3000 psi. The cylinder tests also indicated excellent quality control. (Finding B-36.) In addition, recorded slump readings and the fact that there was no congestion of reinforcement, i.e., closely spaced reinforcement bars that would inhibit concrete flow during casting, indicate no likelihood of critical voids within the concrete. (Finding B-37.) Moreover, Professor Sozen's field investigation verified that the appearance of the concrete does not suggest defects in the casting process. Had large voids existed around groups of reinforcing bars, unusual surface cracks would most probably have appeared after over 20 years of use. (Finding B-38.)

The Code procedures for evaluation of existing structures contain no requirement for precise information on amount and arrangement of the reinforcement, which is assumed to conform to the structural drawings. (Finding B-39.) For the Big Rock Point pool, however, an existing construction photograph, introduced into evidence, shows the reinforcing bars for the floor slab before the concrete was poured. The detail is sufficient to show that placement and spacing of the bars conforms to the structural drawings and suggests that the job was well controlled. (Finding B-40.) In addition, there is no visible indication that would suggest that a serious omission of reinforcement occurred. (Finding B-38.)

Professor Sozen concluded that the limiting strength criteria used in the NUS analysis are correct for and applicable to the spent fuel pool structure for several reasons. They are based on accepted engineering principles consistent with current professional practice. Furthermore, they are comparable to if not more conservative than those used at the time of construction. Finally, the available information about the pool as built is adequate to substantiate these strength criteria. (Finding B-41.)

This conclusion was reinforced by the testimony of Jerome D. Lescoe, who, as Licensee's construction superintendent for the Big Rock Point Plant, was responsible for Licensee's overview of the performance of Bechtel Corporation, the engineer-constructor of the facility. Mr. Lescoe was knowledgeable in good construction practices for pouring reinforced concrete structures and he observed concrete pours on a daily basis at Big Rock, including pours for the spent fuel pool. (Finding B-42.) He observed that Bechtel followed their drawings and specifications and used appropriate methods to form and place concrete. Before a pour was made, the general foreman and an engineer saw that rebar placement complied with drawings and that the area was free from rust or debris. During the pour, they used techniques to keep the concrete from separating and complied with good practice in the use of vibrators to eliminate voids. (Finding B-42.) The photograph admitted in evidence showing construction of the pool was taken under Mr. Lescoe's supervision and he confirmed that placement of the rebar in the photo conforms to the structural drawings. (Finding B-44.) He also observed that the concrete cylinders used for the compression tests were kept in the immediate area of the pour so they would cure under the same conditions. (Finding B-45.)

Mr. Persinko read the testimony of Professor Sozen and Mr. Lescoe and examined the concrete test records and a construction photograph. Nothing presented by Mr. Lescoe or Professor Sozen in their oral testimony or in their written testimony caused him to change his conclusions in the SSER or his testimony. After his review of the concrete test records and construction photograph, he reached the same conclusion as Professor Sozen, namely, that the assumptions in the licensee's analysis appear to be applicable to the as-built structure. Mr. Persinko also physically inspected the pool structure and did not detect any visible defects. (Finding B-46.)

D. Conclusion

The structural analysis of the fuel pool presented by Licensee is extremely detailed and persuasive. The Staff's review of the analysis was rigorous. Intervenors presented no testimony on this issue and cross-examination of the Licensee and Staff witnesses did not cast doubt on the validity of their conclusions. The Board finds that the Licensee's analysis assures the adequacy of the pool structure under the assumed accident conditions. Based on the evidence, the Board also finds that the Licensee's analysis validly applies to the pool structure as it was built. The current Code criteria used in the analysis are comparable to or more conservative than those used at the time of construction, and there is sufficient information about the construction of the pool to conclude that these criteria are applicable to the structure as built. (Finding B-48.)

III. O'NEILL CONTENTION II.E-4 — SHIELDING

A. Background

O'Neill Contention II.E-4 states:

> In the event of an accident which results in a substantial release of radioactivity from the expanded fuel pool, the containment building does not provide adequate shielding to protect the public health and safety.

Testimony on this contention was heard from two witnesses: Mr. Roger Sinderman, Director of Licensee's Radiological Services Department and Mr. Millard Wohl, a nuclear engineer with the NRC's Accident Evaluation Branch. Intervenors withdrew their prepared testimony and presented their case solely through cross-examination of the witnesses.

B. Applicable Law

The contention was restated by the Licensing Board in Order Following Special Prehearing Conference, dated January 18, 1980. (LBP-80-4, *supra*, 11 NRC at 130.) The term "shielding" was deliberately used by the Board to reflect Mr. O'Neill's concern about the shielding capability of the Big Rock Point containment building. The containment consists, in part, of a ¾-inch steel shell which can be penetrated by gamma radiation emanating from inside containment. (*See* Prehearing Conference, December 5, 1979, Tr. 179-80; *see also* Tr. 4281-82.) Consistent with this intended meaning and the use of the "shielding" as a term of art, we limited this contention to a consideration of the adequacy of the containment building to protect the public from gamma radiation shining through the containment as a result of a substantial release of radioactive material from the spent fuel pool. (Tr. 4282, 4313.)

The contention did not specify the accident which results in a substantial release of radiation from the spent fuel pool, nor does it provide insight for identifying such an accident. Thus, the witnesses were required to identify the accident that would result in a substantial release of radioactivity from the spent fuel pool.

Generally, a cask drop accident is considered a design basis accident involving a substantial release of radiation from a spent fuel pool. However, our decision on O'Neill Contention II.C regarding the drop of a spent fuel transfer cask and the acceptability of its safety slings renders such an accident incredible at Big Rock Point. (*See* our decision regarding the cask drop aspect of O'Neill Contention II.C, in Finding C-5.)

Thus, the design basis accident that could result in the largest release of radioactivity from the spent fuel pool was determined to be the drop of a spent fuel assembly onto fully loaded spent fuel racks. (Finding C-4.) In turn, it was assumed that all of the gap activity of the dropped spent fuel assembly would be released into the spent fuel pool and the containment building. (Finding C-7.)

The controlling NRC Staff guidance for the evaluation of the consequences of an accident such as the drop of a spent fuel assembly is found in § 15.7.4, "Radiological Consequences of Fuel Handling Accidents," of the Standard Review Plan ("SRP"), NUREG-0800. The acceptance criteria for SRP § 15.7.4 are based on General Design Criterion ("GDC") 61 with respect to appropriate containment systems, and on 10 C.F.R. Part 100, with respect to calculated radiological consequences of a fuel handling accident.

GDC 61 provides:

> *Criterion 61 — Fuel storage and handling and radioactivity control.* The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall

be designed to assure adequate safety under normal and postulated accident conditions. . . .

SRP § 15.7.4 provides that plant site and dose-mitigating engineered safety features are acceptable with respect to the radiological consequence of a postulated fuel handling accident if the calculated doses at the exclusion area boundary are well within the exposure guideline values of 10 C.F.R. § 100.11. The limits established by that section require that an individual located at the site boundary for 2 hours following the onset of the postulated fission product release will not receive a total radiation dose to the whole body in excess of 25 rem. (We disregard guidance concerning thyroid exposure inasmuch as the contention assumes no release of radioiodine to the environment; hence the thyroid dose would be equal to the whole-body dose.)

C. Discussion

1. Potential Public Exposure

Under the parameters of this postulated accident of the drop of a spent fuel assembly in the spent fuel pool, the NRC Staff witness, Mr. Wohl, calculated the exposure to an individual at the site boundary over a 2-hour period to be 0.2 millirem. (Finding C-9.) Licensee's witness, Mr. Sinderman, calculated this dose to be 0.0035 millirem. (Finding C-8.) Both witnesses based their calculations on the attenuation provided by the containment building and the atmosphere. (Findings C-8, C-9.)

The disparity between their conclusions was explained by the witnesses. Mr. Wohl used a conservatively selected gamma-ray air buildup factor to compute the offsite dose resulting from radionuclides within containment. Mr. Sinderman calculated the gamma radiation dose using the attenuation coefficients and buildup factors for each gamma ray of each separate nuclide, by means of a licensee-proprietary computer code. (Tr. 4437-38.) The witnesses believed either method was acceptable for calculating the public exposure. We agree that both methods are acceptable and that they demonstrate conclusively that the radiation levels at the plant boundary would be well within the guidance of 10 C.F.R. Part 100. (Finding C-10.)

Mr. Sinderman also calculated the radiation dose to persons located at the nearest residence and the nearest approach of a public highway. His calculations showed those doses to be less than a microrem and 0.0029 millirem, respectively. (Finding C-8.) These doses also fall well within Part 100 limits.

Mr. Sinderman also calculated the dose that would be received by a person spending over 2 hours on the lakeshore at the nearest approach to the containment. This evaluation was appropriate because people often fish at that location. His calculations showed the exposure there to be 58 millirem, also well within the Part 100 limits. (Finding C-8.) Further, this calculation was conservative in that it did not take credit for the shielding provided by other buildings between the containment and that location. (Tr. 4303.) (We note here also that Mr. Sinderman did not take any credit for the attenuation provided by other buildings or trees in his calculations.) Mr. Sinderman also assured us that, during an accident, measures would be taken to remove people from the site and to prevent access to the site. (Tr. 4306.)

D. Conclusion

Based on the uncontroverted testimony, we conclude that the containment building provides adequate shielding to protect the public health and safety in the event of an accident which causes a substantial release of radioactive material from the spent fuel pool into containment. Accordingly O'Neill Contention II.E-4 is dismissed for lack of merit.

IV. O'NEILL CONTENTION II.D — RISKS FROM AIRCRAFT

A. Background

O'Neill Contention II.D states:

> The licensee has not adequately provided for the protection of the public against the increased release of radioactivity from the expanded fuel pool as a result of the breach of containment due to the crash of a B-52 bomber.

This contention was accepted as an issue in controversy because the United States Air Force conducts low-level training missions in the vicinity of the Big Rock Point Plant on a route known as the Bayshore Route.[7] Motions for summary disposition were filed by the Licensee and

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[7] In a letter to the Board dated April 5, 1984, the NRC Staff suggested that the issue of the risk of a B-52 crash at Big Rock Point was rendered moot by a letter it had received from Col. Dennis K. Bush, USAF, stating that the USAF intended to close the Bayshore facility by September 30, 1984. The Licensee, in a response dated April 24, 1984, argued that the issue should not be considered moot because (1) it expects our decision on the application for the license amendment in advance of September 30, 1984, and (2) the expressed intentions of the Air Force do not guarantee that the facility will, in fact, be closed by that date. We find Licensee's arguments persuasive. Therefore, we have considered the evidence on B-52 crashes in this decision.

the NRC Staff on this contention. In support of its motion, Licensee presented the deposition of Maj. (now Lt. Col.) Gary Betourne of the U.S. Air Force along with an analysis prepared by him in 1980 estimating the risk of a B-52 crash at the Big Rock Point Plant ("the 1980 USAF estimate").

In our February 19, 1982 Memorandum and Order (Concerning Motions for Summary Disposition), we denied the motions for summary disposition and identified eleven genuine issues of fact regarding the 1980 USAF estimate. We also accepted as genuine issues of fact the safety of the Big Rock Point Plant from aircraft used by the Ohio Air National Guard, which conducts low-level tactical training exercises in the area of Big Rock, and from the flights of small unscheduled aircraft. (Finding D-3.)

During the course of the hearing, evidence was presented regarding flights of other military fighter aircraft using training routes in the Big Rock Point area. Accordingly, we include consideration of that activity in weighing the risk of aircraft hazards under this contention.

Licensee presented five witnesses. (Finding D-4.) Lt. Col. Gary Betourne provided supplemental testimony responding to the eleven issues we posed regarding his B-52 risk analysis. In addition, a previous deposition taken of Lt. Col. Betourne and his B-52 risk analysis were introduced into evidence. (Licensee Exhibit 20; Tr. 4458, 4464.) Capt. William Hickey and Maj. John V. Lyczkowski addressed the activities of the Ohio Air National Guard and the military training routes. Mr. Anthony Tome and Mr. Robert Marusich addressed the probability of a breach of containment due to the crash of a small unscheduled aircraft at Big Rock Point. The NRC Staff presented the testimony of Dr. Kazimieras M. Campe, who addressed all aspects of the contention. (Finding D-5.) In addition, in response to Board orders reopening the record on the B-52 crash probability (LBP-84-12, March 6, 1984 (unpublished); LBP-84-12A, March 7, 1984 (unpublished)), the NRC Staff filed an affidavit of Dr. Campe in which the affiant provided a detailed, critical review of the 1980 USAF analysis prepared by Lt. Col. Betourne. (Affidavit of Kazimieras M. Campe Concerning Board Questions on B-52 Bomber Crash Probability, April 5, 1984.) Finally, Intervenors Christa-Maria and John O'Neill testified regarding flights observed by them around Big Rock Point. (Finding D-6.)

B. Applicable Law

Section 100.10 of 10 C.F.R. requires that reactors reflect through their design, construction, and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. Accidents attributable to aircraft hazard are encompassed by § 100.10. It is not intended, however, that nuclear power reactors be designed to meet this regulation for all theoretically possible accidents. Accidents of a sufficiently low probability of occurrence may be neglected in reactor design.

Section 2.2.3, "Evaluation of Potential Accidents," of the Standard Review Plan (NUREG-0800) ("SRP § 2.2.3") provides guidance as to the definition, from a probability standpoint, of those accidents that need not be considered in reactor design. SRP § 2.2.3 provides that accidents, including those involving aircraft, may be neglected in reactor design if the expected rate of occurrence of potential exposures in excess of 10 C.F.R. Part 100 guidelines is below the NRC Staff design objective of approximately 10^{-7} per year. Recognizing the difficulty of performing accurate calculations of the probabilities of low-probability events, SRP § 2.2.3 provides further that the expected rate of occurrence of potential accidents in excess of the 10 C.F.R. Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability of occurrence can be shown to be lower.

The Licensee and the NRC Staff advocate that the probability of aircraft accidents at Big Rock Point is not sufficiently high, using SRP § 2.2.3 as a guide, to warrant redesign of the Big Rock Point Plant. All postulated aircraft accidents are said to be of sufficiently low probability to be excluded from the design basis. (Finding D-35.) But as was indicated by the NRC Staff witness, Dr. Campe, a proper analysis under the Standard Review Plan requires consideration of the cumulative probability of all aircraft hazards, rather than a separate review of each hazard. (Finding D-35.) Hence, a consideration of the cumulative probability of the hazard to Big Rock Point from B-52s, the Ohio Air National Guard, military training routes, and unscheduled small aircraft, is relevant to determine the validity of the Licensee's and NRC Staff's positions.

C. Discussion

1. B-52s

Lt. Col. Gary Betourne is currently attached to the Office of the Assistant Secretary of Defense for International Security Policy. He prepared the 1980 USAF estimate during his former employment with Air Force Studies and Analysis. The 1980 USAF estimate was prepared in response to a request from the NRC Staff to validate the results of a prior USAF analysis prepared in 1971. The 1980 USAF estimate was based on B-52 crash data gathered since the time of the prior analysis. The 1980 analysis also included data on FB-111 aircraft, but Lt. Col. Betourne's supplemental testimony stated that FB-111s have not used the Bayshore route recently and there are no plans for such use before the projected closure of the Bayshore route in 1984. The 1980 USAF estimate also considered data regarding the number of runs and gross navigational errors occurring on the Bayshore route. At that time, the Bayshore route passed 5.7 nautical miles from Big Rock Point at its closest point. (Finding D-7.) Because of changes in the location, in abort criteria, and in utilization rate of the Bayshore route, Lt. Col. Betourne provided a new analysis of the B-52 crash probability in his supplemental testimony. (Betourne, ff. Tr. 4464 and ff. Tr. 4736.)

Lt. Col. Betourne estimated in his 1980 analysis that the probability of a crash at Big Rock is less than 10^{-5}. (Finding D-7.) The NRC Staff reviewed the 1980 USAF estimate as part of its review of the safety significance of aircraft hazards at Big Rock Point under its Systematic Evaluation Program. In addition, the analysis was independently reviewed and verified by Dr. Campe of the NRC Staff. He found that the 1980 USAF estimate was reasonable and that it provided an adequate basis for the B-52 crash probability estimates.

Dr. Campe has been evaluating such analyses for about 7 or 8 years. (Tr. 4731.) He pointed out several conservatisms in Lt. Col. Betourne's risk calculations that would compensate for any uncertainties used in the analysis. (Finding D-30.) First, it was assumed that any B-52 that strayed outside the corridor of the Bayshore route (a "gross navigational error") would overfly the plant. Dr. Campe stated that it was reasonable to believe that a navigational error could just as likely cause an errant B-52 to fly away from the Big Rock Point Plant. Also, the assumption implies that every error will remain uncorrected. (Finding D-30.) A second significant conservatism in the 1980 USAF estimate is its assumption of a 3-nautical-mile-square area, centered on the Big Rock Point Plant, in which a crash would be deemed to damage the plant. Dr. Campe stated that the effective plant impact area, which SRP § 3.5.1.6 defines as including the plant area, the shadow area behind the plant in reference to an aircraft approaching along a descent angle, and the skid area in front of the plant in reference to an aircraft approaching along the same angle, is no more than 0.16 square nautical mile. This is about 56 times smaller than the 9 square nautical miles assumed in the 1980 USAF estimate. (Finding D-30.)

Dr. Campe concluded that if the conservatisms were replaced by the more realistic estimates, the annual probability that a B-52 would crash into the Big Rock Point Plant would be much less than 10^{-8}. (Finding D-30.)

In our Memorandum and Order (Concerning Motions for Summary Disposition), dated February 19, 1982, we set forth eleven genuine issues of fact. In his supplemental testimony presented at the hearing, Lt. Col. Betourne addressed those eleven issues. Particularly, we were concerned about a B-52 which overflew the plant in July 1979. We inferred from Lt. Col. Betourne's deposition that the error resulted from the use of the Big Rock Point Plant as an offset aiming point, that is, an accurate range and bearing radar point used for aiming the simulated release of weapons. Apparently the crew of the aircraft used the plant as a direct aiming point, rather than an offset, and thus flew directly over the plant. Though the Air Force has since prohibited the use of the Big Rock Point Plant as an offset aiming point, we questioned whether such a mistake may nonetheless recur, inasmuch as the plant remains a highly visible landmark.

Lt. Col. Betourne explained that while the plant still remains a suitable radar return for navigational cross-checking, the B-52 air crews are no longer provided with the detailed range and bearing information that would enable them to use Big Rock Point as an offset aiming point for their training missions. Further, to discourage a navigator from developing the plant as an offset point, photographs of the radar scope, taken automatically during the flight at a pre-set rate, are reviewed to discover any use of an illegal offset. (Finding D-8.)

Another issue we raised regarding Lt. Col. Betourne's 1980 estimate was the extrapolation of 2 months of data to derive the annual number of sixty gross navigational errors assumed in the analysis. We challenged the adequacy of this sample and sought to have additional statistical verification. Lt. Col. Betourne stated that he has since learned that in the year of interest for his analysis, there were actually only thirty-six gross navigational errors. Thus, the assumption of sixty errors was conservative. (Finding D-10.)

We also questioned whether the vulnerable crash area of 3 nautical miles square (= 9 square nautical miles) assumed in the analysis was conservative. Lt. Col. Betourne assured us that it was indeed conservative. He stated that a more realistic yet still conservative assumption would be to use the expected debris area of a crash on smooth terrain.

This would reduce the estimate of vulnerable area to about 0.10 square nautical mile.[8] (Finding D-12; Betourne, ff. Tr. 4464, at 12.)

Lt. Col. Betourne also assured us, in addressing the other issues we posed, that there is no reason to assume that low-level missions are more hazardous than other flight activities, that a random communication failure due to any cause does not add significantly to the risk, and that the crash data used in his analysis account for crashes due to all causes and thus would account for any dependency between the probabilities of navigational error and a crash. (Findings D-13, D-14.)

In his critical review of the 1980 USAF analysis, Dr. Campe compared the 1980 crash rate on low-altitude runs, 1.3×10^{-5} crashes per run, with the 1982 estimate calculated by Lt. Col. Betourne in his supplemental testimony. (Campe Affidavit at 2; Betourne, ff. Tr. 4736, at 1.) Campe concluded that the closeness of these estimates, as well as Staff experience from past reviews of military aircraft, supports the conclusion that crash rates do not fluctuate orders of magnitude from year to year. (Campe Affidavit at 2.) He then explained in detail conservatisms in the 1980 analysis; these involved the estimation of navigational error rate, the frequency with which loss of communications causes an overflight, and the size of the plant impact area. These conservatisms lead to a probability estimate of about 9×10^{-9} that a B-52 will crash into the plant each year. (Campe Affidavit at 2-6.)

Dr. Campe then proceeded to calculate a *realistic* estimate of the probability that a B-52 will crash into the plant, by eliminating the conservatisms from the calculations. First he reduced the estimate of crash per run by multiplying 1.3×10^{-5} by 0.06, on the grounds that only about 6% of all B-52 crashes occur during low-altitude training runs. We find this adjustment to be inappropriate. Lt. Col. Betourne testified that "the 1972-79 crash data constituted all crashes of B-52s and FB-111s *while on low-level training runs,* whatever the cause." (Emphasis supplied.) (Betourne, ff. Tr. 4464.) Since Lt. Col. Betourne's data were

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[8] Lt. Col. Gary P. Betourne testified that the expected debris area caused by a crash on relatively smooth terrain would be "600 x 6000 feet or about 0.09 square miles." (ff. Tr. 4464, at 12.) In the proposed findings of the Licensee, this area was said to be 0.009 square miles. (Consumers Power Company's Draft Opinion and Proposed Findings of Fact and Conclusions of Law on Certain Contentions, December 19, 1983, at 60.) The NRC Staff appears to accept 0.09 as the fraction of a square mile occupied by an area 600 x 6000 feet. (NRC Staff Response to Intervenor John O'Neill's Proposed Findings of Fact on • • • Aircraft Hazards Contentions, March 1, 1984, at 9.) We accept Lt. Col. Betourne's statement that the expected debris area would be 600 x 6000 feet, because we assume that he obtained those figures from accepted Air Force documentation. When we make the calculation Lt. Col. Betourne made to get from square feet of debris area to fraction of square nautical mile occupied by debris area, we obtain 0.0975; in our view this should have been rounded up to 0.10, not rounded down by dropping the last two digits. As we noted, *supra*, the Staff estimated the vulnerable area to be no more than 0.16 square nautical mile.

derived from crashes during low-level runs only, it is incorrect to multiply his result by the fraction of *total* crashes occurring on low-level runs.

Because the effective plant area, according to Dr. Campe, is 0.16 nautical square mile rather than 9 nautical square miles, Dr. Campe reduced the 3-mile flight segment by the square root of the ratio 9/0.16, or a factor of 7.5. He thus obtained a probability of about 9.3×10^{-4} of being in the appropriate segment of the route for crashing into the Big Rock Point Plant. Finally, the navigational error probability was reduced by a factor of 2 by Campe, since it is reasonable to assume that flight errors could occur away from the plant as well as in the direction of the plant. Dr. Campe's calculation of a realistic probability involved the following: (incorrect probability of a crash during a low-level run) x (probability of a navigational error that would cause the plane to overfly the plant) x (incorrect probability that a power failure would prevent communication with the errant plane) x (probability that the crashing plane will strike the plant) x (number of runs) =

$$(7.8 \times 10^{-7})(1 \times 10^{-2})(1.7 \times 10^{-3})(9.3 \times 10^{-4})(2986) = 3.7 \times 10^{-11}.$$

We cannot accept Dr. Campe's estimate of a realistic probability, however, because we believe it was erroneous to multiply the probability of a crash during a low-level run by the fraction of total crashes which occur during low-level runs. To correct this error we have calculated a realistic probability by using the crash rate on low-level runs, 1.3×10^{-5}, without dividing by 0.06. The result obtained by us for realistic probability of a B-52 crash into the Big Rock Point Plant is 6.1×10^{-10} per year.

Intervenors presented the testimony of Dr. Arthur J. Schwartz, an expert in the area of probabilistic risk assessment. Dr. Schwartz emphasized the need to use "common sense" and to incorporate all relevant, available experimental data when attempting to use probabilistic risk assessment theory as a tool to evaluate plant safety. (Schwartz Deposition of November 16, 1983, at 7-8, 46, 48.) He directed several criticisms at the USAF 1980 B-52 crash analysis.

Dr. Schwartz criticized the USAF estimate because of the small number of samples (observed crashes) used to obtain the estimate. (Schwartz Affidavit at 9-10, 36-37.) It is true that the accuracy of probabilistic estimates is linked to sample size. This fact is recognized in the guidance provided the Staff by SRP § 2.2.3, which says:

> [B]ecause of the low probabilities of the events under consideration, data are often not available to permit accurate calculation of probabilities. Accordingly, the expected rate of occurrence of potential exposures in excess of the 10 C.F.R. Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

(*See* p. 2.2.3-2 of Reference 3 attached to Campe Testimony, ff. Tr. 4655.) Based on the foregoing guidance regarding the approximate nature of low-probability estimates and the acceptability of qualitative arguments, the NRC Staff viewed the sample size used in the USAF analysis as not being a significant deficiency, and concluded that the probability was well within the acceptable criteria of SRP § 2.2.3.

Dr. Schwartz also alleged that the formula used in the USAF analysis, which called for multiplying probability values, was incorrect. (Schwartz Affidavit at 13.) The methodology used by Lt. Col. Betourne, however, is expressly endorsed in SRP § 3.5.1.6, which concerns the assessment of hazards from aircraft. (*See* p. 3.5.1.6-3 of Reference 10 attached to the Campe Testimony, ff. Tr. 4655.)

Dr. Schwartz also criticized the USAF analysis for ignoring the possibility that some of the variables might be statistically dependent. (Schwartz Affidavit at 21-25, 42-44.) The possibility of dependency between the variables *was* considered in the USAF analysis, however, and was judged either to be nonexistent or, if present, to have a negligible effect on the results because of the conservatisms built into the analysis. (Betourne, ff. Tr. 4464, at 18; ff. Tr. 4655, attachment to Campe Testimony, USAF Memorandum dated January 2, 1980, item 6.)

Dr. Schwartz asserted, further, that the USAF analysis failed to consider factors that could contribute to B-52 crashes, such as drunkenness of crew members, insanity in the crew, sabotage, "St. Elmo's fire, and weird things of that sort."[9] The USAF analysis, however, considered all low-level B-52 crashes, regardless of their cause. (Betourne, ff. Tr. 4464, at 8 and 18; Tr. 4471.) Therefore the data base includes such factors, to the extent they may be relevant to crashes.

We acknowledge the validity of Dr. Schwartz's criticisms from an academic standpoint. In the scientific arena, statistical standards used to test hypotheses can and should be rigorously applied. In setting standards by which national policy can be enacted, however, it is often neither practicable nor possible to demand the same rigor in decisionmaking that would be demanded for reaching a scientific conclusion. Thus, the

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[9] While some of the very rare events, or "weird things," mentioned by Dr. Schwartz could conceivably lead to the crash of a B-52, St. Elmo's fire probably is not one of them. St. Elmo's fire is a corona discharge from pointed conducting objects and is observed frequently on the surfaces of aircraft flying in the vicinity of thunderstorms; it is not considered hazardous so long as the aircraft does not discharge fuel. Dr. Schwartz appears not to distinguish St. Elmo's fire from ball lightning, however. (Schwartz Deposition at 36.) Ball lightning is not observed as commonly as St. Elmo's fire and is less well understood. It is usually seen almost simultaneously with a cloud-to-ground lightning discharge and usually is observed within a few yards of the ground. What has appeared to be ball lightning has, however, been observed in the cabins of aircraft, and while it has startled crew members, as Dr. Schwartz suggests it might (Schwartz Deposition at 51), we are unaware of a plane crash having been attributed to it. (Martin A. Uman, *Understanding Lightning*, ch. 13 (1971).)

guidelines set forth in SRP § 2.2.3 attempt to minimize the risk from nuclear power while simultaneously recognizing the impossibility of obtaining data on rare events sufficient to allow the rigorous application of statistical standards. Although Dr. Schwartz (and Mr. O'Neill) might prefer that rigorous scientific standards always be applied by the NRC, the agency must often use its judgment in order to discharge its statutory responsibility. We find that the Staff guidance which employs conservative estimates of the probability of very rare events, projected from the small sample sizes that are often the only data available, provides an adequate basis for protecting the public health and safety.

We conclude that the 1980 USAF estimate is reasonable and therefore accept as an upper bound estimate of the probability that at B-52 bomber on a low-level mission at the Bayshore route will crash into the Big Rock Point Plant a value on the order of 10^{-8} per year. Further, we accept as a realistic estimate of that probability a value on the order of 10^{-9}, which we calculated from Dr. Campe's figures.

We note that the 1979 move of the Bayshore route to its present location should contribute significantly to the reduction of the risk to Big Rock Point from B-52 aircraft. The present route now passes Big Rock Point at 11.5 nautical miles at its closest point, which more than doubles the distance to the closest point of approach of the former route location. Along with this change in route location, the Air Force has changed its criteria for notifying aircraft on the Bayshore route of a navigational error. The route "corridor" now extends only 3 nautical miles either side of the center line of the route, whereas before, the corridor was 4 nautical miles either side, thus assuring tighter operational control over navigation errors. (Finding D-17.) We note that, so far as we are aware, there have been no reports of any B-52 overflights since the route was moved to its present location in 1979.

2. *Ohio Air National Guard*

Concerns about flights of the Ohio Air National Guard in the vicinity of the Big Rock Point Plant arose out of a flight of two of its aircraft, which plant personnel said flew over the plant at treetop levels on July 22, 1981. Captain William Hickey, formerly of the Ohio Air National Guard, testified that he believes the witnesses were mistaken. Capt. Hickey testified that he had led that flight of two planes and that at no time did he or his wingman fly over the Big Rock Point Plant. (Finding D-18.) He stated that lay persons often misestimate the range at which they see A-7D aircraft because these planes are larger than most single-engine jet aircraft. Their size, the noise they generate, and their high speed give the impression that the aircraft are closer to the observer than they actually are. (Finding D-19.)

The aircraft on the July 22, 1982 flight were conducting exercises in what is known as the Wolverine Military Operations Area (MOA). This is an area in which military aircraft are permitted to perform high-speed flight at low levels for purposes of practicing tactical maneuvers. The Wolverine MOA is roughly 45 miles square and encompasses the Big Rock Point Plant. Its use is managed by the Ohio Air National Guard, specifically by Major John Lyczkowski, who was preceded in this responsibility by Capt. Hickey. Both officers testified as to the procedures which must be followed by pilots of the Ohio Air National Guard or other Air National Guard units who request activation of the Wolverine MOA. These procedures include briefing pilots about designated no-fly areas within the MOA which prohibit direct overflight of the Big Rock Point Plant below 5000 feet above sea level (about 4500 feet above ground level) and further prohibit flight below 1500 feet above ground level within 2 miles of the plant. (Finding D-20.) Further, it was noted that use of the Wolverine MOA is permitted only when conditions allow at least 5 miles of visibility. (Finding D-23.)

Dr. Campe testified regarding the NRC Staff's analysis of the probability of the crash of a military aircraft at Big Rock Point in connection with the activities of the Wolverine MOA. An upper-bound estimate of that probability, based on extremely conservative assumptions, was found to be on the order of 10^{-6} (7×10^{-7}). When more realistic assumptions were used which removed some of the conservatisms based on reasonable qualitative judgments, an estimate of the probability derived was on the order of 10^{-8} (7.6×10^{-9}). (Finding D-31.)

The methodology used by Dr. Campe to derive the upper-bound and realistic estimates was to multiply together the flight frequency, the probability of the aircraft crash, and the probability of the crash occurring at Big Rock Point. Dr. Campe explained the different assumptions used in the two estimates. In the upper-bound estimate, a conservative assumption of 1500 aircraft per year operating in the MOA was used. In the more realistic estimate, an assumption of ninety-nine aircraft, based on actual data for 1980, was used. (Finding D-32.)

The probability of the crash occurring at the Big Rock Point Plant was determined by dividing the crash area of interest by the total area of the Wolverine MOA. In the upper-bound estimate, the crash area was determined by equating the total potential crash area with the effective plant impact area, i.e., the maximum potential range for a crash from a low-altitude flight. The realistic estimate considered an impact area based on the actual plant area together with the skid and shadow areas. (Finding

D-32.) An overall conservatism in both estimates is the assumption of uniform distribution of flight paths throughout the MOA, thus discounting the no-fly restrictions around Big Rock which can reasonably be expected to reduce the number of flights around Big Rock. (Finding D-32.)

Captain Hickey, who has experience in performing probability analyses, estimated the annual probability of a crash of an A-7D aircraft of the Ohio Air National Guard at Big Rock Point while using the Wolverine MOA to be on the order of 10^{-8}. (Finding D-22.) Capt. Hickey based his probability on the historical accident rate of A-7D aircraft, times the critical flight time that the plant would be vulnerable, given one overflight per year. His independent analysis confirms the reasonableness of the estimates developed by Dr. Campe.

3. *Military Training Routes*

In the course of this proceeding, it was determined that in June 1983, military aircraft from an Air National Guard unit other than the Ohio unit flew near the plant while waiting to enter a low-level military training route, VR-1634, which passes 5.2 miles from the Big Rock Point Plant at its closest point. (Another training route, VR-1636, passes at 33.4 miles.) Such flights, which do not activate the Wolverine MOA and which do not occur in the military training route, are not bound by the no-fly restrictions that apply to the Wolverine MOA; they are bound only by FAA regulations which limit low-level flight to not less than 500 feet above the top of the stack at the Big Rock Point Plant. (Tr. 4395-96, 4428-30.)

The Ohio Air National Guard also serves as the scheduling unit for military training routes VR-1634 and VR-1636. Major Lyczkowski testified that since the June 1983 occurrence, the Air National Guard has instituted the practice of requesting all units which schedule the use of those routes to respect the no-fly areas designated for Big Rock Point in the Wolverine MOA. (Finding D-21.)

Dr. Campe's testimony included a probability analysis of activity associated with military training route VR-1634. (He did not address route VR-1636, apparently discounting any risk from that route as insignificant due to its distance from the plant.) As with his Wolverine MOA crash probability analysis, Dr. Campe's analysis of the VR-1634 crash probability also produced two estimates: an upper-bound estimate on the order of 10^{-6} (5.7×10^{-7}), and a realistic estimate, which discounted some of the conservatisms used in the upper-bound estimate based on reasonable qualitative judgments, on the order of 10^{-9} (2.5×10^{-9}). (Finding D-33.)

The methodology used here by Dr. Campe was identical to that for the Wolverine MOA analysis: frequency of flights, times probability of crash, times the probability of the crash occurring at Big Rock Point. In the upper-bound estimate, Dr. Campe assumed that every flight using VR-1634 overflies the Big Rock Point Plant, which amounted to a projected annual rate of 1500 planes. Flight records maintained by the Ohio Air National Guard indicate that the actual flight frequency is considerably less, about 240 planes annually. For the realistic estimate, Dr. Campe assumed only one overflight per year from route VR-1634, which he believed to be reasonably conservative. (Finding D-36.)

4. Unscheduled General Aviation

The NRC Staff's Systematic Evaluation Program evaluated the risk of an aircraft crash at the Big Rock Point Plant resulting from general aviation to be about 8.7×10^{-7} per year. Using this crash rate and a projected 71,000 aircraft operations per year at the Charlevoix Airport, Dr. Campe obtained a very conservative estimate of 8.5×10^{-4} crash per year onto the plant from aircraft using the airport. An absurd conservatism in this estimate is the assumption that all 71,000 operations at the airport result in an overflight of the plant. (Finding D-35.)

Mr. Anthony E. Tome, Jr., a consulting engineer to Licensee, performed an elaborate analysis of the probability of the crash of an unscheduled general aviation flight into the containment at Big Rock Point. His analysis concluded that the probability of such an event is 1.33×10^{-6}. (Finding D-25.) However, his methodology was unduly conservative. In the absence of actual observed data, Mr. Tome assumed that there were more than 54,000 overflights of the plant per year by small aircraft, which would amount to about 1 overflight every 10 minutes. (Finding D-30.) Consequently, we view Mr. Tome's estimate to be an extremely conservative upper-bound value which supports Dr. Campe's upper-bound estimate of 5×10^{-7}.

Intervenors Christa-Maria and John O'Neill testified as to low-level flights they had personally observed in the vicinity of Big Rock Point. (O'Neill, ff. Tr. 4740; Christa-Maria, ff. Tr. 4744, 4744-50.) Sightings of three aircraft by Christa-Maria involved a B-52, a helicopter, and a small unidentified aircraft which, by the witness' own admission, were not observed to overfly the plant. One additional aircraft, a red biplane which was performing acrobatic maneuvers, was seen by Christa-Maria to fly over the plant's stack. (Tr. 4745.) We conclude that, with the possible exception of the acrobatic biplane, these flights would be accounted for by the unscheduled aviation overflights assumed by Dr. Campe and Mr.

Tome; the increased risk from the single "acrobatic" flight of a biplane would be accounted for by the excessively large number of overflights assumed by Dr. Campe and Mr. Tome in preparing their probability analyses.

5. Cumulative Probabilities

As indicated earlier, Dr. Campe stated that an appropriate review of the hazards to Big Rock Point from aviation activity requires summing the probabilities and then measuring the resulting cumulative probability against the standards of SRP § 2.2.3. To perform this addition, we begin by adding together those estimates which we believe are conservative yet reasonably founded upper-bound values.

Specifically, the Licensing Board accepts the 1980 USAF estimate (less than 10^{-8}) and Dr. Campe's three upper-bound estimates regarding the risks from the aviation activity of the Wolverine MOA (7×10^{-7}), military training route VR-1634 (5.7×10^{-7}), and small unscheduled aircraft (5×10^{-7}). The addition of these probabilities gives a sum of about 2×10^{-6}. Although this sum fails to meet the NRC Staff design objective of approximately 10^{-7}, we consider 2×10^{-6} to be "approximately 10^{-6}" as that acceptance standard is used in SRP § 2.2.3. An expected rate of occurrence "of approximately 10^{-6} is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower." (SRP Rev. 2 at 2.2.3-2.) We look now to the evidence to determine whether this criterion is met.

We obtained from Dr. Campe's figures a realistic probability of a B-52 crash into the Big Rock Point Plant of 6.1×10^{-10}. For his risk estimates associated with the Wolverine MOA and VR-1634, Dr. Campe provided more realistic estimates of 7.9×10^{-9} and 2.4×10^{-9}, respectively. Further, we accept Dr. Campe's judgment that 1×10^{-8} is a realistic estimate of the risk from small unscheduled aircraft.

Based on the reasonable qualitative arguments presented by Dr. Campe regarding the conservatisms in these estimates, we conclude that Dr. Campe's reasoning is sound and we accept his realistic estimates of those probabilities. Specifically, in the B-52 analysis, Dr. Campe used a more realistically sized target area, and he made a more realistic assumption about the effect of a navigational error. In the Wolverine MOA analysis, Dr. Campe estimated the number of flights based on recorded data and used a crash area that more closely approximated the actual expected area to be impacted by a crash. In his VR-1634 risk estimate, Dr. Campe assumed only one flight per year, which he considered conservative yet more realistic than the assumption of his upper-bound estimate that all flights using VR-1634 overfly Big Rock Point. Finally, Dr. Campe stated that a more realistic assumption about the number of overflights by small, unscheduled aircraft would make the probability of a crash from such aircraft about 1×10^{-8}. (Finding D-35.)

The sum of these probability estimates provides us the cumulative realistic probability of an aircraft crashing into the plant: about 2×10^{-8} per year. This realistic estimate is 2 orders of magnitude lower than the upper-bound estimate. Thus, the aircraft crash probabilities satisfy the criteria for acceptability set forth in SRP § 2.3.3.

D. Conclusion

We conclude that the evidence has demonstrated that the risk from aircraft to the Big Rock Point Plant is sufficiently low that it need not be considered further in the design of the plant, and O'Neill Contention II.D is dismissed.

V. O'NEILL CONTENTION II.C — SEISMIC STABILITY OF OVERHEAD CRANE

A. Background

O'Neill Contention II.C states:

> Is the spent fuel pool safe from a rupture which might be caused by a drop of a spent fuel transfer cask or of the overhead crane?

This contention was admitted by the Licensing Board in its "Memorandum and Order (Concerning Motions for Summary Disposition)," dated February 19, 1982. LBP-82-8, 15 NRC 299 (1982). The Board admitted as a genuine issue of fact under this contention the question of whether the overhead crane, used for handling fuel assemblies and casks, has been designed adequately to withstand seismically induced ground motion without falling into the Big Rock Point spent fuel pool.

To address this question, Licensee presented the testimony of seven witnesses, Messrs. Norman, VandeWalle, Chan, Beachum, Campbell, Yanev and Dr. Eggenberger. (Finding E-3.) The NRC Staff submitted the testimony of Drs. Cheng and Chokshi, and Dr. Reiter. (Finding E-4.) Intervenors presented no testimony, relying instead on cross-examination of Licensee and Staff witnesses. (Finding E-5.)

B. Applicable Law

In 1977, the NRC Staff initiated the Systematic Evaluation Program (SEP) to, among other things, reevaluate the seismic design criteria and safety of older plants, including Big Rock Point, which had been built prior to current NRC safety regulations and criteria. The SEP plants are being reevaluated against selected safety issues including seismic design considerations which require that structures, systems and components important to safety shall be designed to withstand the effects of seismic loadings.

As part of the SEP, the NRC determined that an alternative methodology to that set forth in 10 C.F.R. Part 100, Appendix A was needed to make a realistic determination of the appropriate design basis earthquake, considering the seismic hazard at the SEP plants' sites. (Findings E-10, E-11.) Because Appendix A was not intended to apply retroactively, the Staff decided not to apply the Commission's current standards for determining the geological characteristics and seismicity to plants already built and operating. Appendix A to 10 C.F.R. Part 100 was proposed in 1971, 36 Fed. Reg. 22,601 (November 25, 1971), and adopted in 1973, 38 Fed. Reg. 31,279 (November 13, 1973). Prior to the proposal and promulgation of this regulation, the NRC had no specific seismic standards. *See Department of Water and Power of the City of Los Angeles* (Malibu Nuclear Plant, Unit No. 1), 3 AEC 179, 183 (1967.)[10] The Commission did not make Appendix A applicable to plants that had received their operating licenses prior to its proposal and enactment (38 Fed. Reg. 31,279). For these reasons, 10 C.F.R. Part 100, Appendix A need not be applied to the Big Rock Point Plant. *See Dairyland Power Cooperative* (La Crosse Boiling Water Reactor), LBP-83-23, 17 NRC 655, 658, *aff'd,* ALAB-733, 18 NRC 9 (1983). Under these circumstances, the Board believes that the site-specific spectra developed for the SEP for the Big Rock Point site is the appropriate seismic motion for evaluating the seismic structural adequacy of the overhead crane. *See La Crosse,* LBP-83-23, *supra,* 17 NRC at 658-59.

Once the appropriate ground motion for the site has been defined, it is necessary to model the plant's response to that motion. Current NRC guidance, Standard Review Plan (SRP) § 3.7.2, was used to develop the floor response spectra and to model seismically the Big Rock Point reactor building. At the time many of the structural analyses of the overhead crane were performed on behalf of Licensee, SRP § 3.7.2, "Seismic

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[10] This case only supports the proposition that the NRC had no specific seismic standards at that time.

System Analysis" (June 1975), was in effect. Subsequently, SRP § 3.7.2 was revised and reissued as SRP § 3.7.2, "Seismic System Analysis," Rev. 1 (July 1981).

The floor response spectra are used as input into structural analyses of the overhead crane; this procedure models this structure in accordance with SRP § 3.7.2. The stresses which result from the imposition of seismically induced loadings on the crane structure are compared with the crane component materials' allowable stresses. The American Institute of Steel Construction ("AISC") Manual specifies the allowable stress for structural steel components and is referenced in SRP § 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments" (July 1981). SRP § 3.8.3.II.5, provides that the allowable stresses for steel materials may be increased above the loadings specified in the AISC Manual for the purpose of seismic analysis, which contemplates an event expected to occur, at most, only once.

It is against these guidelines that Licensee's evidence on the seismic structural adequacy of the overhead crane must be weighed, and a determination made as to whether there is reasonable assurance that the overhead crane can be operated without endangering the health and safety of the public, consistent with 10 C.F.R. § 50.57(a)(3)(i).

C. Discussion

The matters involving the seismic structural analysis of the overhead crane may conveniently be separated into three issues: (1) the definition of the appropriate seismic motion for the Big Rock Point site, (2) the translation of the ground motion for the Big Rock Point site into floor response spectra useable as input in the structural analyses of the crane, and (3) the structural analyses of the overhead crane.

1. Seismic Ground Motion

The first difficulty encountered in assessing the structural adequacy of the Big Rock Point overhead crane is the determination of the earthquake motion the crane should be expected and required to withstand without falling into the Big Rock Point spent fuel pool. The Big Rock Point Plant was designed and constructed in the early 1960s (Finding E-7), in accordance with the then-existing seismic criteria inherent in the Uniform Building Code, namely 0.025*g* static for all major structures and 0.05 for the reactor containment vessel. (Finding E-8.) This design basis is being reevaluated by the NRC Staff under its SEP.

A design basis earthquake, site-specific response spectra and peak ground acceleration values have been established for the Big Rock Point site under the SEP. (Findings E-11, E-12, E-18.) The site-specific spectra were developed by Lawrence Livermore Laboratory on behalf of the NRC Staff as an attempt to make a realistic determination of the appropriate earthquake based upon the true seismic hazard of the Big Rock Point site. (Findings E-11, E-12.)

Dr. Leon Reiter appeared before this Board to explain the methodology used to develop the peak ground acceleration values and site-specific spectra for the SEP plants, including Big Rock Point. The results of the site-specific program are set forth as uniform hazard spectra, where each spectral amplitude has the same subjectivity probability of being exceeded.[11] All potential earthquakes contributing to the seismicity at the site were considered using appropriate seismicity, attenuation and exposure models. (Finding E-15.)

Since there is insufficient historical data on earthquake experience in the central United States, judgment must be exercised in the selection and limitations of certain data and empirically derived parameters. Accordingly, the methodology relies heavily an expert opinion. (Findings E-15 through E-17.) The study solicited expert opinion in key seismic input parameters, including seismic zonation, frequency of earthquake occurrences, upper magnitude cutoff, and characterization and attenuation of ground motions. (Finding E-16.) The experts who contributed to the study are well known in the field of geophysics, and include authorities such as Dr. G.A. Bollinger, president of the eastern section of the Seismology Society; Dr. P.W. Pomeroy, the current chairman of the Committee on Seismology of the National Academy of Sciences, and Dr. O.W. Nuttli, a leading authority on earthquakes east of the Rocky Mountains.

The site-specific spectra developed by Lawrence Livermore Laboratory for Big Rock Point were anchored at 0.08*g*. The NRC Staff, pursuant to its policy of setting minimum deterministic levels for each SEP site, raised the site-specific ground acceleration to 0.105*g*. The site spectra anchored at 0.105*g* were approved and supplied to Licensee in June 1981. (Finding E-18.)

At the hearing, the Board was concerned with the site-specific spectra methodology's treatment of amplification. (Tr. 4995-5014.) The site-specific spectra for the Big Rock Point site did not include site-specific

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[11] The estimates used to generate the spectra were those of a panel of experts.

factors for amplification due to shallow soil conditions, a soil characteristic which theoretically could significantly increase the seismic motion the plant would experience.

The responses to our questions indicate that the amplification problem is a difficult one, with a great difference of opinion as to which analyses should be done. The amount of amplification at shallow soil sites have varied significantly, and may have been affected by phenomena such as focusing and radiation. (Finding E-20.) Moreover, the difficulties posed by amplification due to the lack of data and theoretical understanding are not limited to the probabilistic analysis used to generate the site-specific spectra; they also handicap the type of deterministic analyses required by 10 C.F.R. Part 100, Appendix A.

As a corollary, the responses to our questions have also persuaded us that the site-specific spectra approved by the NRC Staff and provided to Licensee in June 1981 need not be altered to reflect the possibility of amplification due to the Big Rock Point site's shallow soil conditions. Dr. Reiter evaluated the possibilities of amplification at the Big Rock Point site and concluded that the uncertainties allowed in the site-specific spectra adequately accommodate the possibility of amplification. (Finding E-21.)

Subsequent to the development of the site-specific spectra, Chen and Bernreuter of Lawrence Livermore Laboratory were requested to compute the soil amplification at Big Rock Point using empirical and theoretical techniques. Although the site-specific theoretical technique used to compute the amplification has several uncertainties, the results indicated essentially no amplification for the frequencies important for evaluating the structural adequacy of the crane, i.e., frequencies less than 4 Hz; amplification gradually increases to a factor of 2 for frequencies greater than 10 Hz. (Finding E-21.)

Applying a factor-of-2 amplification at all frequencies to an appropriately computed rock spectrum at Big Rock Point results in a spectrum approximately equal to the original recommended spectrum anchored at 0.105*g*. (Findings E-18, E-21; Reiter, ff. Tr. 4902, Attachment 1, at 11; Reiter, Tr. 5000-01, 5009-11.) Furthermore, the Staff witnesses testified that the amplification factor could be doubled again, and at certain frequencies the site-specific spectra would be about equal to the interim seismic design criterion used by Licensee to initiate the seismic evaluation of the containment crane. (Reiter, Tr. 5001-02; Chokshi, 5006-07). Dr. Reiter also testified that the seismic hazard at the Big Rock Point site is so low that there is very little chance that there will be any earthquake ground motion of significance. (Finding E-22.)

At the hearing, the Board also raised questions concerning how uncertainties were accommodated in the results of the SEP site-specific program. As Dr. Reiter explained, uncertainty concerning input parameters was taken into account in each experts' distribution of earthquake probability. The final results of each expert were then integrated into a single hazard curve by means of weights supplied by each expert. (Finding E-16.)

The methodology used is designed to accommodate the difficulties associated with estimating earthquake ground motion in a region known for its lack of seismic activity. Accordingly, the site-specific spectra approved by the NRC Staff and provided to the Licensee in July 1981 are the appropriate seismic ground motion for evaluating the seismic adequacy of the overhead crane.

The Board is aware, however, that Licensee has not used the site-specific spectra for all of the crane structural analyses. Licensee began the structural analyses before the SEP site-specific spectra were available, and selected to use as an interim seismic design criterion the ground response spectra recommended by NRC Regulatory Guide 1.60 anchored at 0.12*g* (hereinafter "interim criterion"). The use of the interim criterion is acceptable since this criterion bounds the site-specific spectra at all frequencies. (Findings E-13, E-14, E-19.) Stresses on the reactor building and overhead crane induced by the site-specific spectra would be less than stresses induced by the earthquake loadings associated with the interim criterion. (Findings E-26, E-29, E-77, E-80.)

2. *Floor Response Spectra*

The ground motions caused by an earthquake, and represented by ground response spectra, introduce vibratory motions into the base of structures. These motions in turn induce vibrations throughout the entire structure. The characteristics of vibratory motions at different levels or floors of the structure depend on the dynamic characteristics of the structure and are represented in floor response spectra. These floor response spectra are used as seismic input for the structural analysis of equipment such as the overhead crane. (Finding E-23.)

D'Appolonia Consulting Engineers performed seismic analyses on behalf of Licensee which, among other things, generated floor response spectra at various elevations of the reactor building. (Finding E-24.) Floor response spectra for the support locations of the overhead crane were generated using both the interim criterion and the SEP site-specific response spectra. (Finding E-25.) The site-specific floor response spectra, in the frequency range of importance to the evaluation of the overhead crane, indicate acceleration responses approximately 50% of the floor response spectra computed using the interim criterion. (Finding E-26.)

D'Appolonia generated the floor response spectra using seismic analysis which models the reactor building using three-dimensional beam elements. (Finding E-27.) The seismic analyses, consistent with SRP §§ 3.7.2.1.a(1), 3.7.2.5 and 3.7.2.6.b (Finding E-31) considered torsional, rocking, and translational response and used an adequate number of degrees of freedom in accordance with SRP §§ 3.7.2.1.a(3) and (4) (Findings E-27, E-36); also they complied with SRP § 3.7.2.1.a(7), by accounting for significant effects, such as piping interaction, externally applied structural restraints, and the hydrodynamic loads generated by the spent fuel pool water (Finding E-32). The seismic modeling of the reactor building accounted for the possibility of nonlinear responses due to the presence of Fesco boards along expansion joints which isolate the reactor cavity structure and the horizontal shear key. The analysis was first performed by modeling the reactor building as a single-stick, neglecting the presence of expansion joints and treating the reactor cavity, spent fuel pool, and steam drum enclosure as monolithic. A second, multi-stick analysis was performed in response to a Staff request for consideration of the interaction at the expansion joints. The results of the two analyses were not dissimilar, with the single-stick model generating the most conservative input for the evaluation of the overhead crane. (Findings E-34, E-35.)

At the hearing, questions were raised as to whether D'Appolonia had complied with the intent of SRP § 3.7.2.11, since the seismic analyses did not follow the SRP recommendation to include in the analyses an accidental torsion moment equal to the product of story shear times 5% of the dimension of the building. This question was resolved by Dr. Eggenberger's testimony that the modeling of the plant structures was performed with careful consideration of the structures' geometrical mass and stiffness distribution, and that the accidental torsion accounted for in the model was approximately 4 to 7 times the factor for accidental torsion recommended by the Standard Review Plan. (Eggenberger, ff. Tr. 4784, at 24; Eggenberger, Tr. 4804-06.) In all other respects the analyses performed by D'Appolonia adequately accounted for torsional effects as recommended by the Standard Review Plan. (Finding E-36.)

At the hearing, Intervenors inquired into the damping values used in modeling the reactor building. The seismic modeling of the reactor building assumed damping equal to 7% of critical for the steel containment shell in accordance with the recommendations set forth in NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear

Power Plants" (1973). (Eggenberger, ff. Tr. 4784, at 25.) The SSI damping values were computed in a conservative manner compared with the Senior Seismic Review Team (SSRT) SEF recommendations. Dr. Eggenberger testified that in the analyses of the reactor building, damping values were conservatively reduced by 50% for all translational and rotational modes (Finding E-38), as contrasted with the reductions of 75% recommended by SSRT. (Eggenberger, ff. Tr. 4784, at 26; Eggenberger, Tr. 4808.)

D'Appolonia analyzed the soil-structure interaction using the half-space (lumped-parameter) method. (Findings E-37 through E-39.) During the hearing the Board inquired into the reasons for not following the SRP recommendation that the soil-structure interaction be analyzed using both the lumped-parameter and finite boundaries methods.

The Board received satisfactory answers. D'Appolonia used an acceptable method that complies with the intent of the guidance provided by the Standard Review Plan and is not required absolutely to use both methods. Generally, the SRP recommends use of both methods because the methods may produce different results at different frequencies. However, for the frequencies important to the analysis of the crane, between 2 to 3 Hz, the lumped-parameter and finite boundaries methods would produce similar results. (Cheng and Chokshi, Tr. 4929-33; Eggenberger, Tr. 4802-03.) Furthermore, Licensee performed several soil spring/soil structure analyses, verifying that altering soil springs did not have a significant effect on the results of the analyses. (Finding E-40.) Under the circumstances, D'Appolonia satisfied the intent of SRP § 3.7.2.4, without having utilized both methodologies.

In conclusion, the seismic analyses using the interim criterion and site-specific spectra comply with the recommendations set forth in NUREG/CR-0098 and the practices given in published Regulatory Guides and Standard Review Plans, particularly SRP § 3.7.2. As such, the floor response spectra provided to Licensee by D'Appolonia are the appropriate seismic input for the structural analyses of the overhead crane.

3. *Structural Analyses of Overhead Crane*

The structural adequacy of the crane, its crane rail support anchorages, and the steel structure which supports the crane when it is parked in the vicinity of the spent fuel pool must be assessed to evaluate the seismic stability of the crane. (Finding E-6.) Licensee initiated three analyses to evaluate these components. First, Whiting Corporation utilized the floor response spectra generated by D'Appolonia and the interim criterion to calculate the stresses induced by seismic loadings on the crane structure, and then compared the stresses with the crane structure materials' capacity to withstand stress. (Findings E-43 through E-49.) Second, Licensee utilized the maximum wheel loads calculated in the Whiting Corporation's analyses to assess the structural capabilities of the crane stops at the eastern end of the crane's runway and of the steel structure which supports the southern end of the overhead crane when it is operating in the vicinity of the spent fuel pool. (Findings E-6, E-68 through E-74.) Third, Structural Mechanics Associations, Inc. (SMA) performed a structural analysis of the crane rail anchorages using the floor response spectra associated with the interim criterion and site-specific spectra to scale down the maximum wheel loads reported in the Whiting Corporation's analyses. (Findings E-80 through E-84.)

Mr. Norman appeared before us and explained the methodology Whiting Corporation used to calculate the seismically induced stresses. The analyses performed by Whiting Corporation used the finite element method, which models the crane as an assemblage of many discrete beams. Mathematical expressions reflecting the structure's design and material properties are then formulated for each beam, and then solved using the ANSYS computer program to determine the forces and moments throughout the crane structure. From the forces and moments, stresses are calculated and compared to the crane materials' strength capacities. Components not included in the mathematical representation of the crane, such as bolts and welds, were analyzed independently of the ANSYS program using the moments and forces generated by the ANSYS analysis. (Findings E-47 through E-49.)

The crane materials' strength capacities are specified in the American Institute of Steel Construction ("AISC") Code, as modified by NRC SRP § 3.8.3, for assessing seismic loadings. (Finding E-51.) SRP § 3.8.3 increases the allowable stresses for steel materials above the loadings specified in the AISC Code because of the infrequent occurrence of seismic loadings. (Norman, Tr. 4804-10.)

Strength capacity for shear was calculated in accordance with Whiting Corporation's standard procedures because the AISC Code does not address irregularly shaped components and such phenomena as local buckling. (Finding E-53.) With the exception of bolts, Whiting Corporation's standard for allowable shear is more conservative than the AISC Guidelines. (Finding E-54.) The AISC Code, as modified by SRP § 3.8.3, occasionally permits allowable stresses to exceed yield. Whiting Corporation's standards never permit allowable stresses to exceed yield. (Norman, Tr. 4809-12.)

Whiting standards giving allowable shear capacities for bolts are slightly less conservative than under AISC Guidelines. However, Mr. Norman testified that Whiting Corporation uses the classical moment-of-inertia method to calculate the loads on bolts, which defines the loads more conservatively than the more modern total-area method. (Norman, ff. Tr. 4784, at 12.) Further, the allowable capacities for bolts calculated using Whiting Corporation's standards are well within the materials' yield strength in shear. (Finding E-55.)

The analyses performed by Whiting Corporation generally indicate that the maximum stresses caused by the seismically induced loadings associated with the interim criterion are generally low and well within established allowables. (Finding E-58.) All critical welds, plates and columns had margins of safety in excess of unity. (Finding E-62.) Structural members will not buckle locally (Finding E-63), and the crane will not become unstable and jump from its support rails. (Finding E-64.)

The Whiting analyses did point out two weaknesses in the crane structure, and Licensee has corrected these problems. The maximum stresses on the bolt connections between the knee brace and the crane bridge box girders exceeded the bolts' allowable strength. This problem was remedied by Licensee's replacement of the bolts with high-strength bolts whose allowables are not exceeded by the earthquake-induced stresses. (Finding E-60.) The Whiting analyses also established that the 5-ton monorail hoist attached to the crane's west bridge box girder needed to be strengthened to withstand the postulated seismic loadings. This modification has been completed. (Finding E-61.)

A third point of overstress was identified by the Whiting analyses. The calculations indicated that the maximum stress on the crane's gantry leg exceeds allowables by approximately 3%. The Board has been persuaded, however, that this slight overstress of the crane's gantry leg does not present a safety problem. Mr. Norman testified that the stress calculated to be in excess of allowable is localized, limited to one of the four corners of the cross-section of a single gantry leg, and does not exceed the materials' yield strength. (Finding E-59.) Mr. Norman further testified that the gantry leg would not fail until 50% of the total cross-section had reached a level of stress exceeding the materials' yield point. In response to Board questions, Mr. Norman explained that ductile materials, such as the steel used in the containment crane, retain the capability for withstanding stresses and carrying loads even after they have begun to deform. (Norman, Tr. 4891; Norman, ff. Tr. 4784, at 16-17.)

Most persuasive, however, is the fact that the overstress is created by the seismic loadings associated with the interim criterion. The floor response spectra associated with the site-specific spectra, which we have concluded is the appropriate seismic motion for the evaluation of the crane, indicate acceleration responses approximately 50% of the floor response spectra used in the Whiting analyses. (Finding E-26.) Use of the site-specific floor response spectra would reduce the stresses imposed on the crane by 50%, and would not overstress the gantry leg. (Chokshi, Tr. 4946.)

We find the Whiting Corporation's analyses of the crane structure both thorough and persuasive. At the hearing the Board examined Mr. Norman at length and found him to be not only an expert in his field, but a particularly informative and forthcoming witness.

One of this Board's major concerns throughout this hearing is the appropriateness of using current standards to judge the adequacy of equipment built years ago without quality assurance programs. The Board has been troubled by the possibility of attributing to structures characteristics, such as strength capacity, which may not accurately reflect their properties. Mr. Norman has resolved our concerns in this regard with respect to the crane.

While there may not have been quality assurance programs providing the detailed documentation currently required by 10 C.F.R. Part 50, Appendix B, Whiting Corporation, the manufacturer of the overhead crane, had quality control. Mr. Norman testified to the high quality of Whiting Corporation's personnel and workmanship. The methods, sizing and design of welds have not changed since the overhead crane was manufactured. The weld material used to manufacture cranes was standard A-7 material. The methods for manufacturing the bolts used to construct the crane have not changed. Further, Mr. Norman testified that he has participated in the re-rating of several of the cranes produced at the time of the Big Rock overhead crane and that the workmanship for all of them was high quality. Mr. Norman has also re-inspected the Big Rock Point overhead crane and attested to the excellent quality of its workmanship. (Norman, Tr. 3827-29.)

Mr. Chan testified concerning Licensee's evaluation of the structural capabilities of the runway crane stops and the steel support structure. These components were evaluated using the maximum wheel and crane stop loads generated by the Whiting Corporation analyses. These loads were combined with dead loads and the seismically induced motion of the condenser deck to which the steel support structure is attached. The loads were then statically applied to each structural member. These two components were also evaluated using the site-specific response spectra. (Findings E-68 through E-71, E-73.)

All moments, shear, and axial forces were combined for each individual member of the steel support structure and crane stops, and translated into stresses. The stresses were then compared to the allowable stresses specified in the applicable codes. (Finding E-72.) Licensee's evaluation demonstrates that the crane stops and steel support structure's allowable strength capacities exceed the stresses which would be induced by seismic loadings. (Finding E-74.) The one exception involved the overstressed bolted connection between the steel support's crane support girder and horizontal strut. The steel support structure has been modified to correct this problem. Messrs. Chan and Beachum testified that Licensee has welded a tee section between the horizontal strut and the support girder, which will alleviate the shear stress on the bolted connection. (Finding E-75.)

Licensee's initial analysis of the crane stops, which used the maximum stresses induced by loadings associated with the interim criterion, indicated that the northern crane stop tension anchor bolts would be overstressed by 44%. However, as with the gantry crane, this overstress is not a safety concern. When the stresses were calculated using the site-specific response spectra, the margin of safety for these bolts exceeded unity. (Findings E-76, E-77; Chokshi, Tr. 5008-09.)

SMA's analysis of the crane rail anchorages similarly provides confidence that the anchorages are capable of withstanding the seismic loadings associated with the SEP site-specific spectra. Mr. Campbell testified as to the methodology used to assess the anchorages. SMA scaled down the wheel loads reported in the Whiting Corporation's analyses by comparing the spectral accelerations of the floor response spectra associated with the interim criterion and site-specific spectra. This resulted in the wheel loads being scaled down by a factor of 2. (Findings E-79, E-80.)

The strength of the rail anchorages was analyzed using a simple linear elastic model of the rail, and the clips were modeled as rotational springs. The calculated stresses were compared with the allowables specified in the AISC Manual, as modified by SRP § 3.8.3. (Finding E-82.) The results of the SMA analysis show that all crane rail anchorages meet the AISC Code allowables. (Findings E-83, E-84.)[12]

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[12] Intervenor Christa-Maria proposed a finding in this area which stated that "no consideration was given to the effect of the impact from other objects to the crane, nor the results of debris hitting the SFP" (citation omitted). While Christa-Maria has cited page 6 to attachment 1 to Mr. Yanev's testimony (a schematic drawing of the plant) to sustain the first part of her proposed finding, the Board believes that a more proper cite is to Tr. 4886-87, where Christa-Maria questioned Licensee's witnesses concerning the jib crane. As the Board stated at the hearing (Tr. 4887), there is no basis for including the jib crane within this contention. Therefore, the Board cannot accept this finding. Additionally, there is no reason to consider debris hitting the pool, an allegation completely outside of this contention.

The Board received additional testimony from Mr. Yanev on the performance of cranes similar to the overhead crane under seismic loadings (peak accelerations) equal to or stronger than those associated with either the interim site criterion or the site-specific spectra. (Findings E-87 through E-103.) Mr. Yanev's figures indicated that none of the approximately thirty cranes surveyed by EQE were damaged while experiencing estimated peak ground accelerations of less than 0.35*g*. However, Dr. Reiter testified that caution should be observed in utilizing the specific ground motion estimates. Most of Mr. Yanev's estimates were based upon extrapolation techniques which have not been laid out in detail and for which the uncertainty has not been sufficiently emphasized. (Findings E-91, E-93, E-104.) These results included single-leg gantry cranes similar to the Big Rock Point overhead crane (Findings E-94, E-95.) Mr. Yanev's visual inspections and photographs showed that none of the cranes surveyed suffered buckling of the gantry legs. (Findings E-96, E-104.)

In one instance a damaged crane had a feature in common with the Big Rock Point crane. At Big Rock Point the overhead crane's rail support crosses an expansion joint. This design feature was one of the reasons crane rail anchorages at the Pleasant Valley Pumping Station were damaged during the Coalinga Earthquake in 1983. (Finding E-101.) But this does not draw into question the adequacy of the Big Rock Point crane rail anchorages. The Pleasant Valley Station suffered earthquake ground motion in excess of the motions postulated for Big Rock Point. (Finding E-101.) Also, the Pleasant Valley Station is a flexible steel building, and the expansion joint crossed by the crane rail moved about an inch during aftershocks following the 1983 earthquake and probably much more than that during the main quake. (Yanev, Tr. 3682, 3704.) At Big Rock Point, the crane is located on a large concrete structure; the movement across the expansion joint would be expected to be on the order of small fractions of an inch. (Yanev, Tr. 3704-05.) Finally, in spite of the damage to the crane rail anchorage at Pleasant Valley during the 1983 earthquake, the crane itself did not fail. (Finding E-101.)

During cross-examination, counsel for the NRC Staff questioned Mr. Yanev concerning the manner in which he extrapolated ground motion estimates at power plant sites from data recorded at neighboring locations. (Tr. 3692-3700.) Some of the power plants investigated by him had ground acceleration records from the vicinity of the plants; others require extrapolation, which Mr. Yanev performed using conventional techniques acceleration recordings during the earthquakes. (Yanev, Tr. 3643-95, 3699-3700.) In any case, the Board recognizes that extrapolating to estimate ground acceleration at some distance from the location of measurement results in a rather imprecise estimation. Mr. Yanev's investigation was not intended as a substitute for structural analyses. Because of the lack of precision in Mr. Yanev's data and analyses and the availability of appropriate engineering analyses, the Board has not relied on Mr. Yanev's investigation in reaching its conclusions.

D. Conclusion

After reviewing the analyses and their results, and the testimony presented during the evidentiary sessions, we conclude that the overhead crane is seismically safe. The record demonstrates that the overhead crane will not permanently deform or become unstable and that no affixed component will become dislodged under the seismic loadings associated with the SEP site-specific spectra.

The Board notes, however, that the structural analyses of the Big Rock Point crane assumed a maximum operating load over the spent fuel pool of 24 tons. Licensee owns a 60-ton cask, which Licensee has committed not to use until certain commitments made to the NRC Staff are satisfied. Mr. Norman explicitly testified that the structural analyses performed by Whiting Corporation cannot be used to seismically qualify the crane operating with the 60-ton cask. (Norman, Tr. 4826.) Accordingly, the Board expects that the NRC Staff will review this matter prior to any use in the future of the 60-ton cask over the spent fuel pool.

The Board concludes, therefore, that the Big Rock Point overhead crane is seismically safe, and that there is reasonable assurance that the crane can be operated without endangering the health and safety of the public, consistent with 10 C.F.R. § 50.57(a)(3)(i).

VI. CHRISTA-MARIA CONTENTION 2 AND O'NEILL CONTENTION II.A — SOUTH WALL

A. Background

These contentions originally raised concerns about radiation risks to the general public and to workers at the Big Rock Point Plant resulting from radiation shine through the south wall of the spent fuel pool where the pool wall is thinnest. (For a statement of the contentions, *see* Finding J-1.) In our Memorandum and Order (Concerning Motions for Summary Disposition), dated February 19, 1982, LBP-82-8, *supra*, 15 NRC at 321-22, we found that the contentions and the evidence presented in support of and in opposition to motions for summary disposition raised eight genuine issues of fact, which, with two exceptions,[13] essentially narrowed the contentions to concerns regarding only occupational radiation exposure of workers. (Finding J-2.)

Licensee's witnesses and the NRC Staff witness appeared as a panel. Testifying for Licensee were: Mr. Roger Sinderman, Director of Licensee's Radiological Services Department; Mr. Charles Axtell, the former Plant Health Physicist at the Big Rock Point Plant (now at Licensee's Midland Plant); and Mr. Edward Benz, an engineer with NUS Corporation. Mr. Seymour Block, an NRC Senior Health Physicist, testified for the NRC Staff. With the exception of one exhibit, Intervenors withdrew their testimony on this contention. (Finding J-3.)

B. Applicable Law

Six of the genuine issues of fact that survived the motions for summary disposition related to the Licensee's radiation protection program for workers who would modify the proposed spent fuel pool. Although each issue was narrowly drawn, concern was whether the installation of the three additional spent fuel racks in the Big Rock Point spent fuel pool could be performed in compliance with the "as low as reasonably achievable" ("ALARA") standard articulated in NRC regulations.

Section 20.1(c) of 10 C.F.R. provides that licensees should "make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable." Further, that section explains that

> [t]he term "as low as is reasonably achievable" means as low as is 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.

Additional guidance in the application of the ALARA standard is found in Regulatory Guide 8.8. With the ALARA standard in mind, we turn now to a discussion of the eight issues.

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[13] Genuine issues of fact (4) and (5) raised limited questions concerning the Licensee's and NRC Staff's calculations of the offsite dose attributable to radiation emanating through the south wall of the pool. These calculations were submitted in support of the motions for summary disposition.

C. Discussion

1. South Wall Radiation Dose Calculations

The first five genuine issues of fact concerned the calculations presented by the Licensee and the NRC staff regarding the shielding capability of the south wall of the spent fuel pool. The Licensee performed such an evaluation to assure that any radiation exposure to workers and the public were within the limits established by 10 C.F.R. Part 20 and were ALARA. Some apparent discrepancies needed clarification or correction before we could accept the evidence presented by the Licensee and the NRC Staff.

> (1) What caused the discrepancy between staff and applicant statements about the relevant dimensions of the south wall of the spent fuel pool and what effect, if any, has this discrepancy had on radiation calculations?

On motion for summary disposition, Licensee stated that the thickness of the south wall ranged from 5 feet 9 inches to 3 feet 6 inches. The NRC staff witness, on the other hand, stated that the thickness ranged from 6 feet to 3 feet. Staff witness Block explained that the 3-foot dimension suggested by the NRC Staff was in error. Mr. Block acknowledged that Licensee was correct in stating that the thinnest section of the south wall is 3 feet 6 inches thick and that, therefore, Licensee's calculations of the dose rate at the south wall based on that thickness are correct. (Finding J-4.)

The second issue asked:

> (2) What is the combined radiation from the pool and filter sock tank?

In his affidavit in support of Licensee's motion for summary disposition, Mr. Axtell testified that the radiation dose of about 2 mrem/hr emanating from the south wall was small compared to the dose of 30-40 mrem/hr from the spent fuel pool sock tank which is located near the south wall. Intervenors on the other hand believed that the combined dose rate would amount to 68-78 mrem/hr. As a consequence, we heard evidence to reconcile this disagreement.

However, as Mr. Axtell explained in his affidavit in support of Licensee's motion for summary disposition and in his testimony at the hearings, Licensee will not store spent fuel adjacent to the thinnest section of the south wall. There is currently a channel rack at that location which Licensee intends to keep there after the pool modification. The channel rack cannot be used for the storage of fuel assemblies. The thinnest section of the south wall at which spent fuel will be stored is 4 feet

5 inches thick. Only spent fuel with a decay time of at least 1 year will be stored in the outer three rows of the proposed rack adjacent to that location on the south wall. (Finding J-5.)

Mr. Benz calculated the radiation dose emanating from 1-year-old spent fuel through the 4-foot 5-inch section of the south wall to be 2.7 mrem/hr. Mr. Axtell stated that the radiation dose at the radiologically controlled area of the filter sock tank is generally 30-40 mrem/hr. When combined with the radiation dose from the south wall, the radiation dose near the south wall is between 32.7 and 42.7 mrem/hr. (Finding J-5.)

Intervenors offered no evidence on this issue.

We accept Licensee's representations concerning the configuration of stored fuel and, on that basis, are satisfied that radiation levels are ALARA. However, Licensee's commitment depends on satisfactory implementation of its procedures. In view of that circumstance, we conclude that a procedure also should be implemented to investigate the fuel configuration in the pool should radiation exceed expected levels. Hence, we require that CPC implement a procedure requiring a prompt investigation and adequate resolution of the problem whenever radiation level in the pool sock tank area is detected to be 50 mrem/hr or more.

The third genuine issue of fact asked:

> (3) What point on the south wall was used as a reference point for calculating dose estimates?

This question was raised to clarify which point along the south wall was used by the Licensee for calculating the radiation dose estimates. Mr. Benz explained that he calculated the dose based on a wall thickness of 4 feet 5 inches, the actual location at which Licensee intends to store spent fuel along the south wall. (Finding J-6.)

The fourth genuine issue of fact asked:

> (4) What is the reason that applicant stated that it used "mass absorption coefficients" in radiation estimates when it apparently used linear absorption coefficients?

This question concerns a dose calculation presented by Mr. Sinderman in his affidavit in support of Licensee's motion for summary disposition regarding the calculated radiation dose level at the site boundary. In the affidavit, he stated that "mass absorption coefficients" were used in the calculations when it appeared he was actually using linear absorption coefficients. Mr. Sinderman confirmed, as we suspected, that this was a misstatement and that indeed linear absorption coefficients were used. (Finding J-7.)

The fifth genuine issue of fact asked:

> (5) What was the location and reference level to which staff applied the inverse square rule to calculate offsite doses?

This question concerned the NRC Staff's analysis of the radiation dose to the public emanating from the south wall of the spent fuel pool. Referring to page 9 of the dose calculation presented in the affidavit of Mr. William Bell filed in support of Licensee's motion for summary disposition, Mr. Block explained that the NRC Staff used the distance from coordinate 0.0.0 to the center of fuel assembly which is equal to about 4.1 feet. This distance plus 2900 feet to the site boundary were used to determine offsite dose by the inverse square rule. (Finding J-8.)

Based on the evidence presented by the Licensee and the NRC Staff, we can now accept their calculations of the radiation dose emanating from the south wall as reliable. We conclude that this dose poses no undue risk to the public or workers at the Big Rock Point Plant. Further, we believe Licensee's commitment to store only fuel with 1 year's decay time along the south wall and to not store spent fuel at the thinnest section of the south wall is in keeping with the ALARA principle inasmuch as it will minimize the radiation dose emanating from the south wall of the spent fuel pool.

2. *ALARA Concerns During the Pool Modification*

Genuine issues of fact 6 and 7 ask:

> (6) What hiring, training and supervision methods and what health physics safeguards will be used during the installation of the new fuel rack?
>
> (7) What has applicant done to correct alleged health physics deficiencies identified by the Institute of Nuclear Power Operations in its August 1981 report?

These questions were posed in response to a concern that inexperienced temporary workers might be employed to carry out the spent fuel pool modification and in response to criticisms of certain health physics practices at the Big Rock Point Plant made by the Institute of Nuclear Power Operations in their 1981 report ("the 1981 INPO report"). Moreover, we are concerned whether Licensee's ALARA program is adequate to assure that occupational radiation exposure for workers assigned to the spent fuel pool modification will be kept as low as is reasonably achievable. (*See* Memorandum and Order, dated February 19, 1982, LBP-82-8, *supra*, 15 NRC at 320-21.)

In his affidavit filed in support of Licensee's motion for summary disposition, Mr. Axtell outlined the steps involved in the spent fuel pool reracking operation. He also explained the measures which will be taken to reduce the radiation dose in the area of the spent fuel pool. (Finding J-9.) Mr. Axtell estimated the total man-rem dose for the spent fuel pool modification to be about 18.2 man-rem. (Finding J-10.) Mr. Axtell expressed confidence in this estimate because it was based on radiation exposures associated with similar racking operations previously conducted at the Big Rock Point Plant. (Axtell, ff. Tr. 5025, at 5-6.) It would appear that a total dose of 18.2 man-rem, which is about 6% of the average annual man-rem exposure of 290 man-rem for the past 5 years at Big Rock Point, indicates a well-structured ALARA program for the pool expansion job. (Finding J-11.) We will examine, however, Licensee's training and management practices to ascertain whether reasonable assurance exists to expect that this ALARA goal will be achieved.

(a) Hiring

Although Licensee has not ruled out the use of temporary workers, it is almost certain that the pool modification will be accomplished with the staff presently employed at the Big Rock Point Plant. In any event, if contractor personnel are used to assist in the pool modification, the number used would be small. (Finding J-12.) Moreover, as Mr. Axtell stressed, the hiring, training and supervising of contractor personnel have been common practice throughout the 21-year history of the Big Rock Point Plant. (Finding J-13.)

(b) Training

All temporary workers receive a basic 6-hour course covering radiation protection, respiratory protection, plant security, industrial safety, and other topics. This course is followed by a test to assure comprehension. Any temporary workers who may be employed to assist in the spent fuel pool modification will be required to take this course. (Finding J-14.) Workers who may use respiratory protection masks will receive training in their fitting. (Finding J-15.)

Plant personnel have already received more detailed training in radiation protection. All Maintenance Department employees have received approximately 40 hours of thorough training in radiation protection and receive additional training at monthly safety meetings. (Finding J-16.) Chemistry and Radiation Protection Technicians, who will provide radiation protection and monitoring during the pool modification, receive extensive training in numerous radiation topics, including ALARA principles, in a program consisting of a 12-week Basic Course and an 11-week Advanced Course. Technicians also receive training under a "Practical Factors" program which assures their practical ability to perform radiation protection tasks. Those technicians involved with the pool modification will have completed the practical factor sheets pertaining to radiation protection work associated with the fuel pool modification. (Finding J-17.)

(c) Supervision

Mr. Axtell detailed the procedures that will be followed during the pool modification. (Finding J-18.) These procedures are designed to assure that work is carried out efficiently, thus minimizing personnel radiation exposure. Experienced personnel will supervise the activities carried out under those procedures. (Finding J-18.) The procedures also emphasize radiation protection practices such as the wearing of proper protective clothing. (Finding J-19.) Qualified Health Physics Technicians will supervise the radiological protection aspects of the pool modification. They will perform continuous radiological surveillance of the work area according to well-established and approved procedures. (Finding J-20.)

In addition to the inquiry about the hiring, training and supervision of workers during the pool modification, we also asked what health physics safeguards would be used during the installation of the racks. We find this concern to be closely related to an overall concern regarding the adequacy of Licensee's ALARA program in view of the criticisms stated in the 1981 INPO report. Specifically, the 1981 INPO report found that no comprehensive ALARA program existed at the Big Rock Point Plant.

Mr. Sinderman offered his views on Licensee's ALARA Program in response to our specific interest in his opinion as Licensee's Director of Radiological Service (Memorandum and Order, dated February 19, 1982, LBP-82-8, *supra,* 15 NRC at 321.) Mr. Sinderman explained that Licensee had come to a similar conclusion regarding the need for a company-wide comprehensive ALARA program. Consequently Licensee developed a Corporate Radiation Safety Plan which includes the ALARA practice to be followed at Licensee's nuclear facilities. The position of ALARA coordinator was created at each of Licensee's nuclear power plants, including the Big Rock Point Plant. The program requires pre-job planning, worker training, and post-job reviews of all high-exposure jobs. Licensee's Corporate Radiation Safety Plan also requires an in-depth surveillance of selected areas where it is believed that improvement can be made. (Finding J-22.)

The expansion of the spent fuel pool capacity will be conducted under Licensee's Comprehensive ALARA program. Moreover, with respect to the pool modification itself, Mr. Sinderman has reviewed the steps being taken to reduce exposure and he concludes that the resulting exposure will be ALARA. (Finding J-23.)

Having now evaluated Licensee's plan for hiring, training and supervising workers, including any temporary workers, who may assist in the pool modification, and having been assured that Licensee has developed a good ALARA program and that it is otherwise responding to criticisms made by INPO regarding health physics practices, we believe that Licensee has taken adequate measures to successfully meet its ALARA goals in the spent fuel pool modification.

3. *Radwaste Demineralizer*

Our eighth genuine issue of fact asked:

> (8) To what extent will the radwaste demineralizer be employed on a continuing basis to attenuate radiation from the spent fuel pool?

In his affidavit filed in support of Licensee's motion for summary disposition, Mr. Axtell stated that the radwaste demineralizer would be used to recycle fuel pool water prior to the modification but that the radwaste demineralizer would not be used to recycle pool water on a continual basis. Consequently we questioned whether Licensee's use of the radwaste demineralizer is in keeping with the ALARA standard with respect to minimizing radiation levels over the spent fuel pool surface. LBP-82-8, *supra*, 15 NRC at 321.

The radwaste demineralizer is used to recycle the spent fuel pool for several weeks prior to a shutdown and before personnel spend any significant amount of time near the spent fuel pool surface. (Finding J-25.) The effect of this recycling is to reduce the radiation dose levels over the pool from approximately 25-30 mrem/hr to 12 mrem/hr. (Finding J-26.) The water in the spent fuel pool is seldom recycled through the radwaste demineralizer at other times because very few man-hours are spent in this work area during normal operation.

During normal plant operations, the radwaste demineralizer is used to process other plant water streams. (Finding J-25.) This activity, however, does not monopolize the use of the demineralizer, and it could be used more often to recycle water from the spent fuel pool. But as Mr. Sinderman explained, the changeover from one use to the other involves a manual valving operation in a relatively high radiation area. Thus, any dose savings from recycling the spent fuel pool water would probably be more than offset by the occupational exposure incurred during the valving operation. (Finding J-27.)

Based on the foregoing, we believe that Licensee's proposed use of the radwaste demineralizer is in keeping with the ALARA principle.

D. Conclusion

Based on the uncontroverted evidence, we conclude that Licensee is taking adequate and reasonable steps to assure that the spent fuel pool modification will be performed in keeping with the ALARA standard. Accordingly, Christa-Maria Contention 2 and O'Neill Contention II.A are dismissed, with the single exception that Licensee shall implement a procedure to monitor radiation in the pool sock tank area and take appropriate action whenever the radiation level in that area equals or exceeds 50 mrem/hr.

VII. CHRISTA-MARIA SUBCONTENTION 9(1) — SIZE OF THE EPZ

A. Background

Christa-Maria subcontention 9(1), states:

> The increased inventory of the fuel pool requires that the emergency plan be based on an inhalation pathway of 10 miles rather than 5 miles and on a 50-mile rather than a 30-mile ingestion pathway.

The Board admitted this contention based on its finding that Intervenors had made plausible arguments concerning both the presence of an increased inventory of radioactive products and the mechanisms for dispersal. LBP-82-32, 15 NRC 874, 881 (1982). In addition, the Board expressed concern that it was unable to determine whether a specific analysis of the appropriate size of the emergency planning zones ("EPZ") for Big Rock Point had been performed by the NRC Staff. The Board found such an analysis particularly necessary because of the use of reduced planning zones. *Id.* at 881.

To address this contention, Licensee submitted the testimony of Roger W. Sinderman, who is employed by Consumers Power Company as director of radiological services. (Finding G-2.) The NRC Staff presented the testimony of Monte Phillips, an emergency preparedness analyst and section chief. (Finding G-3.) Intervenors presented no testimony on this contention, relying instead on cross-examination of Licensee and Staff witnesses.

B. Applicable Law

Section 50.47(c)(2) of 10 C.F.R. provides that, generally, the plume exposure pathway EPZ for nuclear power plants shall be about 10 miles in radius and the ingestion pathway EPZ about 50 miles in radius. Essentially, the outer radius of the 10-mile zone is based upon the substantial reduction in early severe health effects (injuries or deaths) from whole-body doses at distances greater than 10 miles from the worst postulated accidents at large reactors. The outer radius of the 50-mile zone is based on the minimal potential for significant contamination of food supplies from similar accidents. (NUREG-0396 (EPA 521/1-78-016), "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," December 1978, at 15-17, Appendix I; NUREG-0654, at 12-13; *see* Phillips, ff. Tr. 2859, at 3-4.) Section 50.47(c)(2) provides that the size of the EPZs may be determined on a case-by-case basis for reactors with an authorized power level of less than 250 megawatt thermal (MWT).

NUREG-0654 provides at page 11 that small water-cooled reactors (less than 250 MWT) may use a reduced plume exposure pathway EPZ of about 5 miles in radius and a reduced ingestion pathway EPZ of about 30 miles in radius. This conclusion is based on the lower potential hazard from these facilities, because they have a lower radionuclide inventory and require longer periods of time to release significant amounts of radiation in many accident scenarios.

NUREG-0654, at pages 6-7, further indicates that no single accident or set of accidents serves as a design basis for emergency planning. Thus, we must determine the appropriate size of the EPZ by considering the possibility that the entire gaseous, halogen and other semi-volatile radionuclide inventory of the reactor might be released.

C. Discussion

The particular issue raised here by Intervenors is whether the amendment being sought, which would increase the radioactive inventory of the spent fuel pool, necessitates an increase in the size of the Big Rock Point EPZs, which are now 5 and 30 miles. As a baseline consideration, however, we must first determine whether 5- and 30-mile EPZs are sufficient for Big Rock Point in the absence of an expanded pool. If such is the case, we can then determine whether the proposed expansion requires that the size of the EPZs be increased.

On April 24, 1980, Licensee submitted an evaluation to the NRC Staff justifying a 5-mile plume exposure pathway EPZ for Big Rock Point. (Finding G-7.) The results of this evaluation show that the whole-body dose at the 5-mile EPZ boundary would be 34 rem to a person at that location for the duration of the radiological release. This dose is well below the health-threatening dose of 100 rem used for emergency planning purposes. (Finding G-8.) Moreover, the projected dose rate at the 5-mile boundary during average meteorological conditions would be approximately 100 times less. (Finding G-9.)

NRC personnel from Region III reviewed Licensee's evaluation and determined that the methodology used was appropriately conservative. Accordingly, on June 13, 1980, the Staff informed Licensee that 5- and 30-mile EPZs are appropriate for the Big Rock Point Plant. (Finding G-10.) We find this evaluation to be acceptable.

Intervenors attempted to establish through cross-examination that the fission product inventory used by Licensee in its evaluation was inadequate because plutonium was not included. (Tr. 2759-62, 2783-90, 2814.) However, Licensee need not consider the plutonium because it is an oxidized and nonvolatile component of the pool that would not be released during an accident. (Finding G-11; Phillips, ff. Tr. 2859, at 6-7.)

We now consider whether the proposed increase in the pool inventory at Big Rock Point warrants any increase in the size of the EPZs. First, we note that Licensee's evidence shows that only about 7% of the total fission product inventory at Big Rock Point is provided by the stored spent fuel, and less than one-fifth of this 7% is attributable to Licensee's proposal to expand the capacity of the spent fuel pool. (Finding G-12.) Second, using the meteorological dispersion data provided in Regulatory Guide 1.3, and taking into account the much smaller source term at Big Rock Point as compared to a typical large reactor (3800 MWT), Licensee has determined, without contradiction, that the dose rates received at 1.4 and 7 miles from Big Rock Point would be no more than the dose rates that would be received at 10 and 50 miles from a large reactor under identical accident and meteorological conditions. This determination by Licensee takes into account the proposed expanded inventory of the spent fuel pool. (Finding G-13.) Assuming that EPZs of 10 and 50 miles are sufficient for a large reactor, as 10 C.F.R. § 50.47(c)(2) expressly provides, it necessarily follows that the 5- and 30-mile EPZs are highly conservative for Big Rock Point even considering an expanded pool inventory.

During the hearing both Intervenors and the Board asked whether a criticality accident in the spent fuel pool might increase the fission product inventory enough to affect the size of the 5-mile EPZ. In response, Licensee presented supplemental testimony by Roger Sinderman and Frank Turski. (ff. Tr. 4346.) Their testimony postulated criticality occurring in the spent fuel pool at the maximum power and resultant temperature levels that can occur. They assumed that this condition, which would automatically stop when the pool boiled dry, would nevertheless continue for 3 years. At that time an equilibrium concentration of radionuclides similar to that of the reactor core would be achieved in the spent fuel pool. (Finding G-14.)

Mr. Sinderman and Mr. Turski calculated that the total fission product inventory available for release from the plant even after the pool inventory achieved an equilibrium concentration of radionuclides would increase by only 1.6%; therefore the calculated plume exposure EPZ radius would be increased by only 64 meters, from 1.4 miles to 1.44 miles. (Finding G-14.) This testimony confirms Mr. Phillips' earlier judgment that a criticality accident would not significantly add to the fission product inventory in the spent fuel pool. (Phillips, Tr. 2981.) For these reasons, both Licensee and the Staff have concluded that, even assuming the occurrence of the incredible criticality scenario postulated for purposes of analysis, the 5- and 30-mile EPZs remain more than adequate for Big Rock Point. (Finding G-14.) We agree.

Finally, the Board expressed concern over the possibility that certain rain or snow conditions could cause a substantial fraction of the radioactive material leaving the reactor to be deposited in localized areas near the plant. (Tr. 2824-25.) Such localized concentrations are known as "hot spots." (Finding G-15.) The Board asked how far from the plant site weather conditions could possibly produce a hot spot that would cause one or more fatalities. (Tr. 2826.)

Mr. Sinderman testified that many variables would contribute to such a situation, including the rate of snowfall or rainfall, wind speed, and other meteorological conditions. (Tr. 3201.) Mr. Sinderman therefore postulated an extreme scenario in which all of the semi-volatile particulates and halogens from the reactor core are smeared over a 22.5° sector originating at the site and extending out 3 miles Under such conditions, a person standing in the middle of the sector would receive a dose rate of only 5.5 rem per hour. (Finding G-16.)

Mr. Phillips testified that the likelihood of rain or snow conditions that might cause hot spots occurring concurrently with core melt and containment failure is extremely low. (Finding G-17.) He also stated that hot spots are disregarded when determining the size of the EPZs, but they are relevant to protective action and can be taken into account during an emergency on an *ad hoc* basis. (Finding G-18.)

For these reasons, the Board finds that hot spots need not be considered in determining the sizes of the EPZs at Big Rock Point.

D. Conclusion

We conclude that the 5- and 30-mile EPZs for Big Rock Point meet the requirements of the Commission's emergency planning regulations and guidance. Furthermore, the Board concludes that the additional fission product inventory resulting from the proposed expansion of the capacity of the Big Rock spent fuel pool does not warrant any increase in the size of the EPZs.

VIII. CHRISTA-MARIA SUBCONTENTION 9(6) — RADIATION MONITORING

A. Background

Christa-Maria Contention 9, subpart (6) states:

> Applicant should comply with regulations requiring adequate radiation monitoring.

This subcontention was admitted by the Licensing Board in its "Memorandum and Order (Motion to Strike Emergency Planning Contention)," LBP-82-32, *supra*. The Board admitted the issue on the basis of an allegation by Intervenors that Licensee was not complying with certain emergency planning requirements promulgated in response to the TMI-2 accident. In particular, the Intervenors alleged that installation of monitoring equipment had been continually deferred by the Licensee or was being reduced.

Licensee filed the testimony of three witnesses on this subcontention. Charles E. Axtell, the Plant Health Physicist at Big Rock Point for 14 years, described three types of radiation monitoring in use at Big Rock Point: effluent monitoring, in-plant iodine monitoring, and containment radiation monitoring. (Finding H-2.) Robert M. Marusich, a staff engineer in Licensee's Radiological Services Department, described Licensee's ability to promptly assess the degree of core damage following an accident. (Finding H-3.) Donald L. Swem, a general engineer at Big Rock Point, addressed the power sources and the calibration of the high-range containment radiation monitors in use at Big Rock Point. (Finding H-4.)

The Staff submitted the testimony of Monte Phillips, an NRC Region III emergency preparedness analyst, who evaluated Licensee's radiation monitoring systems in the context of emergency planning. (Finding H-5.) Licensee filed a motion to strike parts of Mr. Phillips' testimony on the ground that it addressed aspects of radiation monitoring and emergency planning outside the scope of the subcontention. On the basis of this motion, the Staff withdrew its tender of those portions of Mr. Phillips' testimony and the attached Safety Evaluation Report dealing with emergency planning generally. (Tr. 2846.) The Board, however, ruled that all radiation monitoring devices discussed in the Staff's testimony were relevant to the subcontention. (Tr. 2851, 2853.)

Intervenors presented no testimony on this subcontention, relying instead on cross-examination of Licensee and Staff witnesses.

B. Applicable Law

Section 50.47(b)(9) of 10 C.F.R. states that emergency plans must, among other things, provide adequate methods, systems and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition. Guidance as to the meaning of this regulation is provided in NUREG-0654, Item II.I.2, which states:

> Onsite capability and resources to provide initial values and continuing assessment throughout the course of an accident shall include post-accident sampling capability, radiation and effluent monitors, in-plant iodine instrumentation, and containment radiation monitoring in accordance with NUREG-0578, as elaborated in the NRC letter to all power reactor licensees dated October 30, 1979.

The performance details of the monitoring systems recommended in NUREG-0654 were contained in NUREG-0578. These details were carried forward to NUREG-0737 at the time it superseded NUREG-0578. Guidance concerning post-accident sampling capability is contained in Item II.B.3 of NUREG-0737. Recommendations concerning effluent monitoring, in-plant iodine monitoring and containment radiation monitoring are contained in Item II.F.1 of NUREG-0737.

Item II.F.1, Attachment 1, of NUREG-0737 recommends installation of wide-range noble gas effluent monitors designed to function during accident conditions as well as normal operating conditions. Attachment 2 of Item II.F.1 recommends continuous sampling of plant atmosphere for releases of radioactive iodines and particulates, designed to minimize noble gas interference. Attachment 3 of Item II.F.1 recommends installation of at least two high-range containment radiation monitors built to function in an accident environment.

With respect to post-accident sampling, NUREG-0737, Item II.B.3, recommends that the licensee determine whether it is possible to sample the reactor coolant and containment atmosphere promptly and to perform a radiological spectrum analysis promptly to quantify certain radionuclides indicating the degree of core damage. If such analyses are not possible with existing equipment, NUREG-0737 recommends design modifications or additional equipment procurement.

Guidance with respect to other radiation monitoring equipment at Big Rock Point, which the Board held was relevant to the monitoring contention, is found in NUREG-0654, subpart b of Criterion II.H.5. This Criterion states:

> 5. Each licensee shall identify and establish onsite monitoring systems that are to be used to initiate emergency measures in accordance with Appendix 1, as well as those to be used for conducting assessment.
>
> The equipment shall include:
>
> b. radiological monitors, (e.g., process, area, emergency, effluent, wound and portable monitors and sampling equipment); . . .

It is against the above Commission regulation and Staff guidelines that the Licensee's evidence on the monitoring issues should be weighed.

C. Discussion

Licensee's prepared testimony focused on the kinds of monitoring recommended in NUREG-0737: (1) noble gas effluent monitoring, (2) in-plant iodine monitoring, (3) containment radiation monitoring, and (4) Licensee's ability to assess the degree of core damage, if any, throughout the course of an accident. The Staff's presentation discussed these types of monitoring as well as other radiation monitors in use at the Big Rock Point Plant. Following the Board's ruling that all radiation monitors at Big Rock Point were relevant to the contention, Licensee presented additional testimony by Mr. Axtell regarding these other monitors.

Mr. Phillips testified that radiological monitors are placed throughout the plant to provide both local and control room annunciation and readouts. He explained that the effluent process monitoring system measures gross radioactivity levels of all airborne and liquid effluents released from the plant via the liquid and gaseous radwaste systems and the plant ventilation systems. (Finding H-6.) Mr. Axtell described a wide range of radiation monitors in use at Big Rock Point, including process monitors, area monitors, emergency effluent monitors, wound monitors, portable monitors and sampling equipment. Mr. Axtell stated that these monitoring systems comply with the recommendations contained in subpart b of Criterion II.H.5 of NUREG-0654. (Finding H-7.) Mr. Phillips agreed, and concluded that the Licensee has the methods, equipment and expertise to rapidly assess the actual or potential magnitudes and locations of radiological hazards through liquid or gaseous release pathways. (Phillips, ff. Tr. 2859, at 10.)

The specific types of monitoring listed in Item II.I.2 of NUREG-0654 will now be separately examined.

1. Noble Gas Effluent Monitors

The noble gas effluent monitors at Big Rock Point measure all radioactivity released from the plant's gas stack during operation or shutdown, either in the form of noble gases, iodines or particulates. (Finding H-8.) Under accident conditions, effluent monitors would enable Licensee to measure the radiation dose to the public at the site boundary. (Finding H-9.) When NUREG-0578 was issued, recommending installation at all plants of effluent monitors of greater range and reliability than those in use, monitors having all the recommended features were not yet commercially available. The Commission therefore required installation of interim high-range effluent monitors. In January 1980, such an interim monitor was installed at Big Rock Point and was approved by the NRC Staff. (Findings H-12, H-13, H-14.)

NUREG-0737, published in November 1980, recommended that permanent wide-range monitors be installed by January 1, 1982. Because certain equipment was unavailable, however, the installation date at Big Rock Point had to be deferred several times. (Finding H-15.) The permanent system was finally delivered in the Fall of 1982. Even then spare parts were not yet available, so that the system could not be quickly restored to operation if it broke down and required repair. This caused a further postponement of the in-service date. (Finding H-16.) The Staff approved this further postponement on condition that the interim high-range effluent monitor remain in operation until the permanent wide-range monitoring system is placed in service. (Finding H-17.) On February 16, 1983, Licensee committed to placing the permanent effluent monitoring system in service by December 31, 1983. This commitment was confirmed by Commission Order, dated March 14, 1983 (unpublished). (Finding H-18.) Mr. Axtell testified that he expected spare parts to become available for the new monitoring system beginning in late November or early December of 1983. (Finding H-19.)

The permanent wide-range effluent monitoring system consists of two individual monitors: a low-range monitor to replace the effluent monitor in use since 1962, and a high-range monitor to replace the interim high-range monitor. (Finding H-20.) Mr. Phillips testified that the Staff would inspect the monitoring equipment to determine that it has been installed and calibrated correctly. (Finding H-21.)

Subsequent to the closing of the record in this proceeding, the Board received a letter from Joseph Gallo, counsel for Licensee, dated February 27, 1984, advising that the permanent effluent monitoring system had been installed and made operational on December 23, 1984. The letter also disclosed that the new stack gas monitoring system fails to comply with the guidance provided by NUREG-0737, Table II.F.1-1, as regards to display criteria because it does not automatically provide direct radiation measurements. Licensee indicated that it had requested the NRC Staff to approve an alternate approach to satisfy the criteria.

In a memorandum to this Licensing Board and to the Appeal Board, dated March 14, 1984, the NRC Staff reported that it had reviewed the Licensee's request for a variance and had concluded the proposed variance was unacceptable. Specifically, the permanent effluent monitor displays in units of counts per minute from the beta scintillation detector and mR/hr from the ionization chamber. Item II.F.1-1 stipulates that the monitor's display should be in units of equivalent Xe-133 concentrations or microcuries per cubic centimeter (μCi/cc) of actual noble gases. Staff believes that having to convert to the appropriate units by reading from graphs, as Licensee proposes to do, could result in errors — especially under the pressure of accident conditions — and would require operators to divert their attention from controlling the plant during an accident. Moreover, microprocessors that will give a direct readout from the stack monitors in μCi/cc are available and are being used by other licensees. Therefore, Staff requested Consumer's Power Company to modify the Big Rock Point Plant's stack gas monitoring system to bring it into compliance with Item II.F.1-1. Finally, the Staff's March 7, 1984, letter concluded by advising Licensee it could direct comments on burden and duplication to the Office of Management and Budget. *See* letter, dated March 7, 1984, from Dennis M. Crutchfield, Chief of NRC's Operating Reactors Branch #5, to David J. VandeWalle of Consumers Power Company.

On April 3, 1984, Intervenors filed a motion to reopen the hearing record with respect to this contention, because of the foregoing facts. Pursuant to stipulation among all parties, the record was reopened to admit further evidence relating to the effluent monitors.

Licensee committed to modify the wide-range effluent monitors at Big Rock Point by May 31, 1984, to provide readout capability in curies per second (Ci/sec) of actual noble gases released. (Letter of R.M. Krich to Dennis M. Crutchfield, dated April 6, 1984.) The modification would be achieved by the addition of scales to the meter faces; the new scales would have units of Ci/sec. (*Id.*) The NRC Staff concluded that with this modification the Big Rock Point wide-range effluent monitors would satisfy the guidance of NUREG-0737. (Letter of Dennis M. Crutchfield to David J. VandeWalle, dated April 20, 1984.)

The Intervenors raised four principal points in their proposed findings on the modified stack gas monitors. First, they argue that the monitors are not in compliance with single-mode failure requirements. To put the importance of the noble gas stack monitor in proper perspective, one should realize that the monitor is part of the NRC's insistence on defense in depth. The stack monitor is not relied upon to measure core degradation. Other systems perform that function. Instead, the stack monitor is designed to assist plant management to make prompt, appropriate recommendations about emergency actions to protect the public. Even if the monitor failed, however, releases could still be detected by offsite monitoring with portable survey instruments. (Emch, Dep. Tr. 21-22, 42; Beer, Dep. Tr. 38.) Furthermore, the Licensee has committed to implement alternative procedures of getting the reading should the monitor become inoperable for some specified period of time. (Emch, Dep. Tr. 43.) Thus, there is no need for redundancy in the stack gas monitor; we shall therefore apply the guidance set forth in NUREG-0737, § II.F.1, making redundancy unnecessary.

Second, Intervenors contend that the readings from the "paste-on scales" on the monitor faces may yield inappropriate information. Staff maintains, however, that the add-on scale, and the operator effort necessary to estimate offsite doses from the monitor readings, is equivalent to or better than the display scales called for in NUREG-0737. The add-on scales will require the same kind of calculation that would be necessary using the NUREG-0737 recommended scales, and the assumptions required to use the add-on scale are the same kind of assumptions required if the display specified in NUREG-0737 is used. (Emch, Dep. Tr. 67,068, 67,080.)

Third, Intervenors argue that CPC has not completed the human engineering requirements of NUREG-0737. Staff, in NRC Staff Response to Revised Proposed Findings of Fact and Conclusions of Law Regarding the Monitoring Subcontention, dated July 5, 1984, stated, "Licensee's witness Krich indicated that the human factors considerations stated in NUREG-0737, § II.F.1, were included in the design and installation of the high range noble gas effluent monitor." (Krich, Dep. Tr. 135, 148.) It is not clear from a reading of the entire transcript of the deposition, however, that the recommendations of NUREG-0737, § II.F.1, were in fact followed when the displays of the monitors were installed.

When directly asked whether there was a human factors analysis performed before the monitors were installed, Mr. Krich responded, "I'm not aware of any. I cannot speak with certainty as to whether analysis was performed or not." And Mr. Beer responded, "[t]o my knowledge, there was no human factors analysis." (Krich, Dep. Tr. 98-99; Beer, Dep. Tr. 99.) Further, witness Beer stated that the monitor displays are located on the back panel in the control room, so that the operator has to walk around the front panel to get the readings on the back. (Dep. Tr. 99.) When asked to interpret the statement in NUREG-0737, § II.F.1, indicating that it was important that the installation of a noble gas effluent monitor not increase the potential for operator error and that therefore a human factors analysis should be performed, witness Beer responded as follows:

> The meters which display the reading from the stack gas monitors • • • should consider things like ease of readability from the operator's standpoint; does he have to look above or over or underneath to obtain the readings, or can he obtain them in an expeditious and reasonably accurate fashion? Are the procedures which relate to those meter readings easily obtained, easily referenced, and such that he doesn't have to hunt for half an hour to find what the reading means.

(Dep. Tr. 133-34.)

Then when asked "whether or not these considerations were taken into account in the *acquisition* of the stack gas monitor for Big Rock Point" (emphasis supplied), witness Beer replied, "I don't know whether they were or not." Mr. Krich, in answer to the same question, said, "[t]o the best of my knowledge, these criteria were included in the review of the system [b]y the organizations responsible for the design and procurement of the system." Those organizations, according to the witness, were PM&MP and the radiological services department; he stated that he believed that these considerations were taken into account by those two departments in the design and procurement of the system. (Krich, Dep. Tr. 135, 147-51.) The testimony indicates that human factors were taken into consideration in the design and procurement of the noble gas stack monitors, but it fails to show that any consideration of human factors was applied in determining where, in the control room, the meters were to be installed.

Fourth and last, Intervenors argue that the record on the stack gas effluent monitors is contradictory, and that some of it indicates that the add-on scales do not satisfy the regulations. The Staff feels otherwise. Staff points out that originally the Licensee did state that the monitor could not be designed to correct for changes in radionuclide mix, especially the change in mix with time after shutdown. (Krich-Beer-Emch Dep., Exhibit 4; Krich, Dep. Tr. 73-74; Emch, Dep. Tr. 74-75.) After discussions with the NRC and further study of the problem, Licensee concluded that the display guidance of NUREG-0737, § II.F.1 could be met by assuming a radionuclide mix that would be accurate early in an accident and that would remain conservative throughout the accident. (Beer, Dep. Tr. 80; Krich, Dep. Tr. 73-74.) Later into an accident, health physics personnel would be available in the Tech Support Center to take over the task of estimating dose, including making the appropriate adjustments for variation in the radionuclide mix, thus relieving the operators of this responsibility. (Emch, Dep. Tr. 74-75.)

We conclude that the record shows that the noble gas stack monitors have been installed and are consistent with the intent of NRC rules and guidelines, except for a failure by Licensee to comply fully with NUREG-0737, § II.F.1-1 in that a human factors analysis was not performed before the monitor meters were installed in the control room. Consequently, as a condition of this decision, we are directing that the NRC Staff require Licensee to conduct such an analysis now, to determine whether the meters are satisfactorily situated or whether they should be moved. Staff should subject the Licensee's analysis and conclusions to a critical review and impose any technical specifications that its review indicates are necessary.

We urge both the Licensee and the Staff to consider locating the monitor where the operator can see it without having to go behind the control panel. An operator might, understandably, be reluctant to leave his post during an emergency in order to read the noble gas effluent display, or might fail to detect a significant change in the readout because the monitor is not available for continuous observation. Indeed, the apparent lack of concern about whether an operator, under the pressure of accident conditions, would or could take the time to walk behind the control panel and read the display seems to us to be inconsistent with the expressed concern about providing the operator with a direct readout, in order to avoid errors that might result from an operator having to divert attention from controlling the plant during an accident. We would decide this issue ourselves had we before us evidence on the full human factors context in which the decision must be made. Absent that evidence, we trust that the Licensee and the Staff will, in the public interest, deal with this issue promptly and reasonably.

2. *In-Plant Iodine Monitors*

In-plant iodine sampling methods have recently been improved as a result of lessons learned from Three Mile Island. The filter medium, silver zeolite, provides accurate sampling with only negligible interference from noble gases, something not possible with previous methods. (Finding H-22.) Permanent iodine monitoring equipment using silver zeolite filters was installed at Big Rock Point in the Fall of 1982. The major components of this system are inspected at least weekly and recalibrated as necessary, at least annually. (Finding H-23.) In-plant iodine is sampled by a high-volume air sampler through which room air is drawn. The sampler contains both a particulate filter and a silver zeolite filter, which are regularly removed and analyzed. (Finding H-24.)

The Board concludes that Licensee's in-plant iodine monitoring system complies with the recommendations of NUREG-0737 and is adequate to provide a quantification of in-plant iodine.

3. *Containment Radiation Monitors*

The high-range containment radiation monitors at Big Rock Point directly measure the radiation level inside containment. They will be used to assess and follow the course of a core damage accident. (Finding H-25.) These monitors were installed in April 1982, and were approved by the NRC Staff in a Safety Evaluation Report dated October 18, 1982. (Finding H-26.) They are located just outside containment, in the cable penetration room. This placement, which is feasible because the containment has no concrete shielding, is preferable to within-containment placement because it avoids subjecting the monitors to hostile environmental conditions (such as steam, high humidity, high temperature and high pressure) during an accident. (Finding H-27.) The containment radiation monitors activate an alarm in the control room at a reading of approximately 12 rem per hour. (Finding H-28.) Plant operators have been adequately trained to read the containment radiation monitors, which is a straightforward task. (Finding H-29.)

The fact that Big Rock Point has two high-range containment radiation monitors, as suggested by NUREG-0737, provides redundancy: if one fails the other will be sufficient to provide a readout. (Finding H-30.) Further reliability is provided by redundant power sources. The two containment radiation monitors are connected by individual circuit breakers to the emergency AC power bus, which normally receives electricity from off site. (Finding H-31.) If offsite power is lost, an emergency diesel generator automatically powers the emergency bus and a second diesel generator provides additional backup. (Finding H-32.) The Licensee regularly tests both the emergency bus and the diesel generators. (Finding H-32.)

The containment radiation monitors were originally calibrated by the vendor over their entire range and are certified to remain in calibration for 18 months. They are recalibrated every year during the maintenance and refueling outage. In addition, an electronic calibration check is performed monthly. (Finding H-33.)

The calibration of all high-range radiation detectors at Big Rock Point was discussed in the report, "Evaluation of Big Rock Point Nuclear Plant," by the Institute of Nuclear Power Operations (INPO Report), a copy of which was provided the Board and parties to this proceeding on February 20, 1984. (*See* letter to Board members from Mr. Gallo, with the INPO Report, dated November 1983, and an affidavit of Joseph Leman Beer, Chemistry/Health Physics Supervisor at the Big Rock Point Plant, attached.) We have reviewed the INPO Report and the affidavit to determine whether their contents warrant our reopening the record. Only two sections appear to bear on this case, and we have determined that they do not warrant reopening the case, for reasons which we shall now explain.[14]

Finding RP.6-2 at 17 of the INPO Report recommends extending the range of calibration of all high-range radiation detectors at Big Rock Point to the region above 20 R/hr by acquiring a high-level radioactive source. This finding by INPO may well affect the high-range containment monitors; if so, we agree that extending their calibration range may be advisable. The Beer affidavit, however, attests that the high-range accuracy of the monitors will be verified annually.

Finding RP.6-1 at 16 recommends that certain portable radiation detectors should be periodically checked. According to the affidavit of Mr. Beer the portable detectors used during the installation of new racks in the spent fuel pool will be calibrated on the high-range scale before use. Moreover, portable detectors in general will be checked prior to each use or daily, whichever is less frequent, on the scale normally used, and they will be verified on the high scales annually. It appears that the Licensee is making a good-faith effort to comply with the recommendations of the INPO Report; therefore we have determined that it would not be justified for us to reopen the record.

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[14] The INPO Report was cited in Intervenor O'Neill's filing of Findings of Fact and Conclusions of Law dated February 9, 1984. He urged us to either require that the report be introduced into evidence or take judicial notice of it. Treating this request as if it were a motion, we are denying it, for reasons that become apparent in our discussion of the report.

The containment radiation monitors at Big Rock Point have error bands similar to those on other such monitors in use in the nuclear industry. (Finding H-34.) While the manufacturer specifies an error band of plus or minus 36%, Licensee's own calibration procedures specify an accuracy of plus or minus 45%. During calibration at the time of the last refueling outage, the accuracy of the monitors was determined to be within 20%. Mr. Phillips testified that the Staff considers a margin of error within an order of magnitude to be acceptable. (Finding H-35.)

This Board considers an error of plus or minus 20% to be quite reasonable and therefore acceptable. We do not, however, agree with the Staff that accuracy within an order of magnitude would be either reasonable or acceptable. Such a large margin of error could, we think, lead to an erroneous conclusion with regard to the condition of the core during an accident. Apparently this monitoring equipment can be manufactured and calibrated to produce error margins between 25 and 30%. (Phillips, Tr. 3071.) Therefore, we think Staff should use a more rigorous standard. Further, we see no reason why the Licensee's calibration procedures should not specify the error band specified by the manufacturer of the equipment, i.e., 36%. Accordingly, we shall so require.

The Board concludes that Licensee's containment radiation monitoring system has the capability to detect and measure, within an allowable margin of error, the radiation within the reactor containment during and following an accident, as recommended in NUREG-0737, Item II.F.1, Attachment 3. To provide assurance that this capability will be maintained in the future, we require as a condition to this license amendment that Licensee adopt an error band at least as narrow as that specified by the manufacturer for the monitoring equipment.

D. Licensee's Ability to Assess the Degree of Core Damage During the Course of an Accident

NUREG-0737, Item II.B.3, recommends post-accident sampling and analysis of reactor coolant and containment atmosphere as a means for determining, on a continuing basis, the degree of core damage following an accident. (Finding H-36.) Mr. Marusich testified that the Licensee has developed a calculational procedure, based on data from the containment radiation monitors, as an alternative method of assessing core damage during an accident.[15] (Finding H-37.) The Staff has approved

[15] Idaho National Engineering Laboratory (INEL) has concluded that the Big Rock Point containment high-range radiation monitors provide an adequate method of determining the extent of core damage during an accident, in spite of their vulnerability to fire occurring in the outside cable separation room.
(Continued)

Licensee's calculational procedure as an adequate alternative to post-accident sampling for estimating the degree of core damage. (Finding H-38.)

Most core damage scenarios involve the release of reactor coolant into containment. Containment radiation monitors automatically measure the radiation level generated by the coolant. The extent of damage to the core may then be estimated by comparing the actual radiation level with the estimated level that would be present following a 100% core meltdown. (Finding H-39.) Based on the source terms set forth in Regulatory Guide 1.3 (as recommended by NUREG-0737), a curve associated with 100% core melt as a function of time has been developed. (Finding H-41.) Dividing the actual radiation level following an accident by the appropriate value from this curve reveals the approximate percentage of the core which has been damaged. (Finding H-42.) The calculation is simple and will likely take less than a minute to perform, and the mathematical models used have been verified by the NRC Staff. (Finding H-48.)

The Board expressed concern that such a procedure might, under some accident scenarios, underestimate the degree of core damage. (Tr. 2966-67.) Mr. Marusich agreed that under some scenarios a 100% core melt would not produce the source terms postulated in Regulatory Guide 1.3, and that the Licensee's calculation may, therefore, underestimate core damage by as much as a factor of 2 under such circumstances. Regulatory Guide 1.3 remains the NRC Staff's best judgment of the appropriate source terms given a 100% core melt scenario. (Finding H-49.)

Furthermore, both Messrs. Marusich and Phillips testified that precise core damage information is of little value in determining whether to take public protective actions. Rather, decisions on protective actions are based on the actual radiation release rate, not the extent of core damage, and releases are measured by the high-range effluent monitors and by radiation protection technicians in the field. (Finding H-50.) The status of the core, i.e., whether it is intact or whether significant cladding failure or melt has occurred, will be used as consistency checks against such measurements. Both Licensee and Staff agree that the margin of error for the containment radiation monitors is acceptable for the purpose of making core damage estimates. (Finding H-51.)

(INEL, "Review of Big Rock Point Plant's Request for Deferral of TMI Action Plan Items Considered Non-Essential," at 13-15). The effect of such a fire would be mitigated by the existence of a secondary method of assessing dose rates inside containment and, thus, the likely extent of core damage; that method would be hand-held radiation instruments outside containment, which would provide adequate information to assess core damage. (Phillips, Tr. 3099.)

Protective action recommendations may be made based on core damage estimates in conjunction with containment radiation levels prior to a release. These readings, coupled with the Emergency Plan Implementing Procedures, which address releases outside containment, provide sufficient information to determine adequate public protective action. (Finding H-52.)

Mr. Richard Emch testified that a preliminary report by the NRC Staff has concluded that the containment radiation monitors are adequate to assess the degree of core damage, and that the coolant and atmosphere sampling recommended in NUREG-0737, Item II.B.3, is not necessary at Big Rock Point. (Finding H-53.) The Board agrees with this preliminary assessment by the Staff. The Board therefore finds that the containment radiation monitoring system, in conjunction with the calculational procedure described by Mr. Marusich, provides an alternative means to determine the degree of core damage under most accident conditions and therefore satisfies the intent of NUREG-0737, Item II.B.3.

E. Conclusion

Based on the weight of the evidence, the Licensing Board finds that inplant iodine monitors and containment radiation monitors at Big Rock Point satisfy the recommendations of Items II.H.5 and II.I.2 of NUREG-0654, as well as the more detailed performance recommendations in NUREG-0737. The noble gas effluent monitors also satisfy the recommendations of Items II.H.5 and II.I.2 of NUREG-0654, but apparently it was placed in the control room without a human factors analysis having been performed with regard to the placement of the monitor displays in the control room, as required by NUREG-0737, Item II.F.1. Consequently, the NRC Staff shall require Licensee to carry out such an analysis. The Staff shall subject that analysis to a critical review and on the basis of that review determine whether the displays are properly located to assure that they do not increase the potential for operator error.

IX. CHRISTA-MARIA SUBCONTENTION 9(8) — SUMMER AND WINTER EMERGENCY PLANS

A. Background

Christa-Maria subcontention 9(8) states:

> Applicant should have separate emergency plans for summer and winter.

This subcontention was admitted by the Licensing Board in its "Memorandum and Order (Motion to Strike Emergency Planning Contention)," dated April 20, 1982 (LBP-82-32, *supra*). The Board admitted this issue based on its finding that the proposed expansion of the storage capacity of the Big Rock spent fuel pool could increase the radiological risk to the public and Intervenors' assertion that separate emergency plans are necessary to accommodate the difficulties associated with winter weather and the large numbers of summer visitors.

Contrary to Intervenors' assertion, there is no regulatory requirement for separate emergency plans for the summer and winter seasons. However, after reviewing the record, it is evident to the Board that certain seasonal conditions, such as adverse winter weather and the presence of tourists during the summer, can potentially affect the feasibility of evacuation. Thus, whether emergency planning is contained in one or two documents is immaterial; the real question raised by Christa-Maria subcontention 9(8) is whether the Licensee, State and county emergency plans adequately accommodate varying seasonal conditions.

To address this contention, Licensee presented the testimony of six witnesses, Messrs. Sinderman, Klimm, Muma, Hess, Welch, and Sheriff Lasater. (Finding I-3.) The NRC Staff submitted the testimony of Mr. Phillips. (Finding I-4.) Intervenors presented testimony of Ms. Christa-Maria and Ms. Liane Christiansen. (Finding I-5.) The Board called Mr. Hennigan as a Board witness. (Finding I-6.)

B. Applicable Law

Section 50.47(a)(1) of 10 C.F.R. requires reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at a nuclear power plant. In furtherance of this objective, 10 C.F.R. § 50.47(b) requires that onsite and offsite emergency response plans provide a range of protective actions for the public within the plume exposure pathway EPZ.

Implementing guidance against which the Licensee, State and county emergency plans must be assessed is provided in NUREG-0654. Although NUREG-0654 does not explicitly address the accommodation of seasonal conditions in emergency planning, § II.J identifies the features of the onsite and offsite plans which must obviously reflect the seasonal vagaries of the Charlevoix area if the regulatory standards set forth in 10 C.F.R. §§ 50.47(a) and (b) are to be satisfied. These features include the evacuation time estimates, the decisionmaking process for the choice of the appropriate protective measure, and the capability to implement the public protective measure, including the identification of and means for removing potential impediments to the use of evacuation routes. NUREG-0654, § II.J(9)-(10), at 60-64.

The guidance provided by NUREG-0654 is applicable to the offsite emergency plans developed for the area surrounding Big Rock Point Plant pursuant to the Michigan Emergency Preparedness Act, 1976 Pub. Act 390, which governs emergency planning and preparedness for the State of Michigan. (Attachment No. 1.) This act vests the County of Charlevoix with the power and authority to develop emergency plans and programs in accordance with the policies and plans established by the appropriate federal agencies, including the Nuclear Regulatory Commission. Emergency Preparedness Act, § 10. The relationship between the County and City of Charlevoix with respect to the provision of emergency planning is defined by § 9(2) of the Emergency Preparedness Act, which provides: "A municipality having a population of less than 10,000 may appoint a coordinator who shall serve under the direction of the county coordinator."

In evaluating the adequacy of the emergency plans for the Charlevoix area, the Board has limited its consideration to the major difficulties seasonal conditions might pose to the implementation of the public protection measures. This proceeding concerns an application to modify an operating license in order to expand modestly the storage capacity of the Big Rock Point spent fuel pool. Under these circumstances, the Board has determined the scope of litigation on this issue is limited to gross problems due to seasonal conditions. (Tr. 3151-52.)

C. Discussion

The goal of emergency planning and the implementation of public protective measures is to minimize the public's radiation exposure during a radiological emergency. There are two primary public protective measures, evacuation and sheltering. (Finding I-11.) Timely evacuation, if feasible, is the preferred protective action since it removes the public from the source of exposure. Sheltering is the appropriate protective measure when evacuation is either impractical or cannot be timely implemented such as during adverse weather conditions which may create undue risk. (Finding I-12.) Both sheltering and evacuation are considered acceptable protective actions by the Staff under varying factual scenarios. Commission regulations and guidance require that there be adequate advance planning to anticipate and reasonably cope with foreseeable adverse conditions, but there is no requirement that evacuation be feasible under *all* fore[illegible]able circumstance.

After reviewing the contention and the testimony introduced into evidence, it is clear to the Board that the aspect of the protective measures most sensitive to seasonal conditions is the road travel necessary to evacuate the public from the plume exposure EPZ. The feasibility of road travel varies depending upon, among other things, the presence of summer transients and adverse winter driving conditions. These seasonal factors must therefore be accommodated in the decisionmaking process concerning the choice of the appropriate protective measure, and the implementation of the protective action. Notification, communication, meteorological monitoring and radiological assessment actions generally are not significantly affected by seasonal conditions. (Findings I-14 and I-15.)

The choice between protective measures and the manner in which such measures are implemented are actions to be taken by State and county officials who must rely on their own experience and judgment. (Findings I-28 through I-32, I-34, I-40 through I-43.) The officials responsible for choosing between protective measures have adequately evaluated the impact of seasonal conditions on the effectiveness of sheltering and evacuation. (Findings I-13, I-16 and I-17.) Seasonal conditions have little impact on the effectiveness of sheltering but may either lengthen evacuation travel times or make evacuation impractical or impossible. (Findings I-16 and I-17.)

Licensee, the State of Michigan and the County of Charlevoix have developed plans and procedures to aid the decisionmakers in their choice between evacuation and sheltering. (Findings I-10 through I-13.) These plans and procedures have been designed to incorporate seasonal conditions in the decisionmaking process. For example, the evacuation time estimates, an important tool designed to help the emergency response personnel charged with recommending and deciding upon protective actions, adequately account for the effect of seasonal conditions. (Finding I-18.)

HMM Associates has updated the evacuation time study performed for the area surrounding the Big Rock Point Plant in 1980. This 1980 study, relied on at the hearing in this case, was performed prior to the installation of the prompt notification system and defined adverse winter weather in a manner no longer consistent with NUREG-0654, which requires a reduction in both roadway capacity and travel speeds. (Finding I-19.) The updated time study reflects these developments and considers the impact of adverse winter weather conditions by assuming a reduction of roadway capacity and travel speeds on the order of 30%. (Finding I-21.) The updated time study also accounts for peak summer transient populations resulting from summer and recreational facilities. The updated study includes vehicle demand associated with permanent residents, seasonal residents and peak summer population, including campers. (Finding I-20.)

The updated study was completed in February 1984.[16] HMM Associates prepared interim evacuation time estimates for use by Licensee, the State, and the County of Charlevoix until the updated time study was completed. (Finding I-23.) The interim time estimates were developed for both normal and adverse weather conditions. (NUREG-0654, Rev. 1, at 4-6.) The adverse weather condition considered in the interim estimates was a winter weekday with adverse weather assumed to reduce roadway capacity and travel speeds by 30%. (Findings I-24 and I-26.)

Intervenors questioned the adequacy of the guidance provided by the evacuation time estimates because the estimates do not consider every conceivable road incident which might impede the flow of evacuation traffic. The evacuation time estimates do not consider the worst winter condition, a severe snowstorm which would make all road travel impossible, nor other highly specific roadway incidents, such as traffic accidents and whiteouts. (Findings I-21 and I-28.)

Intervenors apparently ascribe to the evacuation time studies a greater purpose than the studies are meant to serve. Although evacuation time estimates are useful devices to aid in the protective action decisionmaking process, such estimates are only one of the tools that decisionmakers will use. The evacuation time estimates cannot possibly evaluate every conceivable evacuation scenario. Information about delays due to specific roadway conditions is best obtained at the time of the emergency from knowledgeable local officials. The county and State emergency response organizations will use the evacuation time estimates as a baseline that can be modified by their own judgment based on an informed evaluation of current conditions. (Findings I-28 and I-29.) For example, under a condition in which an evacuation is not feasible, such as a heavy snowfall, the amount of time necessary to make the roads passable would be considered in conjunction with the evacuation time study. (Finding I-29.)

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16 The revised Evacuation Time Estimate study for the Big Rock Point Plant was submitted to the NRC Staff in February 1984. The Staff reported its review of the study on May 9, 1984; it found the time estimates adequate in all respects. None of these documents are on the record of this proceeding and do not provide any bases for our decision on this issue. Letter from Marck C. Furse, Attorney for Consumers Power Company, to this Board dated May 29, 1984, which enclosed a letter to Consumers Power Company from C.J. Pareirello, Chief of the Emergency Preparedness and Radiological Safety Branch of the NRC, dated May 9, 1984.

In addition, the updated time estimate will evaluate the traffic implications of special events, such as local festivals, and the potential impact of rock concerts,[17] held outside the EPZ, on the flow of evacuation traffic from the EPZ. (Finding I-20.) The updated study will also assess the effect of seasonal conditions, such as adverse winter weather, on preparation and mobilization times. (Finding I-21.)

All of the witnesses questioned concerning the matter testified that local decisionmakers can assess the impediments to evacuation posed by seasonal conditions without significant difficulty. (Finding I-28.) We accept this testimony. The organization of the county and State Emergency Operations Centers (EOC) is conducive to informed and reasoned decisionmaking. Representatives of the county agencies with road management responsibilities, such as the public works and sheriff's departments, sit at the operations table with the local officials responsible for choosing the appropriate protective measure. (Finding I-32.) Information concerning local conditions is conveyed to the State EOC. (Finding I-29.) This organization ensures the decisionmakers ready access to reliable information on local conditions.

This same organizational format provides the county emergency plan and procedures the flexibility necessary to accommodate seasonal conditions in the implementation of the protective measures. The county emergency plan handles seasonal conditions through the delegation of authority to local officials staffing the EOC operations table. (Finding I-31.) The local officials staffing the operations table will keep each other informed of local conditions and are each free to allocate the resources necessary to address special or unusual circumstances. (Findings I-31 and I-43.) Emergency response personnel, vehicles and equipment will be dispatched by the operations table to respond to seasonal conditions, including adverse winter weather, the peak summer population, and such special events as the festivals held within the City of Charlevoix. (Findings I-34, I-35, I-36, I-37, and I-43.) Evacuation routes, traffic and access control points will be selected with prevailing local conditions in mind. (Finding I-34.)

Intervenors presented testimony, primarily anecdotal, on various conditions which would impede evacuation, such as snow-induced road closings, "whiteouts" which drastically reduce traffic flow, rock concerts, and bridge failures. (Christa-Maria, Tr. 3422-61; Christiansen, Tr.

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[17] Intervenor Christa-Maria placed substantial emphasis on traffic congestion caused by rock concerts. The Board notes, however, that the concert site is outside the EPZ (Tr. 3433), that such concerts are only held in the summer, between Memorial Day and Labor Day (Tr. 3438-39), and that the traffic flow would be away from the City of Charlevoix in the event of a nuclear emergency at Big Rock Point, each a factor in minimizing the impact on evacuation of the 5-mile EPZ.

3475-79.) While such incidents are not specifically mentioned in the local emergency plans, the plans and implementing procedures are sufficiently flexible to allow local officials responsible for implementing protective measures to adequately fashion a response to such cited conditions.

Many of Intervenors' questions concerned the relationship between the City and County of Charlevoix. Intervenors asserted that emergency planning cannot be adequate until the City of Charlevoix develops its own emergency plan. We do not agree. The Michigan Emergency Preparedness Act, *supra*, which governs emergency planning for the State of Michigan, vests the County of Charlevoix with the responsibility for emergency preparedness for the City of Charlevoix. The County of Charlevoix is also responsible for the implementation and coordination of public protective actions for the City of Charlevoix. The uncontroverted evidence demonstrates that the County of Charlevoix is satisfying these responsibilities; further, the county has an excellent cooperative relationship with the City of Charlevoix. (Welch, ff. Tr. 3235, at 6; Welch, Tr. 3241, 3380; Lasater, Tr. 3253-55; Muma, Tr. 3259-61, 3264, 3404.) We note, also, that the Federal Emergency Management Agency (FEMA) has issued its formal findings on the adequacy of offsite emergency preparedness around Big Rock Point and has determined that the State and local plans and preparedness for the Big Rock Point Plant are adequate to protect the health and safety of the public living in the vicinity of the plant.[18] (48 Fed. Reg. 22,795-96; Tr. 3587, *et seq.*)

There is one respect in which our review of the record causes us to require further consideration in the emergency planning process. Our concern is that rock concerts and other special events at the Castle Farms site undoubtedly cause serious congestion. Unlike severe winter weather, whose effects appear to have been considered, our record reflects the potential seriousness of congestion from these events but does not indicate that adequate attention has been given to resolving the problem.[19] Given the short duration of the events and the low probability that they would coincide with an accident, we will not order or supervise a specific remedy. However, we will require consideration of remedies, which could include new exit roads or road improvements if the community

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[18] We are reluctant, however, to place exclusive reliance on the FEMA findings. Testimony by a FEMA witness during the 1982 hearings in this proceeding caused us to doubt, at that time, the adequacy of FEMA's findings with respect to emergency preparedness at Big Rock Point Plant. (*See* LBP-82-77, *supra*, 16 NRC at 1101). At the recent hearings, the Staff presented no FEMA witnesses to testify with respect to the FEMA findings. Thus we view FEMA's findings as merely confirming the other evidence.

[19] HMM Associates, "Evacuation Time Estimates for Areas Near the Big Rock Point Nuclear Power Plant" (February 1984) at 7-5 discusses "traffic management" as an approach but does not state whether other measures may be feasible.

decides that benefits from such improvements would serve purposes other than just emergency evacuation, for which purpose the expense may not be justified.

D. Conclusion

Based on the weight of the evidence, the Licensing Board concludes that the Licensee, State and County of Charlevoix emergency plans and procedures adequately accommodate seasonal conditions, including the adverse winter weather and the summer transient population, satisfy the requirements of 10 C.F.R. § 50.47(b) and the guidance provided by NUREG-0654, § II.J, and that separate emergency plans for the summer and winter are neither necessary nor required. The one exception to our conclusion is the need for further consideration of ways to ameliorate congestion that could affect evacuation during events at the Castle Farms site.

X. CHRISTA-MARIA SUBCONTENTION 9(9) — SPECIAL EVACUATION MEASURES FOR CHILDREN AND PREGNANT WOMEN

A. Background

Christa-Maria subcontention 9(9) states:

> Appropriate emergency plans should be made for children and pregnant women to evacuate at appropriate levels of radiation, considering their special susceptibility.

In its "Memorandum and Order (Motion to Strike Emergency Planning Contention)," dated April 20, 1982 (LBP-82-32, *supra*), the Board noted that Licensee had not asserted that its emergency plan made specific provisions for children and pregnant women, nor had Licensee argued that such a provision would not be helpful. The Board ruled that this subcontention, properly interpreted, alleges that a specific provision for children and pregnant women must be included in the emergency plan because it is required for the reasonable protection of the public. The Board admitted the subcontention on this basis.

Licensee submitted the testimony of Roger W. Sinderman, a health physicist with a Masters of Public Health degree from the University of Michigan. (Finding J-2.) Monte Phillips, an emergency preparedness analyst, testified for the NRC Staff. (Finding J-3.)

B. Applicable Law

NRC regulations and supporting guidance require that specific and appropriate consideration be made in emergency plans for the early evacuation of children and pregnant women in the event of a nuclear power reactor accident involving a radiological release. The regulations require that both Licensee's site plans and the offsite emergency plans must provide a range of protective actions for the public, and they must, consistent with federal guidance, provide guidelines for the choice of such actions. 10 C.F.R. § 50.47(b)(10). As the implementing guidance makes clear, the proviso for early evacuation of children and pregnant women is one such protective action.

NUREG-0654 contains general criteria for the preparation of emergency plans; as regards § 50.47(b)(10), implementing guidance is found in Criterion II.J.7 for licensees and in Criterion II.J.9 for State and local governments. Both criteria reference the "Manual of Protective Action Guides for Nuclear Incidents," EPA-520/1-75-001 (hereinafter referred to as "EPA-520") as the source of acceptable dose levels for the protection of the population-at-risk. EPA-520, which was written by the U.S. Environmental Protection Agency (EPA), generally recommends early evacuation for children and pregnant women whenever the whole-body dose is projected to be about 1 rem. (EPA-520, § 2.3 and Table 2.1.) Thus, the Board will review the evidence to determine whether the emergency plans of the Licensee, State of Michigan and Charlevoix County comply with the foregoing regulations and federal guidance.

C. Discussion

This subcontention presents two issues for the Board's determination. First, whether the cognizant emergency plans address the potential need for early evacuation of children and pregnant women, and second, whether the decisionmaking criterion for early evacuation is adequate. Both issues are decided in favor of the Licensee.

It is undisputed that small children and the developing embryo and fetus are more sensitive to radiation than are adults and that therefore emergency plans should provide guidance for their early evacuation in the event of a nuclear accident. (Findings J-4 and J-5.) The evidence is equally clear that Licensee's site emergency plan for Big Rock Point, and the State of Michigan and Charlevoix County offsite plans for the 5-mile plume exposure pathway EPZ at Big Rock Point, both properly implement the 1-rem guideline for the early evacuation of small children and pregnant women. (Findings J-6, J-7 and J-8.) Therefore, the Board finds that the cognizant emergency plans properly recognize and provide for the early evacuation of small children and pregnant women at the 1-rem guideline recommended by EPA and NRC.

Whether the 1-rem guideline is adequate as the triggering point for early evacuation is less obvious. Neither the Commission's emergency planning regulations nor NUREG-0654 provide any detailed explanation of the reasoning that led to the selection of the 1-rem value. It was simply adopted from EPA-520. It does appear that the 1-rem value was selected by EPA on a reasoned basis. EPA balanced the risk of health effects due to radiation exposure against the health and other societal risks of evacuation. (Finding J-9.) However, the witnesses for Licensee and the NRC Staff were unable to provide details of the cost-benefit process used by the EPA to establish the 1-rem guideline. The inability of these witnesses to provide this information is understandable. We are unable to derive any further insight from our own perusal of EPA-520. The document simply does not provide these details.

A guideline established by a federal agency charged with the protection of the environment is entitled to be accorded substantial evidentiary weight. Consequently, we would normally accept the EPA guideline despite the desirability of additional background information, especially in a case such as here where there is no contrary evidence. However, we need not rest on this basis alone. The testimony of Mr. Sinderman, Licensee's witness, provides additional support for our determination that the 1-rem EPA guideline is adequate.

Mr. Sinderman testified that no verified health effect has been measured at doses of 1 rem to human fetuses. These effects include both birth defects and the potential for contracting cancer after birth. (Finding J-10.) Nevertheless, the lack of demonstrable evidence in this regard and the innate conservatism of radiation experts cause them to assume that such effects occur at this low dose rate. Consequently, by the linear extrapolation of the risk of radiation exposure from known health effects at high dose rates, similar risks are mathematically calculated for low dose rates. (Finding J-11.) Thus, on the basis of these assumed health effects, the risk to the fetus from an acute dose of 1 rem is about 1 in 1000 to 1 in 10,000. The Licensing Board concludes that this range presents a minor risk in comparison with the overall risk that in a population of unexposed women, 40% of conceptions will exhibit some type of reproductive failure. (Finding J-12.) Consequently, we think it is adequate to advise pregnant women of this risk, as Licensee plans to do, rather than to recommend evacuation as a matter of government policy.

We note that until recently both the State of Michigan and the Charlevoix County emergency plans provided for early evacuation for children and pregnant women at 0.5 rem. These plans were recently revised to conform with the EPA guideline, which sets 1 rem as a lower threshold for such evacuation. (Finding J-8.) The CPC witness on this matter points out that the 1-rem level is only twice the dose permitted to the general public in *each year* for an individual in an unrestricted area (Sinderman, ff. Tr. 3511, at 2) and twice what would be allowed for a pregnant woman in routine employment (*id.* at 7). We further note that the public will be kept informed of dose at any level in the event of a site emergency at the plant and will be given accompanying evaluative information to help people make more informed decisions about their individual courses of action during the emergency. (Sinderman, Tr. 3547.) The guidelines and the supplementary information which will be provided the public appear to us to represent a reasonable approach to protecting pregnant women and children.

D. Conclusion

Based on a careful consideration of the evidence, the Board concludes that the emergency plans of the Licensee, State of Michigan and Charlevoix County provide specific coverage for the early evacuation of small children and pregnant women. The Board further concludes that the EPA guideline of 1 rem is a sufficiently low value to assure meaningful early evacuation.

Order

For all the foregoing reasons and based on consideration of the entire record in this matter, it is, this 29th day of August 1984,

ORDERED

1. As a condition of the issuance of a license amendment, Consumers Power Company (Licensee) shall adopt and observe a technical specification to assure that the Big Rock Point Nuclear Power Plant (Plant) will not operate if the capacity of the makeup line to remove heat from the spent fuel pool exceeds the heat generating capacity of the pool inventory, considering the power history of each assembly in the pool, the number of assemblies in the pool and the effect of lake temperature on the heat-removal capacity of the makeup water system.

2. As a condition of the issuance of a license amendment, Licensee shall adopt and observe procedures prohibiting the storage of materials in the area between rack B and the east wall of the spent fuel pool. (*See* Finding A-59.)

3. As a condition of the issuance of a license amendment, Licensee shall adopt and observe procedures that will prohibit the use of its

gantry crane for loads of over 24 tons. A change in this procedure for a specific exception or in general may be made with permission from the Staff of the Nuclear Regulatory Commission.

4. As a condition of the issuance of a license amendment, Licensee shall adopt and observe procedures to assure that only spent fuel with a decay time of at least 1 year will be stored in the outer three rows of the fuel rack adjacent to the south wall of the fuel pool. These procedures shall require a prompt investigation by the company whenever radiation in the sock tank area exceeds 50 mrem/hr.

5. As a condition of the issuance of a license amendment, Licensee shall agree to promptly conduct a human factors analysis of the meter for its noble gas stack monitor, including an analysis of the readability of the add-on scale and of the acceptability of the placement of the meter other than on the main control panel. The Staff shall take appropriate action upon receiving and reviewing this study.

6. As a condition of the issuance of a license amendment, Licensee shall adopt and observe procedures to calibrate its containment radiation monitors within the manufacturer's error band of plus or minus 36%.

7. As a condition of the issuance of a license amendment, Licensee shall agree to promptly advise State and local planning authorities of the view of the Atomic Safety and Licensing Board that further consideration should be given to whether there are practicable means of expediting an evacuation that might be required during a rock concert or other major event at the Castle Farms site.

8. Pursuant to 10 C.F.R. § 2.760 of the Commission's Rules of Practice, this initial decision shall become effective immediately. It, and the prior Partial Initial Decisions and Memoranda and Orders issued in this case, will constitute the final decision of the Commission forty-five (45) days from the date of issuance, unless an appeal is taken in accordance with 10 C.F.R. § 2.762 or the Commission directs otherwise. *See also* 10 C.F.R. §§ 2.764, 2.785 and 2.786.

9. Any party may take an appeal from this decision by filing a Notice of Appeal within ten (10) days after service of this Initial Decision. Each appellant must file a brief supporting its position on appeal within thirty (30) days after filing its Notice of Appeal (forty (40) days if the Staff is the appellant). Within thirty (30) days after the period has expired for the filing and service of the briefs of all appellants (forty (40) days in the case of the Staff), a party who is not an appellant may file a brief in support of or in opposition to the appeal of any other party. A responding party shall file a single, responsive brief *only* regardless of the number of appellants' briefs filed. (*See* 10 C.F.R. § 2.762).

10. Pending motions by John O'Neill are covered in part by this order and in part by a separate Memorandum and Order.

11. Time deadlines in this order are applicable to John O'Neill, despite the hardship this undoubtedly will cause him.

THE ATOMIC SAFETY AND
LICENSING BOARD

Peter B. Bloch, Chairman
ADMINISTRATIVE JUDGE

Dr. Oscar H. Paris
ADMINISTRATIVE JUDGE

Mr. Frederick J. Shon
ADMINISTRATIVE JUDGE

Bethesda, Maryland

Findings of Fact

NOTE: The Board has used Licensee's Findings of Fact, as modified by the Staff, as the framework within which its Findings were adopted. To facilitate correct textual references, we retained the original numbering and lettering of findings even when entire findings were deleted by us in the course of our review.

A. O'Neill Contention III.E-2 — Makeup Water System

A-1. O'Neill Contention III.E-2 states:

A. How reliable is the remotely activated makeup water system which will be added to the spent fuel pool? How reliable does it need to be? How many gallons per minute will it be able to make up?

B. Will zircaloy react with steam in a fuel pool which is boiling because its cooling system has failed? Will the reaction become self-sustaining?

A-2. David P. Blanchard is a technical engineer at the Big Rock Point Plant. He has been assigned to the spent fuel pool expansion project since 1978, and was involved in developing a system for remote activation of water addition to the spent fuel pool. Mr. Blanchard's testimony addresses several aspects of this makeup water system. First, Mr. Blanchard's testimony describes the makeup system and its function in an accident situation where normal pool cooling has been lost. Second, he addresses the reliability of the makeup system based on a single-failure criterion. Third, he details the flow capacity of the makeup line under a variety of single-failure circumstances. Finally, Mr. Blanchard considers, and rejects, the possibility that zircaloy might react with steam in the spent fuel pool. (ff. Tr. 3770.)

A-3. Dr. James H. Stuhmiller and Dr. David A. Sargis are "scientists" at JAYCOR, an engineering and scientific research and development firm in San Diego, California. Dr. Stuhmiller has developed computer programs for calculating the dynamics of fluids. Such programs are called EITACC (Equation-Independent-Time-Average-Conformal-Coordinates). Both Dr. Stuhmiller and Dr. Sargis have developed and validated a three-dimensional stratified flow version of the EITACC computer code for application in spent fuel pool (SFP) studies. Using the EITACC-SFP computer code, JAYCOR performed a thermal hydraulic analysis of the Big Rock Point spent fuel pool to determine the temperature distribution within the pool under the 150°F boundary condition established by Licensee. The testimony of Dr. Stuhmiller and Dr. Sargis provides a detailed description of this analysis, its results, and the empirical techniques which were used to validate the results. (ff. Tr. 3849.)

A-4. Arthur K. Smith, a senior engineer in Licensee's nuclear plant support department, determined the structural adequacy of the makeup pipe under dead weight, pressure, thermal and seismic loading conditions. His testimony considered the adequacy of the makeup pipe under all seismic loading conditions. (ff. Tr 3897.)

A-5. Staff presented four witnesses. Fred Clemenson, a principal systems analyst, and Richard L. Emch, Jr., the Big Rock Point project manager, testified to the makeup system's purpose, its capacity to deliver a sufficient rate of flow to the pool, and its reliability. Mark A. Caruso, a senior systems engineer, explained the basis for the Staff's conclusion that spent fuel pool water temperature can be maintained at or below 150°F by using the makeup system if normal pool cooling is lost. Finally, Dr. Pei-Ying Chen, a senior mechanical engineer, assessed the adequacy of the seismic design of the makeup pipe. (ff. Tr. 3979.)

A-6. The makeup system is designed to operate as a result of a postulated loss-of-coolant accident ("LOCA") which causes core damage and results in the long-term uninhabitability of the containment building. (Blanchard, ff. Tr. 3770, at 4; Clemenson, Emch, ff. Tr. 3979, at 4.)

A-7. The normal spent fuel pool cooling system is not qualified for such an accident environment, and thus it is assumed that the accident would disable this system. (Motion of Consumers Power Company for Summary Disposition on Pleadings, Testimony of David P. Blanchard Concerning Christa-Maria Contention 8 and O'Neill Contention III.E-2, at 5-6, October 5, 1981.) Absent the makeup water system, if the cooling system fails, heat from the spent fuel will cause the water in the spent fuel pool to heat up, to exceed the design basis of the concrete and eventually boil and evaporate. (Blanchard, ff. Tr. 3770, at 11-12.)

A-8. Under the postulated accident scenario, the makeup water system will not begin operating until core cooling is provided by the recirculation of water that has collected at the bottom of the containment building. Two core spray pumps recycle water from the containment floor, through the core spray heat exchanger where it is cooled to 100°F, and then directed back to the core. (Blanchard, ff. Tr. 3770, at 5-7; Clemenson, Emch, ff. Tr. 3979, at 4.)

A-9. A minimum of 28 gallons per minute (gpm) of the water drawn from containment by the core spray pump is diverted through the makeup system to the spent fuel pool. This process will begin between 4 and 24 hours after the onset of the accident. (Blanchard, ff. Tr. 3770, at 7-10, 27-28; Clemenson, Emch, ff. Tr. 3979, at 4-5; Blanchard, Tr. 3847; Emch, Tr. 4028-29.)

A-10. It is necessary to cool the spent fuel stored in the pool by keeping it covered with water. (Blanchard, ff. Tr. 3770, at 11.)

A-11. If the possibility of the spent fuel becoming uncovered were credible, the potential for fuel melting and a zircaloy/steam reaction would require analysis. (*Id.* at 31-32.)

A-12. The makeup system must also keep the pool completely full of water, in accordance with the assumptions made by Dr. Prelewicz in determining pool moderator conditions and Dr. Kim in performing the criticality analysis for the spent fuel racks. (*Id.* at 11-12.)

A-13. The makeup system will also prevent steam generation that could result in containment repressurization. (*Id.* at 13.)

A-14. The design outlet temperature of the core spray heat exchanger is 100°F. The maximum predicted decay heat generation rate will be 6 x 10^5 Btu/hr, based on Licensee's expectation that it will discharge twenty-five fuel assemblies into the pool following a refueling outage of at least 30 days. (*Id.* at 8-9; Tr. 4030.)

A-15. Licensee has committed to maintaining the 150°F bulk pool temperature restriction because that is the pool's design basis — the integrity of the concrete pool cannot be demonstrated by analysis for higher temperatures. (Blanchard, ff. Tr. 3770, at 3.)

A-16. One hundred and fifty degrees Farenheit is the temperature below which the American Concrete Institute Code indicates that loss of concrete strength is not significant. (*Id.* at 8, 9.)

A-17. For purposes of evaluating the reliability of the makeup water system, that system is defined to include those portions of the core spray recirculation system on which the fuel pool relies for its source of makeup cooling water. This includes all piping and active components between the suction strainers in the bottom of containment and the fuel pool as well as the piping and active components associated with providing cooling water to the shell side of the core spray heat exchanger. (*Id.* at 5; Tr. 3353-54, 3373.) For purposes of evaluating seismic design, however, only the makeup pipe itself is relevant. (Tr. 3373.)

A-18. There are three pairs of active components in the system feeding the makeup line: two core spray pumps, two fire pumps and two core spray heat exchanger valves (MO-7066 and VPI-5). (Blanchard, ff. Tr. 3770, at 15.)

A-19. These active components are located outside containment, and will not be required to operate in an accident environment. (*Id.* at 15.)

A-20. Either of the two core spray pumps is sufficient to provide cooling water to the spent fuel pool and the reactor vessel. Similarly, either of the two fire pumps is sufficient to provide cooling water to the shell of the core spray heat exchanger through either of the two core spray heat exchanger valves. (*Id.* at 7, 15; Blanchard, Tr. 3791.)

A-21. The power for each core spray pump is supplied by a separate AC power bus. The normal power source for these buses is off site. Either of these buses can be transferred to the emergency power bus if offsite power is lost. (Blanchard, ff. Tr. 3770, at 15; Blanchard, Tr. 3841.)

A-22. The emergency power bus receives power from either of two onsite emergency diesel generators. (Blanchard, ff. Tr. 3770, at 15-16.)

A-23. One fire pump (electric fire pump) is powered by the emergency power bus. The second fire pump is diesel driven. The electric fire pump can be powered by either of the two emergency diesel generators if offsite power is lost. (*Id.* at 16.)

A-24. One of the core spray heat exchanger valves (MO-7066) is an AC-powered, motor-operated valve remotely actuated from the control room. In an emergency it can be powered by either of the two diesel generators via the emergency bus. Alternatively, this valve can be manually operated by a handwheel. The second core spray heat exchanger valve (VPI-5) is hand operated. (*Id.*)

A-25. The remaining components in the system feeding the makeup line are passive. They include the suction and discharge of the core spray pumps, the core spray heat exchanger, the makeup line and valve to the fuel pool, and the piping between the fire pumps and the core spray heat exchanger shell. These components have been designed not to fail. (*Id.* at 18-19.)

A-26. The majority of the passive components in the system feeding the makeup line are located outside of the containment where there are no lines containing high-energy primary coolant. Therefore, these components are not vulnerable to pipe whip or steam impingement or to the hostile environmental conditions inside containment following an accident like TMI-2. Further, the makeup line is routed so that it is not located near the reactor primary coolant line and thus both could not be damaged by the drop of a heavy object such as a cask. This routing also makes it unlikely that a failure of the primary coolant system leading to a LOCA could simultaneously cause a failure of the pool makeup system. (*Id.* at 18-19.)

A-27. No credible mechanisms could cause failure of any passive components following a LOCA. (*Id.* at 18; Blanchard, Tr. 3788.)

A-28. The makeup line is 190 feet long and consists of 115 feet of 2-inch-diameter piping and 75 feet of 1-inch-diameter piping. (Blanchard, ff. Tr. 3770, at 5; Clemenson, Emch, ff. Tr. 3979, at 3; Blanchard, Tr. 3781.) Given the pipe diameters, there is no credible possibility of pipe blockage by crud, scale, rust, or other foreign objects. (Blanchard, Tr. 3807; Emch, Tr. 4033-34.) As an additional precaution, the pipe will be flushed each year with rust-inhibiting chromated water. (Blanchard, Tr. 3943-44.)

A-29. One surveillance test, a hydrostatic test of tubes in the heat exchanger, is performed while the plant is at power. This test temporarily removes the core spray heat exchanger from service. This test is performed each month, during which the makeup system is inoperable for no longer than 4 hours. Since the test is brief, a LOCA is not likely to occur during this time. However, if a LOCA does occur during the surveillance test, the makeup system can be made fully operational before the water in the containment has reached 14 feet and recirculation is initiated. The valve realignments required to return the system to operation are all external to containment and entry into containment is not required to return the heat exchanger to service. The operator is expected to terminate the test and return the heat exchanger to service immediately upon the occurrence of any reactor trip. (Blanchard, ff. Tr. 3770, at 21-23.)

A-30. Licensee has established comprehensive valve position controls and verification procedures to assure that plant workers will not inadvertently render the makeup system inoperable through the mispositioning of valves. (*Id.* at 18-21.)

A-31. The operator can add water to the pool through the makeup line even if neither of the two core spray pumps is running, even if the core spray heat exchanger is isolated and even if there is no water in the bottom of the containment. This is accomplished by opening motor-operated valve MO-7072. This valve can be electrically or manually operated from outside containment. Opening this valve routes water from the fire pumps directly to the spent fuel pool. (*Id.* at 24; Clemenson, Emch, ff. Tr. 3979, at 4, 6.)

A-32. For the best-estimate decay heat rate of 176,000 watts, a 24-gpm makeup rate is sufficient to prevent the bulk pool water temperature, and hence the concrete temperature, from exceeding 150°F. (Blanchard, ff. Tr. 3770, at 9.) A larger decay heat rate of 217,000 watts would require a 30-gpm makeup rate to prevent the bulk pool water temperature from exceeding 150°F. (Caruso, ff. Tr. 3979, at 3-4; Stuhmiller, Sargis, ff. Tr. 3849, at 5.)

A-33. Licensee will institute a technical specification to assure the adequacy of the makeup system before plant startup following any outage where spent fuel has been discharged into the pool, taking into account the power history of each assembly discharged into the pool, the number of assemblies stored in the pool, and the effect of lake temperature on the temperature of the makeup water. (Emch, Tr. 4008; Blanchard, ff. Tr. 3770, at 28-29.)

A-34. Flow testing of the makeup line will be performed before startup after each refueling to make certain the line is free of obstructions. (Blanchard, ff. Tr. 3770, at 28-29.)

A-35. Licensee has used standard hydraulic analysis techniques to determine the water flow rate through the makeup line under a variety of conditions. A mass and energy balance assessment was performed to evaluate flow to the pool through the piping associated with the makeup system. Flow resistance was analyzed using standard algorithms. (*Id.* at 25; Clemenson, Emch, ff. Tr. 3979, at 5.)

A-36. The algorithms were incorporated in a computer program called FLOWNET. FLOWNET was used by Licensee to establish the adequacy of core spray and enclosure spray flows following a postulated LOCA. FLOWNET was also used to design the makeup line to obtain adequate flow to the pool without diverting so much water that the adequacy of flow to the core might be jeopardized. (Blanchard, ff. Tr. 3770, at 25.)

A-37. Two models which include the makeup line were analyzed with FLOWNET: the core spray system in a recirculation mode following a LOCA and the fire protection system with valve MO-7072 in position to supply water directly to the pool. Several cases were run assuming various configurations of the core spray and fire protection systems, including single failures of components in the core spray system. In addition, a series of flow tests on the system were performed. (*Id.* at 26.)

A-38. In no cases were the calculated flows to the core spray lines or the spent fuel pool makeup line below the required minimum rates, even where the worst single active failure was assumed. (*Id.* at 27-28.)

A-39. (Deleted.)

A-40. The makeup pipe is made of Schedule-80 carbon steel. (Smith, Tr. 3923.) A dropped wrench would not dent the pipe enough to stop or significantly reduce water flow. (Smith, Tr. 3928; Blanchard, Tr. 3802.) Nor could the pipe be crushed by being stepped on. (Smith, Tr. 3936-37.) In addition, Licensee has promulgated administrative controls to prevent fuel elements from falling on or near the makeup line. (Smith, Tr. 3929.)

A-41. The structural adequacy of the makeup line under seismic loading conditions was determined by computing potential pipe stresses using the ADLPIPE computer code and comparing these stresses to those allowable under applicable piping and support codes. The maximum potential pipe stress under seismic loading is approximately 8800 psi, while the allowable stress is 36,000 psi. (Smith, Licensee's Prepared Testimony, at 2-3, ff. Tr. 3897; Chen, NRC Staff Prepared Testimony, at 2-3, ff. Tr. 3979.)

A-42. That the makeup line crosses expansion joints has no significant effect on its seismic capability. (Smith, Tr. 3938.) The motion of the expansion joints is very small and will not significantly affect pipe stress. (Smith, Tr. 3940-42, 3957-60.) All pipe supports were evaluated in accordance with American Institute of Steel Construction (AISC) Manual. (Chen, ff. Tr. 3979, at 2.) (Smith, Tr. 3949-50.) Anchor bolts meet NRC guidelines concerning a factor of safety of 4. (Chen, ff. Tr. 3979, at 3.)

A-43. JAYCOR performed a thermal hydraulic computer analysis of the Big Rock Point spent fuel pool to determine whether, despite an average water temperature of 150°F, the pool walls and floors might be subject to higher temperatures in localized areas. (Stuhmiller and Sargis, ff. Tr. 3849, at 3.)

A-44. To determine the greatest temperature which could develop in the spent fuel pool, it was necessary for JAYCOR to calculate the circulation patterns which carry heat away from the fuel elements. (*Id.* at 7.)

A-45. JAYCOR's EITACC-SFP computer program solves the equations governing buoyant flow in spent fuel pool geometries, taking into account the location of the inlet cooling water, the location of the exiting flow, and the geometric blockage and flow resistance of the spent fuel racks. The program output presents detailed data on the temperature and flow quantities in every part of the pool. (*Id.* at 8.)

A-46. JAYCOR divided the spent fuel pool into 1430 computational volumes, in each of which the equations governing buoyant flow were solved. The shape of each computational volume was chosen to capture various geometric features of the pool, including the space between the spent fuel racks and the floor, the large gaps between the racks and the east and south walls, and the shapes of the racks themselves. Certain small gaps between the racks and between the racks and the north and west walls were not explicitly modeled, but their contributions to the total flow area were taken into account. (*Id.* at 8-9.)

A-47. The makeup system was assumed to be operating and pouring 100°F water onto the top of the northeast corner of the pool at a flow rate of 30 gpm and a heat generation rate of 217,000 watts. (*Id.* at 4-5.)

A-48. Several conservative assumptions were made by JAYCOR to maximize localized temperatures. Of the 217,000 watts to be generated by the spent fuel assemblies, assemblies generating 82% of that heat rate were placed in the F rack with 62% of the total heat rate located in the northwest corner. (*Id.* at 9; Stuhmiller, Tr. 3868-69; Sargis, Tr. 3905.) It was assumed that heat was not lost through the walls, floor, or pool surface. (Stuhmiller and Sargis, ff. Tr. 3849, at 9; Sargis, Tr. 3890.)

A-49. Under the most severe loading distribution of fuel, the highest temperature on the pool floor varied from the average by no more than 0.4°F, while the highest wall temperature reached only 2.7°F greater than the average. (Stuhmiller and Sargis, ff. Tr. 3849, at 10-11; Stuhmiller, Tr. 3869.)

A-50. The design basis for the makeup water system was initially conceived by Licensee to be 30 gpm as the maximum amount of flow that could be diverted from the core spray system to feed the makeup line under the worst-case conditions. Worst-case conditions were defined to be the recirculation mode with a containment spray valve; the open containment spray valve is the worst single active failure. The system, as designed and tested, delivers a minimum of 28-gpm flow rate. (Blanchard, Tr. 3768-69; ff. Tr. 3770, at 25-28; Emch, Tr. 4028-30.)

A-51. Using a 28-gpm flow rate with a heat rate of 205,000 watts, JAYCOR determined that the general circulation patterns predicted at the high flow and heat rates remained unchanged. The only difference was a drop of 0.1°F in the temperature at the warmest spot in the pool (2.7°F to 2.6°F). (Stuhmiller and Sargis, ff. Tr. 3849, at 11.)

A-52. The EITACC-SFP computer code used in the thermal-hydraulic analysis was verified under JAYCOR's quality assurance program. (*Id.* at 13-14.)

A-53. Data recorded in cold-leg injection experiments performed by the Electric Power Research Institute have provided extremely detailed information on the mixing of cold and warm streams in a variety of turbulent flow situations. The JAYCOR computer code was applied to this problem and its predictions were compared with the measured temperature fields. Despite the complexity of the phenomena, the JAYCOR calculations were generally within a degree of the measured values and always within the scatter of the experimental data. (*Id.* at 17-18.)

A-54. An attempt was made by another company to actually measure the temperature and flow patterns in the Maine Yankee spent fuel pool during a refueling outage in 1982. In most locations within the pool, however, the measuring devices were not sufficiently sensitive to measure the convective flows that developed. The JAYCOR computer code accurately predicted that the flow velocities would often be beneath the range of the measuring devices. In addition, the computer code was able to reproduce the pool temperature data to within experimental error. (*Id.* at 18-19.)

A-55. JAYCOR performed a scale model experiment to develop data on convective flow patterns in operating spent fuel pools. (*Id.* at 19.) This experiment was presented at the hearing by means of a full-color movie. (Tr. 3859-63; Consumers Power Company Exhibit 17.) The experimental model was used as an additional way to verify the EITACC-SFP computer code. (Stuhmiller, Tr. 3882.) Computer calculations were performed to correspond to the model tests. These calculations were then compared with the actual average and local temperatures and with the observed circulation patterns. (Stuhmiller and Sargis, ff. Tr. 3849, at 20.) The EITACC-SFP computer code was generally accurate to within half of a temperature degree, while the maximum error was between 1 and 2 temperature degrees. (Sargis, Tr. 3903.)

A-56. There are additional items in the spent fuel pool besides spent fuel and spent fuel racks. (Blanchard, Tr. 3830.) Some of these items are stored in the pool on a permanent basis, including special racks containing control blades and a small amount of research and development equipment. (Blanchard, Tr. 3830, 3866.) Other, smaller, items are stored in the pool temporarily and then sent off site. Radioactive maintenance materials are stored temporarily in buckets on the floor of the pool for biological shielding purposes. (Blanchard, Tr. 3870, 3874.) Large casks are periodically stored in a designated area in the southwest corner of the pool. (Blanchard, Tr. 3871.)

A-57. All equipment of significant size permanently stored in the spent fuel pool was taken into account in JAYCOR's computer model. (Blanchard, Tr. 3866.) Moreover, as a conservatism JAYCOR included in its calculations more racks than are presently in the pool. (Stuhmiller, Tr. 3872, 3884.)

A-58. The placement of additional objects on the floor of the spent fuel pool could, under certain circumstances, block or divert flow patterns and influence local temperatures. (Stuhmiller, Tr. 3951.) However, local temperatures will not be affected as long as important flow patterns are not blocked. (Stuhmiller, Tr. 3952.) In this case, the important flow pattern is through the space between rack B and the east wall of the pool. This space, if not blocked, will provide the necessary cooling path, and local temperatures will remain consistent with the JAYCOR analysis. (Stuhmiller, Tr. 3953-55.)

A-59. Licensee has agreed to issue written administrative procedures prohibiting the storage of any materials in the area between rack B and the east wall of the pool. (Blanchard, Tr. 3955-56.)

A-59a. Mr. Caruso of the NRC Staff reviewed the JAYCOR analysis. Based on that review and on the Licensee's proposed technical specification verifying cooling capacity, Mr. Caruso concluded that the pool water will be well mixed by natural circulation and that a bulk pool water temperature of 150°F will be maintained with a maximum localized water temperature of less than 153°F. (Caruso, ff. Tr. 3979, at 4-5.)

A-60. Big Rock Point fuel cladding is made of zircaloy, which can react with steam at high temperatures. The reaction rate becomes significant only at or above temperatures of approximately 2200°F. (Blanchard, ff. Tr. 3770, at 31.)

A-61. The temperature of spent fuel cladding at Big Rock Point could approach 2200°F only if the spent fuel became uncovered by water. Operation of the makeup system will prevent the spent fuel from becoming uncovered. Operation of the makeup line will prevent the bulk pool water temperature from exceeding 150°F. (*Id.* at 31-32.)

B. Christa-Maria Contention 8 and O'Neill Contention III.E-2 — Integrity of the Concrete Pool Structure

B-1. These two contentions are identical and state:

> The occurrence of an accident similar to TMI-2 which would prevent ingress to the containment building for an extended period of time would render it impossible to maintain the expanded spent fuel pool in a safe condition and would result in a significantly greater risk to the public health and safety than would be the case if the increased storage were not allowed.

Genuine issue of fact (5), admitted in the Licensing Board's Memorandum and Order (Concerning Motions for Summary Disposition), February 19, 1982, LBP-82-8, *supra*, 15 NRC at 312, states: "Is the concrete in the fuel pool strong enough to resist a temperature of 247°F and point loading from the storage racks?"

B-2. Licensee subsequently committed to installing a modified remotely activated makeup water line to the spent fuel pool that will maintain 150°F as the maximum bulk pool temperature (Letter of J. Gallo to Licensing Board, September 9, 1982). In view of this, the genuine issue of fact should be revised to read: "Is the concrete in the fuel pool strong enough to resist a temperature of 150°F and point loading from the storage racks?"

B-3. On September 30, 1983, Licensee submitted the testimony of Dr. Howard J. Eckert and Dr. Madarapalli K. Prabakhara. Both witnesses are structural engineers employed by NUS Corporation. Their joint testimony presents a structural analysis, performed by them, of the Big Rock Point concrete pool under dead, hydrostatic and thermal loadings, which assumes that the bulk pool temperature will not exceed 150°F. (Eckert and Prabakhara, ff. Tr. 4058.) In response to a Board question regarding the applicability of the NUS analysis to the pool structure as built some 20 years ago, Licensee submitted the testimony of Professor Mete A. Sozen, a professor of structural engineering with special expertise in reinforced concrete structures and a member of the American Concrete Institute Building Code Committee, who concluded that the analysis applied to the as-built structure. (Sozen, ff. Tr. 5137.) On this issue, Licensee also submitted the testimony of Jerome D. Lescoe, Licensee's construction superintendent during construction of the Big Rock Point Plant. (Lescoe, ff. Tr. 5131.)

B-4. The NRC Staff submitted the testimony of Mark A. Caruso and Drew Persinko, who reviewed the Licensee's analysis for the Staff and prepared the Staff's Supplemental Safety Evaluation Report (SSER) on the pool concrete. Their testimony concluded that the Licensee's analysis adequately assured the integrity of the concrete structure under the assumed accident conditions. (Caruso, ff. Tr. 3979; Persinko, ff. Tr. 4169.) The Staff also agreed that the analysis was applicable to the structure as built. (Persinko, Tr. 5178-83.) The Intervenors submitted no testimony on this issue.

B-5. The Big Rock Point spent fuel pool is a rectangular reinforced concrete structure with its cavity sheathed in a 3/16-inch stainless steel liner. The walls vary in thickness from 2 feet to 6 feet 9 inches. On three sides the pool structure is supported by walls below the pool walls. On the fourth side the pool wall is supported by a shear key, a reinforced concrete member which protrudes from the reactor cavity concrete. (Eckert and Prabakhara, ff. Tr. 4058, at 6-7.)

B-6. As the pool water heats up relative to the surrounding air under the postulated accident conditions, thermal loads, in the form of temperature gradients, are imposed on the walls and floor of the pool structure. Concrete expands as it is heated. Here the inner surfaces of the pool walls and floor will heat first and therefore will tend to expand more than the cooler outer portions. Because the walls and floors are connected, they cannot independently bend to accommodate this growth, and internal forces are created in the concrete. These forces, termed shear forces and bending moments, resist the tendency of sections of the concrete to shear (i.e., slide relative to one another) and to bend (*Id.* at 4.)

B-7. Drs. Eckert and Prabakhara initially determined the thermal loads caused by the assumed accident conditions. They also calculated the loads imposed by the weight of the structure itself and its contents. Using these loads and a mathematical model of the structure, they calculated parameters, such as moment and shear, which portray the structure's behavior. To determine the adequacy of the structure, they then compared these shear forces and bending moments to the strength capacities of the concrete and the adequacy of the steel reinforcing bars embedded in it. (*Id.* at 5.)

B-8. The witnesses assumed a water heatup rate of approximately 1°F per hour from the operating temperature of 101°F to a maximum bulk temperature of 150°F. Because the stainless steel pool liner expands faster than the concrete, they also considered the load this differential thermal expansion would impose. In addition, they considered the hydrostatic pressure applied to the walls and floor by the pool water and the deadweight loading of the water, the racks, the fuel, the floor slab and miscellaneous equipment. In determining the strength capacities of the support walls they also considered the weight of the pool walls. (*Id.* at 7-8.)

B-9. Drs. Eckert and Prabakhara performed a finite element analysis, idealizing the structure as an assemblage of discrete blocks, for each of which shear forces and bending moments were calculated. Because the inner surfaces of the walls and floor will tend to expand more, the inner portions of the structure will be in compression while its outer portions are in tension. (*Id.* at 9.)

B-10. Concrete is relatively weak in tension, hence the need for steel reinforcing bars. When the tensile stress becomes great enough a crack is formed, and as load increases the crack progresses, thereby reducing the flexural stiffness of the section, and affecting the distribution of load as load application continues. To reflect this behavior the witnesses performed a nonlinear analysis, increasing the load in increments, after each of which the stiffness was reduced, until the maximum gradients were reached. (*Id.* at 10.)

B-11. This procedure is approximate because it assumes that the maximum gradients for each wall and the floor occur at the same time. In reality, because of the differing thickness of these elements, they would heat at different rates. Because the NRC Staff questioned whether this method of applying load was conservative, and because of an error in the application of the computer code, Drs. Eckert and Prabakhara reperformed the analysis applying the maximum gradients at the time they actually occurred and correcting the error in computer code application. The NRC Staff also questioned the ability of the structure to resist forces generated by differential expansion of the steel liner and pool concrete since this was omitted in the January 10, 1983 analysis. Therefore, the witnesses also performed a study of the effect of differential thermal expansion of the stainless steel liner on the pool concrete in conjunction with the reanalysis of the January 10, 1983 submittal. (*Id.* at 10; Persinko, ff. Tr. 4169, at 3.)

B-12. The witnesses calculated strength capacities at various cross-sections of the structure in accordance with the ACI Code. The capacities are a function of the yield strength of the steel reinforcing bar, the compressive strength of the concrete and the dimensions of the section. (Eckert and Prabakhara, ff. Tr. 4058, at 11.)

B-13. The Code indicates that the strength properties of concrete are not degraded at a temperature of 150°F, and it allows temperatures of up to 200°F in local areas. (*Id.* at 5.)

B-14. The Code also specifies required development lengths for the reinforcing bar, i.e., the length of embedment necessary to assure that the bar can be stressed to the yield point. Splicing of the bars is normally accomplished by overlapping, and required lap splice lengths are also specified by the ACI Code. The analysis showed that in one location a lap splice was not sufficient to meet the Code criterion. A recalculation which used information from a paper published in an ACI Journal in 1977, and which took into account the strength provided by the 6 inches of concrete covering the splice, showed that this splice was adequate. (*Id.* at 12-13.)

B-15. To compare the strength capacity of the structure to the calculated forces, the witnesses computed ratios of the shear and moment capacities to the calculated values of shear forces and bending moments. They also computed ratios of the length of the reinforcing bars and overlaps to those required to develop the moment capacities. Values of these ratios, or margins, greater than 1 indicate excess capacity, or strength. (*Id.* at 13.)

B-16. (Deleted.)

B-17. The thermal-hydraulic analysis of the Big Rock Point spent fuel pool performed by JAYCOR and discussed in § E of this opinion showed a localized area in one corner of the pool in which the pool water temperature reaches 152°F, i.e., 2.7°F greater than the bulk pool temperature. Such a localized temperature is acceptable with respect to concrete strength properties, since the ACI Code allows temperatures of up to 200°F locally. Also, the strength margins at this location are sufficient to accommodate the effects of this small localized increase in pool water temperature. (*Id.* at 14.)

B-18. Assuming a minimal gap between the pool and the liner, which is reasonable, all average shear, local shear and development length (with the inclusion of the test data) factors of safety are greater than 1 for the pool floor and walls. The moment margin for the pool floor also exceeds 1. In one location in one wall, the moment margin was less than 1. Exceeding the allowable moment locally is acceptable provided the surrounding material can carry the additional load and no collapse mechanism develops. The witnesses examined the margins surrounding this region where moment capacity is exceeded and concluded that the surrounding material is capable of carrying the additional load and that no collapse mechanism will occur. The support walls have margins greater than unity with respect to all applicable parameters when one factors in load redistribution in the walls. (*Id.* at 15; Persinko, ff. Tr. 4169, at 3.)

B-19. In addition, in response to an NRC Staff question, the Licensee analyzed the shear key located on the west wall and determined that it was adequate to support all calculated loads. (Eckert and Prabakhara, ff. Tr. 4058, at 16.)

B-20. Drs. Eckert and Prabakhara subsequently reduced the margins calculated by Licensee when the Staff pointed out that the weight of the wall over the shear key had not been included in the calculation. This analysis showed that some of the local shear margins were less than 1. The witnesses found this to be of no significance, however, because the north and south support walls act in parallel with the shear key, carrying the loads simultaneously, and are more than sufficient to carry the load. (Tr. 4700, 4126-27.)

B-21. Drs. Eckert and Prabakhara also considered the weight of the 120,000-pound shipping cask, the heaviest object that can be set in the spent fuel pool. The effect of this additional weight would be to further reduce the margin for the shear key. However, because the support walls could take the entire load, the shear key is not needed at all to support the fuel pool structure. (Tr. 4141-44.) The witnesses also evaluated the load imposed by the cask on the corner of the pool floor where it would rest. They concluded that the margins were more than adequate to withstand this local pressure. (Tr. 4148-49.)

B-22. With regard to point loading from the storage racks, Drs. Eckert and Prabakhara reviewed and adopted the analysis contained in the Licensee's Consolidated Application. The analysis considered bearing stress, resulting from the weight of the rack and fuel applied through the rack leg, and punching shear stress, the local loading condition under the rack leg which could punch a hole through the pool floor. The analysis determined that margins were greater than 1 in all instances. (Eckert and Prabakhara, ff. Tr. 4058, at 16-17.)

B-23. In addition, Mr. Gary Pratt of Consumers Power Company performed an analysis showing that when the containment atmosphere temperature rises rapidly during a LOCA, so that the outside of the pool structure is heated more than the inside, the loads imposed on the structure are less severe than those analyzed in detail by Drs. Eckert and Prabakhara. (*Id.* at 8; Pratt, Tr. 5192.)

B-24. Mr. Pratt testified that the cooling of the pool walls by containment sprays would not cause significant additional stresses. Three of the pool walls are shielded from the sprays. Moreover, the containment sprays are located high above the pool and the nozzles put out a very fine spray. By the time the spray reaches the pool walls, it will have absorbed all the heat it is capable of and will be at the ambient temperature. The temperature profiles used in the NUS analysis therefore remain valid. (Tr. 5186-88.)

B-25. On the basis of all these analyses, Drs. Eckert and Prabakhara concluded that the spent fuel pool structure is adequate to resist the effects of a temperature of 150°F and point loading from the storage racks. (Eckert and Prabakhara, ff. Tr. 4058, at 16-17.)

B-26. Mr. Persinko engaged in a particularly thorough review of the structural analysis, and requested Drs. Eckert and Prabakhara to perform several reanalyses to assure him of the accuracy of certain details in the modeling. (*Id.* at 6, 10-11, 16.) The SSER, incorporated by reference in Mr. Persinko's testimony, is extremely detailed. (Persinko, ff. Tr. 4169, at 6.) Based on his review, Mr. Persinko concluded that the spent fuel pool structure is adequate to withstand the increased load resulting from the proposed pool expansion for pool water temperatures up to 150°F. (*Id.* at 1.)

B-27. Mark A. Caruso testified for the Staff that the thermal analysis methods used by NUS to calculate temperature distributions in the concrete were appropriate, based on the uniformity of pool water temperatures shown in the JAYCOR thermal-hydraulic analysis. He also testified that the calculated temperature distributions appeared reasonable. (Caruso, ff. Tr. 3979, at 5.)

B-28. Professor Sozen testified that the acceptance criteria used by NUS could validly be applied to Big Rock Point. The criteria assumed by NUS related mainly to flexural, shear and bond strengths and were derived by the procedures specified in the current ACI Concrete Code and the associated ACI Building Code. The Big Rock Point Plant was built according to the 1958 Uniform Building Code, whose provisions pertaining to reinforced concrete were based on the 1951 ACI Building Code. (Sozen, ff. Tr. 5137 (the first of two pages numbered 5137), at 4.)

B-29. Most of the fundamental criteria contained in the current ACI Building Code, ACI 318-77 (1977), are essentially the same as those in the 1951 version. Although the relevant strength criteria contained in the earlier version of the Code may be somewhat less conservative than those of the current Code, the minimum compressive strength ever measured at Big Rock Point was 3025 psi, which compares favorably to the 3000 psi compressive strength assumed by NUS. Moreover, the design requirements on which strength capacities are based have not changed substantially since the 1951 version, that is, although the 1951 version was based on working stress design and the current methods on ultimate strength design, essentially the same sizes and sections would result for a given load. (*Id.* at 4-5; Tr. 5155.)

B-30. Under the criterion used by NUS to determine flexural strength, or the structure's capacity to tolerate bending moments, the strength of a lightly reinforced structure is relatively insensitive to variations in the compressive strength of the concrete; it depends rather on the strength of the reinforcing steel. The current allowances for steel strength have remained unchanged for over 20 years. (Sozen, ff. Tr. 5137, at 5-6.)

B-31. Shear strength is calculated under the current Building Code by a method different from that of the Code under which Big Rock Point was built. The current shear strength limit used by NUS, however, is conservative compared with what would have been used at the time of design. (*Id.* at 6-7.)

B-32. The current Code's criterion for bond strength, which controls the required development length of the rebar, is not directly comparable to that of the earlier Code because of calculational and design changes. Particular comparisons, however, show that the current requirement used by NUS is more conservative. (*Id.* at 7.)

B-33. Professor Sozen also examined the analysis of the shear key and concluded that it was extremely conservative in that actual strength of the shear key would be 3 or 4 times what was assumed. He also concluded that there will be adequate support of the pool along the west edge. (Tr. 5148, 5150-51.)

B-34. The basic parameters of the NUS structural analysis are the pool dimensions, the concrete strength and the amount, arrangement and strength of the steel reinforcement. The current ACI Building Code provides for the application of such an analysis to an existing structure. In such a case, the Code requires a thorough field investigation of dimensions, properties of materials and other pertinent conditions. Such an investigation has been undertaken at Big Rock Point. (Sozen, ff. Tr. 5137, at 7.)

B-35. Dr. Eckert took actual measurements of the spent fuel pool walls and found them to conform to the values indicated in the structural drawings. (*Id.* at 8.)

B-36. NUS assumed a concrete compressive strength of 3000 psi. (Eckert, Tr. 4077-78.) Recorded compression tests of cylinders made from the Big Rock Point spent fuel pool concrete indicated a mean compressive strength of 3686 psi, and no cylinder was below 3000 psi. The cylinder tests also indicated excellent quality control. (*Id.* at 8-9.)

B-37. In addition, recorded slump readings and the fact that there was no congestion of reinforcement indicate no likelihood of critical voids within the concrete. (*Id.* at 9.)

B-38. Professor Sozen's field investigation verified that the appearance of the concrete does not suggest defects in the casting process. Had large voids existed around groups of reinforcing bars, unusual surface cracks would most probably have appeared after over 20 years of use. In addition, there is no visible indication which would suggest a serious omission of reinforcement occurred. (*Id.* at 9-10.)

B-39. The Code procedures for evaluation of existing structures contain no requirement for precise information on amount and arrangement of the reinforcement. which is assumed to conform to the structural drawings. (*Id.* at 10.)

B-40. For the Big Rock Point pool, an existing construction photograph, introduced into evidence, shows the reinforcing bars for the floor slab before the concrete was poured. The detail is sufficient to show that placement and spacing of the bars conform to the structural drawings and suggest that the job was well controlled. (Licensee Exhibit 27, ff. Tr. 5122; Sozen, Tr. 5137-40.)

B-41. Professor Sozen concluded that the limiting strength criteria used in the NUS analysis are correct for and applicable to the spent fuel pool structure for several reasons. They are based on accepted engineering principles consistent with current professional practice. Furthermore, they are comparable to if not more conservative than those used at the time of construction. Finally, the available information about the pool as built is adequate to substantiate these strength criteria. (Sozen, ff. Tr. 5137, at 11.)

B-42. Jerome D. Lescoe, as Licensee's construction superintendent for the Big Rock Point Plant, was responsible for Licensee's overview of the performance of Bechtel Corporation, the engineer-constructor of the facility. Mr. Lescoe was knowledgeable in good construction practices for pouring reinforced concrete structures and he observed concrete pours on a daily basis at Big Rock Point, including pours for the spent fuel pool. (Lescoe, ff. Tr. 5131, at 2; Lescoe, Tr. 5172-74.)

B-43. Mr. Lescoe observed that Bechtel followed their drawings and specifications and used appropriate methods to form and place concrete. Before a pour was made, the general foreman and an engineer saw that rebar placement complied with drawings and that the area was free from rust or debris. During the pour, they used techniques to keep the concrete from separating and complied with good practice in the use of vibrators to eliminate voids. (Lescoe, ff. Tr. 5131, at 2-3.)

B-44. The photograph admitted in evidence showing construction of the pool was taken under Mr. Lescoe's supervision and he confirmed that placement of the rebar in the photo conforms to the structural drawings. (Tr. 5130-34.)

B-45. Mr. Lescoe observed that the concrete cylinders used for the compression tests were kept in the immediate area of the pour so they could cure under the same conditions. (Tr. 5147.)

B-46. Mr. Persinko read the testimony of Professor Sozen and Mr. Lescoe and examined the concrete test records and the construction photo. Nothing presented by Mr. Lescoe or Professor Sozen in their oral testimony or in their written testimony caused him to change his conclusions in the SSER or testimony. After his review of the concrete test records and construction photo, he independently reached the same conclusion as Professor Sozen, namely, that assumptions in the Licensee's analysis appear to be applicable to the as-built structure. Mr. Persinko also physically inspected the pool structure and did not notice any visible defects. (Tr. 5178-83.)

B-47. The Board finds that the Licensee's analysis assures the adequacy of the pool structure under the assumed accident conditions. The Board finds that the Licensee's analysis validly applies to the actual pool structure as built.

C. O'Neill Contention II.E-4 — Shielding

C-1. O'Neill Contention II.E-4 states:

> In the event of an accident which results in a substantial release of radioactivity from the expanded fuel pool, the containment building does not provide adequate shielding to protect the public health and safety.

C-2. Licensee presented the testimony of its Director of Radiological Services Department, Mr. Roger Sinderman. (Sinderman, ff. Tr. 4250.) NRC Staff presented the testimony of Mr. Millard Wohl, a Nuclear Engineer in the Radiological Analysis Section of the Accident Evaluation Branch of the Office of Nuclear Reactor Regulation of the NRC. (Wohl, ff. Tr. 4137.) Mr. Wohl's testimony included consideration of a scenario that assumed a partial release of the radioactive inventory from the containment. The Licensing Board ruled this scenario irrelevant and further ruled that that portion of Mr. Wohl's testimony should be considered struck. (Tr. 4310-17.) Intervenors withdrew their testimony on this contention. (Tr. 4362-63.)

C-3. The accident postulated by the witnesses for purposes of analyzing the shielding capability of the containment is the drop of a spent fuel bundle onto fully loaded spent fuel racks. (Sinderman, ff. Tr. 4250, at 4; Wohl, ff. Tr. 4317, at 2.)

C-4. The drop of a spent fuel bundle is the maximum credible accident, i.e., the design basis accident that would result in the largest release of radioactivity from the Big Rock Point Plant spent fuel pool. (Wohl, ff. Tr. 4317, at 2; Sinderman, ff. Tr. 4250, at 3-4; Sinderman, Tr. 4253.)

C-5. Licensee has demonstrated in this proceeding that a drop of the 24-ton transfer cask into the spent fuel pool is an incredible event. Based on its review, the NRC Staff has concluded that safety slings will prevent the cask from dropping. The Licensing Board's decision on

O'Neill Contention II.C insofar as that contention concerns the possibility of a cask drop has accepted Licensee's and the NRC Staff's position, based on their evidence presented on that issue. (*See* our decision on O'Neill Contention II.C, *above*; Wohl, ff. Tr. 4317, at 2; Sinderman, ff. Tr. 4250, at 3; *see also* "Joint Testimony of Fred Clemenson, Ian Sargent, D.J. Vito, and Richard L. Emch, Jr., Concerning O'Neill Contention II.C," ff. Tr. 2434, and the testimonies of John W. Johnson, Charles R. Norman, John J. Popa, and A. Davis Mulholland, Jr., ff. Tr. 2419 (June 12, 1982).)

C-6. The Final Hazards Summary Report for the Big Rock Point Plant, which was prepared in conjunction at the operating license stage, considered the offsite radiation dose consequences of a core melt accident and showed the consequences to be within the limits of 10 C.F.R. Part 100, on the order of 100 millirem. A core melt accident is more severe than the pool accident considered here. (Sinderman, Tr. 4256-58.)

C-7. Mr. Wohl and Mr. Sinderman assumed that the radioactive inventory of the fuel gap of each pin of the dropped fuel bundle is released to the spent fuel pool and the containment. Mr. Sinderman explained further that the fuel is assumed to have been in the reactor for 3 years, to have operated at the highest peaking factor, and to have been removed from the core 72 hours after plant shutdown. (Wohl, ff. Tr. 4317, at 4; Sinderman, ff. Tr. 4250, at 4.)

C-8. Mr. Sinderman calculated the radiation dose which would be released in the scenario just discussed. At several offsite locations, taking credit for the attenuation provided by the ¾-inch steel containment and the atmosphere between the containment and those locations, the dose to an individual over a 2-hour period at the given locations was calculated to be as follows: 0.0035 millirem at the nearest overland site boundary (2640 feet); less than a microrem at the nearest residence (5280 feet); 0.0029 millirem at the nearest approach of a public highway (2760 feet); and 58 millirem at the nearest shoreline approach. (Sinderman, ff. Tr. 4250, at 6.)

C-9. Mr. Wohl calculated the exposure to a person at the site boundary (800 meters) over a 2-hour period to be 0.2 millirem. His calculation accounted for attenuation from the atmosphere as well as the ¾-inch steel containment building. (Wohl, ff. Tr. 4317, at 3, as corrected at Tr. 4308; Wohl, Tr. 4331.)

C-10. The reasons for difference between Mr. Sinderman's calculation for the dose at the site boundary and Mr. Wohl's calculation were adequately explained by those witnesses and showed both calculations to be reliable. Mr. Wohl's calculation was somewhat less precise since it relied on an estimate of average gamma-ray releases from different elements. (Wohl and Sinderman, Tr. 4436-38.)

C-11. The Board finds that the shielding of the Big Rock Point containment building provides adequate shielding to protect the public health and safety in the event of an accident which results in a substantial release of radioactivity from the spent fuel pool.

D. O'Neill Contention II.D — Risks from Aircraft

D-1. O'Neill Contention II.D states:

> The licensee has not adequately provided for the protection of the public against the increased release of radioactivity from the expanded fuel pool as a result of the breach of containment due to the crash of a B-52 bomber.

D-2. On Motion for Summary Disposition, Licensee presented a probability analysis prepared by Major (now Lt. Col.) Gary Betourne and Mr. Clayton Thomas of the United States Air Force. In our Memorandum and Order (Concerning Motions for Summary Disposition) dated February 19, 1982, LBP-82-8, *supra*, we posed eleven issues concerning the validity of the probability analysis, thus narrowing the focus of the contention with respect to B-52 aircraft. *Id.*, 15 NRC at 327-29.

D-3. As stated in our Memorandum and Order of February 19, 1982, *supra*, we interpreted O'Neill Contention II.D to include genuine issues of fact concerning the safety of Big Rock Point from aircraft used by the Ohio Air National Guard and from small unscheduled airplanes. *Id.* at 330. Evidence was also heard regarding the risks from military aircraft using VR-1634, a low-level route in the vicinity of Big Rock Point.

D-4. Licensee presented five witnesses who addressed O'Neill Contention II.D. Lt. Col. Gary Betourne is experienced as a navigator of B-52 aircraft (Betourne, ff. Tr. 4464) and was formerly with the Air Force Studies and Analysis Division of the USAF, during which time he prepared the probability analysis that was presented in his deposition dated July 13, 1981 (Consumers Power Company Exhibit 2C). Major John V. Lyczkowski of the Ohio Air National Guard and Captain William Hickey of the Arizona Air National Guard (formerly of the Ohio Air National Guard) addressed the contention insofar as it concerns Ohio Air National Guard flights (Hickey and Lyczkowski, ff. Tr. 4369); Anthony Tome, an engineer with Wood-Leaver and Associates, Inc. (Tome, ff. Tr. 4582), and Robert M. Marusich (Marusich, ff. Tr. 4582), an engineer with Licensee's Radiological Services Department, presented analyses of the risks posed by small unscheduled aircraft.

D-5. The NRC Staff presented one witness, Dr. Kazimieras M. Campe, a Senior Site Analyst in the Siting Analysis Branch of the NRC, who addressed all aspects of the expanded O'Neill Contention II.D. (Campe, ff. Tr. 4655.)

D-6. Intervenors John O'Neill and Christa-Maria testified on their own behalf concerning flights near Big Rock Point witnessed by them. (O'Neill, ff. Tr. 4740; Christa-Maria, ff. Tr. 4744.)

D-7. Lt. Col. Betourne's 1980 analysis estimated the probability of a crash of a B-52 on the Bayshore training route at Big Rock Point to be less than 10^{-8} per year. (Consumers Power Company Exhibit 20 (Deposition of Maj. Gary Betourne and Mr. Clayton Thomas, taken July 13, 1981), admitted Tr. 4464, exclusive of Mr. Thomas' answers.)

D-8. The Big Rock Point Plant is no longer useful as a radar offset aiming point of reference since air crews using the Bayshore training run no longer receive detailed range and bearing information about the power plant. This reduces the likelihood that an air crew will mistakenly believe the plant to be a "direct mode" aiming point and fly directly over the plant. Photographs of the radar scope taken during flight and examined thereafter discourage air crews from developing personal radar offset aiming points since they would reveal the use of an illegal offset. (Betourne, ff. Tr. 4464, at 1-3.)

D-9. Although the crash of a B-52 on the Bayshore training route in 1971 was relevant to Lt. Col. Betourne's probability analysis, he arbitrarily excluded it from his data by selecting a sample that excluded this event. He stated that the crash data of interest for the 1980 analysis began with 1972 and thus did not include consideration of the 1971 data. (*Id.* at 3-4.)

D-10. The 2-month sample used by Lt. Col. Betourne in his 1980 analysis has been proven to be conservative inasmuch as the number of gross navigational errors extrapolated from that time period (sixty) were less than actually occurred for the year of interest (thirty-six). (*Id.* at 4-7.)

D-11. The total number of runs used in Lt. Col. Betourne's analysis accounted for the "unscored" USAF activity at the Bayshore Range. (*Id.* at 9-10.)

D-12. Lt. Col. Betourne's 1980 analysis assumes a 3-nautical-mile square of the route around the plant, i.e., the area in which the crash of a B-52 is assumed to damage the containment. This assumption is conservative in that it assumes that debris always would have sufficient kinetic energy to damage the containment when in reality the probability would be less than unity. (*Id.* at 10-13.)

D-13. There is no basis for assuming that low-altitude flight of B-52s is more hazardous than other B-52 flight activity. Low-altitude training crashes represent only about 6% of all B-52 crashes. (*Id.* at 13-14.)

D-14. If it is assumed in Lt. Col. Betourne's analysis that there is a failure of radio communication, due to any reason, with 30% of the aircraft which exceed the corridor of the route, the resultant estimated risk probability would be acceptable in view of the other conservatism assumed, such as the crash area. (*Id.* at 14-16.)

D-15. The Air Force did not correspond with any insurers regarding risk computation. Nor was there any risk assessment performed by the USAF prior to moving the Bayshore route to its present location. Any risk presented by the old route has been virtually eliminated by the route change and the prohibition of the containment facility as an offset target aiming point. The new route is completely over Lake Michigan and thus approaching the shoreline can be readily recognized as a navigation error. (*Id.* at 17.)

D-16. (Deleted.)

D-17. The "no-fly" zone for the Bayshore route would mean any area outside of the corridor of the route. Aircraft on the Bayshore route are continuously monitored by the radar tracking station at Bayshore and notified by radio if the route corridor is exceeded. However, there is no radio communication with aircraft in unscored flight activity. The corridor of the Bayshore route has recently been reduced by a nautical mile on either side of center to assure tighter navigational control. (*Id.* at 7, 18-19.)

D-18. The July 22, 1981 flight of two A-7 jets of the Ohio Air National Guard aircraft was led by Captain Hickey who testified that at no time did they overfly the Big Rock Point Plant at low altitude. (Hickey and Lyczkowski, ff. Tr. 4369, at 5.)

D-19. Lay persons often misestimate, to some extent, the range at which they see an A-7D aircraft because they are larger than most single-engine, single-seat aircraft, they generate a lot of noise, and the high speed of the aircraft gives the aircraft a higher line of sidetracking angle. (Hickey, Tr. 4390.)

D-20. Units which have requested activation of Wolverine Military Operations Area ("MOA"), which encompasses the Big Rock Point Plant, are advised of designated no-fly areas around the plant. The no-fly areas prohibit direct overflight of the plant below 5000 feet above sea level (about 4500 feet above the ground level) and prohibit flight below 1500 feet above ground level within 2 miles of the plant when the Wolverine MOA is activated. (Hickey and Lyczkowski, ff. Tr. 4369, at 6-10.)

D-21. The Military Training Routes VR-1634 and VR-1636 pass Big Rock Point at 5.2 and 33.4 miles respectively at their closest points. Units which schedule the use of these routes are advised of the no-fly areas regarding Big Rock Point in the Wolverine Military Operations area. (*Id.* at 10-11.)

D-22. Captain Hickey performed an analysis of probability of an accident at the Big Rock Point Plant given one low-altitude overflight of the plant per year by an A-7 aircraft of the Ohio Air National Guard. His analysis showed this probability to be 1×10^{-8}. (*Id.* at 11-14.)

D-23. The Wolverine MOA is activated only in daytime and flight is permitted only when prevailing weather conditions allow at least 5 miles of visibility. (*Id.* at 12.)

D-24. The military training routes are used only in conditions that permit 5 miles of visibility and a cloud ceiling no lower than 3000 feet. (*Id.* at 11; Hickey, Tr. 4416.)

D-25. Mr. Anthony E. Tome, Jr., performed an analysis of the probability of the crash of an unscheduled general aviation flight into the containment of the Big Rock Point Plant. His analysis concluded that the probability of such risk was 1.33×10^{-6}. (Tome, ff. Tr. 4582, at 13.)

D-26. Mr. Tome's estimate of risk is extremely conservative for the following reasons: it assumes all flights originating in the area of interest flew in the direction of the plant; the upper bound of log density function (the mode) was used to estimate the number of flights in the Big Rock area; the growth factors assumed have been shown to be greatly overestimated; and the crash density was maximized by assuming a minimal glide angle and altitude. (*Id.* at 13-14.)

D-27. The extremely conservative nature of Mr. Tome's analysis is reflected by the fact that it assumes an annual overflight of the plant by more than 54,000 planes, which breaks down to 1 overflight every 10 minutes. (Tome, Tr. 4614, 4643.)

D-28. Dr. Kazimieras M. Campe of the NRC's Siting Analysis Branch addressed all aspects of civilian and military aviation activities in the vicinity of the Big Rock Point Plant. Aircraft hazards to Big Rock Point were reviewed by the NRC Staff within the Systematic Evaluation Program (SEP) for Big Rock Point. Topic II-1.C of the SEP Safety Assessment for Big Rock Point addressed the hazards to Big Rock Point from the nearby B-52 low-level training route and general aviation activities from nearby airports. Not specifically mentioned in the SEP safety assessment were the activities of military aircraft in the Wolverine MOA and the low-level military training route VR-1634, the flights of small unscheduled aircraft in the vicinity of Big Rock Point, and commercial aviation near Big Rock Point. (Campe, ff. Tr. 4655, at 2-4.)

D-29. The NRC Staff SEP included a review of the 1980 Air Force probability analysis. The Staff's review concluded that the Air Force analysis was reasonable and provided an adequate basis for the B-52 crash risk estimated. The NRC Staff concluded that the probability was well within the acceptance criteria of SRP § 2.2.3. Further, the NRC Staff stated that the route change would reduce the probability of a crash of a B-52 at Big Rock Point to an even lower level. (*Id.* at 4-9, and Campe Reference 1, at 7.)

D-30. Dr. Campe identified several significant conservatisms in the 1980 USAF estimate. First, it assumes that all navigational errors occur in the direction of the Big Rock Point Plant. It is assumed that all navigational errors will remain uncorrected. Further, Air Force data indicate that pilots are increasingly less likely to stray larger distance from routes. The expected frequency-of-deviation errors decrease exponentially with the size of the deviation. Another conservatism is the assumption of a 3-nautical-mile-square area of crash in which the plant would be vulnerable. The effective plant impact area is actually no more than about 0.16 square nautical mile. If the realistic estimates are used in the 1980 USAF estimate, the annual probability of a crash onto the plant is much less than 10^{-8} per year. (Campe, ff. Tr. 4655, at 6-9.)

D-31. Dr. Campe's analysis of the probability of a military aircraft crash at Big Rock Point, in connection with the Wolverine MOA, estimated an upper-bound probability of 7×10^{-7} (i.e., on the order of 10^{-6}), while a more realistic analysis, based on reasonable qualitative judgments, estimated the probability to be 7.6×10^{-9} (i.e., on the order of 10^{-8}). (*Id.* at 9-14.)

D-32. Dr. Campe's upper-bound Wolverine MOA analysis used a projected maximum of 600 flights involving 1500 aircraft per year using the Wolverine MOA. The realistic estimate was based on figures reflecting actual usage which was shown in 1 year to be forty flights involving ninety-nine aircraft. Also, the upper-bound estimate assumed uniform flight distribution within the Wolverine MOA. This is conservative in that low-altitude flights around Big Rock Point are expected to be rare due to the no-fly restrictions that apply in the area of the plant. Further, the upper-bound estimate assumed a conservatively large effective plant impact area. The realistic estimate used an area based on the actual plant area together with the skid and shadow areas. (*Id.* at 12-13.)

D-33. Dr. Campe's upper-bound and realistic risk estimates for air activity associated with military training route VR-1634 show the probability to be 5.7×10^{-7} and 2.5×10^{-9}, respectively. (*Id.* at 15, revised by Campe Supplemental Testimony, ff. Tr. 4655, at 3-4.)

D-34. The upper-bound estimate for the military training route VR-1634 assumed an overflight by every flight using the route. The realistic estimate used the reasonably conservative assumption of one overflight per year from the route. (Campe, ff. Tr. 4655, at 14-15, revised by Campe Supplemental Testimony, ff. Tr. 4655, at 3-4.)

D-35. The probability of a general aviation crash onto the Big Rock Point Plant was estimated by the NRC Staff to be about 8.7×10^{-7} per year. By multiplying this value by the projected annual airport operations at the Charlevoix Airport, Staff estimated the risk to the plant from general aviation using the airport to be about 8.5×10^{-4} crash per year. Dr. Campe testified that a major conservatism in this estimate is the assumption that all of the 71,000 flights per year projected for the Charlevoix Airport resulted in an overflight of the plant. (Campe, ff. Tr. 4655, at 15-17; Campe, Tr. 4708-13, 4725-26.)

D-36. For purposes of the Standard Review Plan, a proper analysis would add together the probabilities of crash for the various types of aircraft operations. Adding together the analyses that were performed concerning the types of aircraft at issue in this contention, the cumulative probability would be acceptably small. Risks posed by other types of aircraft are so remote that their contribution to the overall risk would be insignificant. (Campe, Tr. 4690-92, 4722; Campe, ff. Tr. 4655, at 17-18.)

D-37. Dr. Campe concluded that the probability of an aircraft crash at the Big Rock Point Plant is sufficiently low that aircraft impacts need not be considered as a design basis event. (Campe, ff. Tr. 4655, at 18; Campe, Tr. 4733-34.)

D-38. Dr. Arthur J. Schwartz testified as to the need to use "common sense" and to incorporate all relevant, available experimental data when preparing probabilistic risk assessments. (Deposition, November 16, 1983, at 7-8, 46, 48.) Some of the criticisms raised by Dr. Schwartz were addressed by Lt. Col. Betourne in his testimony. (Betourne, ff. Tr. 4464, at 10, 14-15; Tr. 4462-72, 4510-13.)

E. O'Neill Contention II.C: Seismic Stability of Overhead Crane

E-1. O'Neill Contention II.C states:

> Is the spent fuel pool safe from a rupture which might be caused by a drop of a spent fuel transfer or of the overhead crane?

E-2. In its Memorandum and Order of February 19, 1982, LBP-82-8, *supra,* the Board determined that a genuine issue of fact existed as to whether the overhead crane used for handling fuel assemblies and casks is seismically safe.

E-3. Licensee presented the testimony of seven witnesses. Mr. David J. VandeWalle, employed by Licensee as Nuclear Licensing Administrator, described the applicable seismic design criteria for the Big Rock Point Plant and explained the genesis of the earthquake peak ground acceleration and ground response spectrum used by Licensee to evaluate the seismic stability of the reactor building overhead crane. Dr. Andrew J. Eggenberger, a Project Manager for D'Appolonia Consulting Engineers, Inc., testified about the floor response spectra utilized in the structural analyses of the overhead crane and its support structures. Messrs. Charles R. Norman, Manager of Engineering Services for Whiting Corporation, Robert D. Campbell, a Project Manager for Structural Mechanics Associates, Inc., and Yat F. Chan, Senior Engineer for the Civil Structural Section of Licensee's Plant Modification and Miscellaneous Department, explained respectively, their structural analyses of the overhead crane, the crane rail anchorages, and the crane support structure and crane stops. Mr. Steven B. Beachum, an Associate Engineer in Licensee's Technical Department at the Big Rock Point Plant, reported the status of certain modifications to the overhead crane which have been identified as necessary to assure the crane's seismic stability. Messrs. VandeWalle, Norman, Campbell, Chan and Beachum and Dr. Eggenberger appeared as a panel. In addition, Mr. Peter I. Yanev, an engineer and President of EQE Incorporated, described the results of his research on the performance of cranes similar to the overhead crane under seismic loadings much stronger than the ground motion postulated for Big Rock Point.

E-4. The NRC Staff presented the testimony of three witnesses. Drs. Thomas M. Cheng, Nilesh C. Chokshi, and Leon Reiter, appeared as a panel. Dr. Reiter, Leader of the Seismology Section of Geosciences Branch, Division of Engineering, of the Office of Nuclear Reactor Regulation, explained the site-specific spectra developed by the NRC Staff for the Big Rock Point site. Drs. Cheng and Chokshi, Senior Structural and Structural Engineer, respectively with the NRC Staff, testified concerning the structural adequacy of the overhead crane to withstand seismic loadings.

E-5. Intervenors presented no testimony on this contention, relying instead on cross-examination of Licensee and Staff witnesses.

E-6. The overhead crane is located inside the reactor building and is a modified gantry crane rated for 75 tons. The crane is supported by and travels east-west along two railroad-like rails. On the south side of the reactor building, the gantry legs have been replaced with a bridge truck arrangement. When operating in the vicinity of the spent fuel pool, the southern end of the crane is supported by the condenser deck and a steel support structure. On the northern end, the crane's gantry leg is supported by the fuel pool deck. (Norman, ff. Tr. 4784, at 5-7; Chan, ff. Tr. 4784, at 3.) The crane rails are anchored to the condenser and fuel pool decks by means of clips and single or double bolts. (Campbell, ff. Tr. 4784, at 3.) The overhead crane also has a single-rail, 5-ton monorail hoist suspended from the crane's west bridge box girder. (Norman, ff. Tr. 4784, at 7.)

E-7. The Big Rock Point nuclear plant was designed and constructed from 1959 through 1962. (VandeWalle, ff. Tr. 4784, at 3.)

E-8. The plant structures were designed in accordance with the 1958 edition of the Uniform Building Code. A horizontal force 0.025*g* static was used for all major structures except for the reactor containment vessel. A seismic factor of 0.05*g* static was used for the design of the reactor containment vessel. (*Id.* at 3; VandeWalle, Tr. 4860.)

E-9. In December 1977, the NRC initiated the Systematic Evaluation Program (SEP) to, among other things, assess the safety of older plants, including Big Rock Point, which had been built prior to current NRC safety regulations and criteria. (VandeWalle, ff. Tr. 4784, at 4; Reiter, ff. Tr. 4902, at 2.)

E-10. The seismic design criteria for nuclear power plants have changed significantly since the construction of the SEP plants. The SEP includes a reevaluation of plant seismic design criteria. (VandeWalle, ff. Tr. 4784, at 4-5; Reiter, ff. Tr. 4902, at 2.)

E-11. The NRC Staff determined that an alternative methodology to that set forth in Appendix A to 10 C.F.R. Part 100 was needed to determine the appropriate earthquake for the seismic evaluations to be performed under SEP. The Staff felt a methodology was needed to make a realistic determination of the appropriate earthquake based upon the true seismic hazard that was not a function of changing seismic design criteria or criteria that resulted in regional bias. (Reiter, ff. Tr. 4902, at 2; VandeWalle, ff. Tr. 4784, at 5.)

E-12. In 1978, the NRC Staff undertook a program with Lawrence Livermore Laboratory, and its subcontractor, TERA Corporation, to develop peak ground acceleration values and site-specific spectra for the SEP plants, including Big Rock Point. (VandeWalle, ff. Tr. 4784, at 5; Reiter, ff. Tr. 4902, at 2; Reiter, Tr. 4952-53.) This work was projected to take 3 to 4 years. (VandeWalle, ff. Tr. 4784, at 5.)

E-13. In January 1979, the NRC Staff initiated the review of the structural capability of the Big Rock Point Plant to withstand earthquakes. (*Id.* at 4-5.) The NRC Staff instructed Licensee to develop interim seismic design criteria to be used in the seismic evaluation until Lawrence Livermore Laboratory's efforts were completed and the site-specific spectrum and peak ground acceleration for the Big Rock Point Plant had been established. (*Id.* at 5-6; Reiter, Tr. 4952-53.)

E-14. The interim seismic input selected by Licensee was an earthquake with a peak ground acceleration of $0.12g$ and the ground response spectrum recommended by NRC Regulatory Guide 1.60. The NRC response spectrum recommended in NRC Regulatory Guide 1.60 anchored to $0.12g$ was chosen based, in part, upon the expectation that the interim design criteria would bound the site-specific spectra being developed in the SEP. (VandeWalle, ff. Tr. 4784, at 6-7; Reiter, Tr. 4952-53.)

E-15. Lawrence Livermore Laboratory completed its assessment of the seismic risk for the SEP plants in mid-1980. The results are set forth as uniform hazard spectra, where each spectral amplitude has the same probability of being exceeded. The spectra are intended to represent the equivalent hazard from site to site, which will be of the same order of magnitude as the hazard implicitly associated with the choice of safe shutdown earthquakes using deterministic criteria. The spectrum for a particular site is built point by point by making predictions for each frequency. All potential earthquakes contributing to the seismicity at the site are considered using appropriate seismicity, attenuation and exposure models. The spectral acceleration versus frequency is plotted and the loading corresponding to a particular return period is used as the appropriate spectral amplitude at a given frequency. (VandeWalle, ff. Tr. 4784, at 7; Reiter, ff. Tr. 4902, at 2-3; NUREG/CR-1582, Vol. 1, Licensee Exhibit 24, at 3-6.)

E-16. The study performed by Lawrence Livermore Laboratory included the solicitation of expert opinion on key seismic input parameters, including seismic zonation, frequency of earthquake occurrences, upper magnitude cutoff, and characterization and attenuation of ground motion. The analysis of seismic hazard for the eastern United States was extremely difficult due to the low level of seismic activity and lack of records. Uncertainty concerning input parameters was taken into account in each experts' distribution of earthquake probability. The final results of each expert were integrated into a single hazard curve by means of weights supplied by each expert. (Reiter, ff. Tr. 4902, at 2-3; Reiter, Tr. 4955-56, 4971-74; NUREG/CR-1582, Vol. 1, Licensee Exhibit 24, at 4.)

E-17. An extensive comparison was made with deterministic criteria to assure that the probabilistic spectra were within the appropriate range dictated by deterministic considerations. Minimum deterministic levels for each site were also chosen to assure consideration of a moderate size earthquake irrespective of the earthquake's estimated occurrence. (Reiter, ff. Tr. 4902, at 3-4.)

E-18. The site-specific spectra for the Big Rock Point site were approved by the NRC Staff and provided to Licensee in June 1981. The integrated site-specific spectra developed by Lawrence Livermore for Big Rock Point were anchored at 0.08*g*. The NRC Staff, pursuant to its policy of setting minimum deterministic levels for each site, raised the site-specific ground acceleration to 0.105*g*. (Reiter, Tr. 4985-88; VandeWalle, ff. Tr. 4784, at 7.)

E-19. The interim seismic design criterion, the Regulatory Guide 1.60 response spectrum anchored at 0.12*g*, bounds the site-specific spectrum for the Big Rock Point site at all frequencies. (VandeWalle, ff. Tr. 4784, at 7; Reiter, ff. Tr. 4902, at 4.)

E-20. The site-specific response spectra for the Big Rock Point site did not include site-specific factors for amplification due to shallow soil conditions. It is a difficult problem to deal with. There is a great difference of opinion as to how the analyses should be done. The amount of amplification at shallow soil sites has varied significantly, and may be affected by phenomena such as focusing and radiation. (Reiter, ff. Tr. 4902, Attachment 1, at 1-4; Reiter, Tr. 4995, 4977-99, 5084.)

E-21. Evaluation of the potential for amplification is a matter of judgment. Dr. Reiter evaluated the possibilities of taking into account generalized statistical studies, site-specific theoretical studies, detailed comparisons between available rock and soil records, and other seismological factors; he concluded that, in deriving the spectra for the SEP, enough conservatism had been employed to account for the amplification present at the Big Rock Point site. (Reiter, ff. Tr. 4902, at 4 and Attachment 1; Reiter, Tr. 5001-14, 5078-89.)

E-22. The very low seismic hazard in Northern Michigan indicates that the chance there will be earthquake ground motion of any significance at Big Rock Point is extremely small. (Reiter, ff. Tr. 4902, at 4-5, and Attachment 1, at 11.)

E-23. Earthquake ground motions introduce vibratory motions to the base of structures, which in turn induce vibrations throughout the entire structures. The characteristics of the vibratory motions at different levels or floors depend on the dynamic characteristics of the structures, and are represented in floor response spectra. The floor response spectra can be utilized for the structural analysis of equipment such as the overhead crane. (Eggenberger, ff. Tr. 4784, at 3; Eggenberger, Tr. 4787-88.)

E-24. D'Appolonia Consulting Engineers performed a seismic analysis on behalf of Licensee which, among other things, generated floor response spectra at various elevations of the reactor building, including the support locations of the overhead crane. These floor response spectra were utilized as seismic input for the structural analyses of the overhead crane. (Eggenberger, ff. Tr. 4784, at 2, 4 and Attachment 1 (Vol. II, Appendices A and B, Seismic Safety Margin Evaluation, Reactor Building, Primary Coolant Loop, Big Rock Point Nuclear Power Plant, Charlevoix, Michigan, Rev. 1, dated September 1981); Cheng and Chokshi, ff. Tr. 4092, at 6; Chokshi, Tr. 4945; Norman, Tr. 4792.)

E-25. Floor response spectra for the support locations of the overhead crane were generated for the postulated earthquake defined in accordance with the ground response spectra recommended in NRC Regulatory Guide 1.60, with a zero-period, peak ground acceleration of 0.12*g*, previously referred to as the interim criterion, and the SEP site-specific response spectra. (Eggenberger, ff. Tr. 4784, at 4 and Attachment 1 (Vol. II, Appendices A and B, Seismic Safety Margin Evaluation, Reactor Building, Primary Coolant Loop, Big Rock Point Nuclear Power Plant, Charlevoix, Michigan, Rev. 1, dated September 1981) and Attachment 2 (Derivation of Site-Specific Seismic Floor Response Spectra, Seismic Safety Margin Evaluation, Big Rock Point Nuclear Power Plant, Charlevoix, Michigan, dated August 1983); Cheng and Chokshi, ff. Tr. 4902, at 5.)

E-26. When Regulatory Guide 1.60 is used to compute the site-specific floor response spectra, in the frequency range which affects the overhead crane, the acceleration are found to be responses approximately 50% of the floor response spectra. (Cheng and Chokshi, ff. Tr. 4902, at 10; Campbell, ff. Tr. 4784, at 5.)

E-27. The floor response spectra were calculated using a seismic analysis which models the reactor building using three-dimensional beam elements. These elements have both translational and rotational degrees of freedom. The mass of the structure was represented through 6-degree-of-freedom, lumped-mass elements providing all 6 degrees of freedom at each node. (Eggenberger, ff. Tr. 4784, at 4, 12.)

E-28. The seismic analysis performed using the interim criterion included an evaluation of the induced stresses in the reactor building and containment shell structures under combined seismic and dead loads, and the results show that the margins of safety for the reactor building and containment shell structure exceed 1. This conclusion will be confirmed when the SEP seismic review is completed by the NRC Staff. (Eggenberger, ff. Tr. 4784, at 4; Cheng and Chokshi, ff. Tr. 4902, at 7.)

E-29. Licensee has not evaluated the induced stresses on the reactor building and containment shell structure under the seismic loadings attributable to the site-specific response spectra. Since the site-specific response spectra are bounded by the Regulatory Guide spectra anchored at 0.12*g*, Licensee and NRC Staff have concluded that the stresses induced by the site-specific earthquake would be less than previously analyzed. (Eggenberger, ff. Tr. 4784, at 5; Cheng and Chokshi, ff. Tr. 4902, at 7.)

E-30. The seismic analyses performed by D'Appolonia relied on the recommendations of NUREG/CR-0098 and the practices published in Regulatory Guides and Standard Review Plans, particularly SRP § 3.7.2, Seismic System Analysis, June 1975, which was revised and reissued as SRP, § 3.7.2, Seismic System Analysis, Rev. 1, July 1981. (Eggenberger, ff. Tr. 4784, at 5-26; Cheng and Chokshi, ff. Tr. 4902, at 12.)

E-31. The seismic analyses used the time-history method to conduct dynamic analysis. (Eggenberger, ff. Tr. 4784, at 7, 9-10.)

E-32. The hydrodynamic loads generated by the spent fuel pool water were conservatively accounted for by removing the springs in the spring-mass system representing the sloshing of water, and distributing the horizontal mass of the water among adjacent nodes while lumping the total vertical mass of the water at a node corresponding to the elevation of the bottom of the spent fuel pool. (Eggenberger, ff. Tr. 4784, at 16.)

E-33. Water from the local bay has no effect on the seismic analyses, except to the extent bay water might impact on the water table at the Big Rock Point site. D'Appolonia accounted for this possibility. (Eggenberger, Tr. 4798-4800; Cheng, Tr. 4936.)

E-34. The seismic modeling of the reactor building accounted for the possibility of nonlinear responses due to the presence of Fesco boards along the expansion joints that isolate the reactor cavity structure and the horizontal shear key. The analysis was first performed using a single-stick model of the reactor building which neglected the presence of expansion joints and treated the reactor cavity, spent fuel pool, and steam drum enclosure as monolithic. A second, multi-stick analysis was performed in response to Staff questions, which inquired about the complete absence of interaction at the expansion joints. (Eggenberger, ff. Tr. 4784, at 17; Cheng and Chokshi, ff. Tr. 4902, at 11.) The results of the two analyses were not dissimilar, with the single-stick model generating the more conservative input for the evaluation of the overhead crane. (Eggenberger, ff. Tr. 4784, at 17; Cheng and Chokshi, ff. Tr. 4902, at 11; Eggenberger, Tr. 4796-97; Cheng, Tr. 4917-18, 4922-25.)

E-35. The shear key will not cause any hinging, and any binding due to the shear key would increase damping through the dissipation of energy and make the floor response curves more conservative for overhead crane evaluation. (Cheng, Tr. 4922-25.)

E-36. Torsional effects were considered explicitly in the seismic analyses by incorporating all significant known mass eccentricities. The spent fuel pool and steam drum enclosure were modeled as eccentric masses at their own centroidal locations, due to their significant distance from the center of mass of the rest of the reactor building. This treatment led to coupling between horizontal and torsional responses. (Eggenberger, ff. Tr. 4784, at 23.)

E-37. D'Appolonia analyzed the soil-structure interaction using the half-space (lumped-parameter) method. (*Id.* at 18-21.) Under this method, static spring constants and damping values are calculated using classical half-space solutions. The static spring constants are then corrected for frequency and embedment effects using classical solutions. (*Id.* at 20.)

E-39. The six spring constants and damping values (three translational and three rotational) were combined with the analytical model of the reactor building structure. (*Id.* at 20-21.)

E-40. Several soil spring/soil structure analyses have been performed. Varying the soil springs did not have a significant effect on the results of the analyses. The properties of the soil around the Big Rock Point Plant have an insignificant effect on the seismic analysis of the plant for the frequencies important to the structural analysis of the overhead crane. (Cheng and Chokshi, ff. Tr. 4902, at 12; Chokshi, Tr. 4942.)

E-41. The methodology used by D'Appolonia to generate the floor response spectra was judged to be conservative by a factor of 20-50%. (Eggenberger, Tr. 4790-92.)

E-42. Licensee assessed the adequacy of the overhead crane by using three types of structural analyses: (1) crane analysis, (2) rail analysis, (3) crane support knee brace and end stops analysis. (Chan, ff. Tr. 4784, at 2.)

E-43. Whiting Corporation utilized the floor response spectra generated by D'Appolonia using the ground response spectra recommended in Regulatory Guide 1.60 to perform structural analyses of the overhead crane. Response spectra curves for the elevation of the condenser deck crane rail support were provided for three directions, north-south, east-west, and vertical. Static loads due to gravity were considered as part of applied loads. (Norman, ff. Tr. 4784, at 1, 7-8 and Attachment 1 (Gantry Crane Seismic Report); Chokshi, Tr. 4945.)

E-44. Whiting Corporation performed several structural analyses in order to apply the seismic loading to a variety of crane operating conditions. One analysis considered the crane unloaded and the other considered it loaded with 24 tons, the crane's maximum operating load over the spent fuel pool. The analyses also evaluated the seismic loadings of the crane when the trolley was positioned at three distinct locations on the runway, and with the crane being positioned over the spent fuel pool and at the east end of its travel. All combinations of these loadings and positions were analyzed. The loads used in Whiting Corporation's analysis of the crane's capability to withstand stresses induced by seismic motion are those associated with the crane positioned in locations including the maximum seismically induced loadings. (Norman, ff. Tr. 4784, at 8; Cheng and Chokshi, ff. Tr. 4902, at 8; Norman, Tr. 4794, 4852-53.)

E-45. The structural analyses of the overhead crane assessed the implications of the 5-ton monorail hoist fully loaded. (Norman, Tr. 4816-17.)

E-46. The structural analyses of the crane evaluated the condition of the crane and trolley being pushed against their respective runway and stops. (Norman, ff. Tr. 4784, at 8; Cheng and Chokshi, ff. Tr. 4902, at 8.)

E-47. The analyses performed by Whiting Corporation used the finite-element method, which models the crane as an assemblage of many discrete beams. Using the structure's design and material properties, and the relationship between stress and strain, mathematical expressions are formulated for each finite beam, or element, such that equilibrium of forces and displacements between elements are ensured at each node. The mathematical expressions are solved to determine the forces and moments throughout the crane. From these forces and moments stresses are calculated and compared to the crane materials' capacities to withstand stress. (Norman, ff. Tr. 4784, at 8-11, 13; Cheng and Chokshi, ff. Tr. 4902, at 8; Norman, Tr. 4870.)

E-48. ANSYS, a large-scale, general-purpose finite-element computer program, was used to perform the Whiting analyses. The dynamic analysis performed by ANSYS is of the mode frequency (MODAL) type, in which the computer solves for the shape and amplitude of the vibration of the structure due to seismically induced motion. (Norman, ff. Tr. 4784, at 9.)

E-49. Modes with meaningful participation are evaluated by the computer for element stress and strain. The standard mode approach, as outlined in Standard Review Plan § 3.7.2.II.7, was utilized. (Norman, ff. Tr. 4784, at 9-10.) Components not included in the mathematical representation of the crane, such as bolts and welds, were analyzed independently of the ANSYS program using the moments and forces generated by the ANSYS analysis. (*Id.* at 10.)

E-50. Whiting Corporation did not perform a detailed analysis of the crane trolley since it was designed for 75 tons and carries a maximum operating load over the spent fuel pool of 24 tons. (Norman, ff. Tr. 4789, at 16.)

E-51. The crane materials' strength capacities are as specified in the American Institute of Steel Construction ("AISC") Code, as modified by NRC Standard Review Plan § 3.8.3, for assessing the structural adequacy of steel under seismic loadings. (Norman, ff. Tr. 4784, at 4, 11; Norman, Tr. 4809-11.)

E-52. Allowable strength capacities for tension, per AISC 1.5 and SRP § 3.8.3, are 96% of the crane components' yield strength. Allowable capacities for shear are 60% of the components' strength capacity for tension or 58% of the components' yield strength. (Norman, ff. Tr. 4784, at 11-12.)

E-53. Strength capacity for shear was calculated in accordance with Whiting Corporation's standard procedures. Whiting Corporation's standards were used because the AISC Code does not address irregularly shaped components and such phenomena as local buckling. (*Id.* at 12.)

E-54. With the expansion bolts, Whiting Corporation's standard for allowable shear is more conservative than the AISC guidelines. (*Id.*)

E-55. The allowable capacities for bolts as calculated using Whiting Corporation's standards are within the materials' yield strength in shear. (*Id.*)

E-56. Whiting Corporation's standards for local buckling are in compliance with the standards on local buckling recently promulgated by the Crane Manufacturer's Association of America. (Norman, Tr. 4813.)

E-57. The primary frequency of the overhead crane lies between 2.0 and 2.7 Hz. (Cheng and Chokshi, ff. Tr. 4902, at 9.)

E-58. The structural analyses performed by Whiting Corporation establish that the maximum stresses induced by seismically induced loadings associated with the ground response spectra recommended in Regulatory Guide 1.60 anchored at 0.12*g* are generally low and well within established allowables. (Norman, ff. Tr. 4784, at 13-14, and Tables 1 and 2.)

E-59. The maximum stress on the gantry leg exceeds allowables by approximately 3%. The stress calculated to be in excess of allowable is localized and limited to one of the four corners of the gantry leg's cross-section, and does not exceed the material's yield strength. (Norman, ff. Tr. 4784, at 14; Cheng and Chokshi, ff. Tr. 4902, at 9; Norman, Tr. 4822-23; Chokshi, Tr. 4946.)

E-60. The maximum stresses on the A307 bolt connections between the knee braces and the bridge box girders exceed the allowable strength of the A307 bolts. (Norman, ff. Tr. 3784, at 14; Cheng and Chokshi, ff. Tr. 4902, at 9.) Licensee has therefore replaced the A307 bolts previously used to connect the knee braces to the bridge box girders with high-strength A325 bolts, whose allowables are not exceeded by the maximum stresses induced by the seismic loadings. (Beachum, ff. Tr. 4784, at 3; Norman, ff. Tr. 4784, at 14; Cheng and Chokshi, ff. Tr. 4902, at 9.)

E-61. The Whiting Corporation's analyses established that the 5-ton monorail hoist could not withstand the postulated seismic loading. Accordingly, Licensee has stiffened the monorail track by welding a ¼-inch steel reinforcing plate to the track, strengthened the track's attachment to the west bridge girder by adding eight additional hangers, reinforced welds and replaced bolts in existing hangers, and installed thrust rollers to restrain the underhung trolley from seismically induced sway. With these modifications, the monorail hoist will withstand the postulated seismic loadings. (Beachum, ff. Tr. 4784, at 2-3; Norman, ff. Tr. 4784, at 14-15; Cheng and Chokshi, ff. Tr. 4902, at 10.)

E-62. All critical welds, plates and columns were evaluated and found to have margins of safety in excess of 1. (Norman, ff. Tr. 4784, at 15; Norman, Tr. 3826.)

E-63. All structural members were evaluated for local buckling and found to be within allowables. (Norman, ff. Tr. 4784, at 15.)

E-64. The Whiting Corporation determined that the crane will not jump from its rail supports. (*Id.*; Cheng and Chokshi, ff. Tr. 4902, at 10; Norman, Tr. 4831.)

E-65. The crane's brakes will lock automatically during a seismic incident. Should the crane slide along its rail supports, the seismic forces acting upon the crane will be relieved. (Norman, Tr. 4892-30.) The crane will not slide more than 18 inches under the maximum seismic loadings. (Norman, ff. Tr. 4784, at 16.)

E-66. The seismic effects on the trolley, fully loaded with 24 tons, both static and dynamic, would be 48 tons. This load is less than the original design limits of the overhead crane. (*Id.* at 16.)

E-67. The steel support structure is comprised of three steel members: the crane support girder to which the crane rail is attached, the horizontal strut, and the vertical strut. (Chan, ff. Tr. 4784, at 3.)

E-68. Licensee utilized the maximum wheel loads to assess the structural capability of the steel support structure. The wheel loads were statically imposed on the steel support structure along with the crane stop loads (as determined in the analyses performed by Whiting

Corporation), the loads imposed by gravity, and the seismically induced motion of the condenser deck. (*Id.* at 4-5 and Attachment 1 (Report on Structural Analysis for Reactor Building Crane Support Knee Brace and Crane Stops, dated October 7, 1983); Norman, Tr. 4850-52, 4856.)

E-69. The wheel and crane stop loads accounted for the amplification of seismic motion imposed on the crane by the steel support structure. (Chan, Tr. 4853-55; Norman, Tr. 4850-51, 4856.)

E-70. The seismically induced loads attributable to the movement of the condenser deck were determined by establishing the fundamental frequency of the structural members of the steel support structure and using the floor response spectra developed by D'Appolonia using the ground response spectra recommended in Regulatory Guide 1.60 anchored to 0.12*g*. (Chan, ff. Tr. 4784, at 4.)

E-71. The loads were applied to each structural member as uniform loads. The crane stop loads were applied as concentrated loads at the point of contact between the crane and the crane stops. (*Id.* at 5.)

E-72. All moments, shears and axial forces, due to gravity, seismic, and the crane wheel and crane stop loads were combined for each individual member and connections, and translated into stresses. The stresses were then compared to the allowable stresses specified in the AISC Code, as modified by SRP § 3.8.3. Bond stresses for the smooth anchor bolts were compared with allowable stresses specified in the American Concrete Institute's (ACI) Code 318-63. Adequacy of anchorage connections of the crane support to the condenser deck wall was verified using ACI 349-80, as modified by SRP § 3.8.3. (*Id.* at 5.)

E-73. Licensee also evaluated the steel support structure and crane stops using the floor response spectra developed by D'Appolonia using the site-specific spectra. (Chan, Tr. 4883-84; Eggenberger, ff. Tr. 4784, Attachment 2 (Derivation of Site-Specific Seismic Floor Response Spectra, Seismic Safety Margin Evaluation, Big Rock Point Nuclear Power Plant, Charlevoix, Michigan, dated August 1983.)

E-74. Licensee's evaluation demonstrates that the crane stops' and steel support structure's allowable strength capacities exceed the stresses which would be induced by seismic loadings. (Chan, ff. Tr. 4784, at 5-6.)

E-75. The maximum stress on the bolted connection between the crane support girder and horizontal strut exceeded the bolts' allowable stress. Licensee has welded a tee section between the horizontal strut and the crane support girder, which will transmit the horizontal forces and bending moment directly from the crane support girder to the horizontal strut, thereby alleviating the shear stress on the bolted connection. (*Id.* at 6-7; Beachum, ff. Tr. 4784, at 3.)

E-76. The analysis Licensee performed to evaluate the adequacy of the crane stops using the floor response spectra based on the Regulatory Guide 1.60 ground response spectra indicated the seismic loads on the northern stop's tension anchor bolts exceeded the bolts' allowable stress by approximately 44%. (Chan, ff. Tr. 4784, at 6.)

E-77. When loads are generated using the floor response spectra based on the site-specific spectra, the margin of safety for the northern stop's tension anchor bolts exceeds unity. (*Id.* at 6; Cheng and Chokshi, ff. Tr. 4902, at 9-10; Chan, Tr. 4883-84.)

E-78. Review of the results of the Whiting Corporation's analyses led Licensee to decide to strengthen the overhead crane's upper rail anchorage by replacing all thirteen pairs of single-bolt rail clips. This necessitated replacing twelve pairs of 3-inch x 4½-inch x ½-inch A7 single-bolt clips with 6-inch x 4½-inch x ½-inch A514, grade B single-bolt clips and one pair of 3-inch x 3½-inch x ½-inch A7 clips with a pair of 6-inch x 3½-inch x ½-inch A514 grade B clips. This modification was to be completed by mid-November 1983. (Beachum, ff. Tr. 4784, at 4; Beachum, Tr. 4841-43; Campbell, Tr. 4835.)

E-79. Structural Mechanics Associates, Inc. (SMA) performed a structural analysis of the overhead crane's rail anchorages using the site-specific spectra. (Campbell, ff. Tr. 4784, at 2.)

E-80. SMA evaluated the rail anchorages by using the floor response spectra associated with the site-specific and Regulatory Guide 1.60 ground response spectra to scale down the wheel loads reported in the Whiting Corporation's analysis. The rail loads were reduced by comparing the spectral accelerations at a single frequency in each direction. The wheel loads calculated by Whiting Corporation were scaled down by a factor greater than 2. (*Id.* at 5.)

E-81. The governing condition is the maximum load applied statically directly at the point where the crane rail is secured by a single-bolt clip. (*Id.* at 4-5; Campbell, Tr. 4837-38, 4840.)

E-82. The strength of the rail anchorages was analyzed using a simple linear elastic model of the rail and the clip supports. The rail was modeled as a beam with torsional stiffness and the clips were modeled as rotational springs. The MODSAP computer program was used to solve for the stresses. The calculated stresses were then compared with allowables specified in the AISC Code, as modified by Standard Review Plan § 3.8.3. (Campbell, ff. Tr. 4784, at 5-6; Campbell, Tr. 4840.)

E-83. The analyses performed by SMA show that all crane rail anchorages, including the single-bolt clips which Licensee has committed to modify, meet the AISC Code allowables, as modified by Standard

Review Plan § 3.8.3. The lowest margins of safety for single- and double-bolt shears were 1.98 and 3.96, respectively. The margins of safety for single- and double-bolt clip bending were 2.2 and 4.4, respectively. (Campbell, ff. Tr. 4784, at 6; Cheng and Chokshi, ff. Tr. 4902, at 9-10.)

E-84. With Licensee's modifications, the margin of safety for single-bolt clip bending will increase by a factor of 6. (Campbell, ff. Tr. 4784, at 6.)

E-85. (Deleted.)

E-86. Mr. Norman testified that many of the cranes manufactured by Whiting Corporation have experienced seismically induced stresses and he was not aware of any that had failed. (Norman, Tr. 4831-32.)

E-87. Mr. Yanev testified on the results of his company's investigation into the performance of cranes similar to the overhead crane under seismic loadings throughout the world. (Yanev, Tr. 3599-3743.)

E-88. Licensee requested Mr. Yanev's company, EQE, Inc., to conduct its investigation to determine whether the results of the structural analyses of the overhead crane are consistent with actual experience. (Yanev, ff. Tr. 3598, at 6.)

E-89. The investigation conducted by EQE, Inc., reviewed the performance of cranes whose overall design and configuration characteristics taken together envelope the critical characteristics of the Big Rock Point overhead crane. The cranes reviewed included single-leg and double-leg gantries and bridge cranes. The peak acceleration of the estimated earthquake ground motions reported upon in the crane survey were equal to or exceeded 0.12*g* which Licensee used as an interim criterion. (*Id.* at 5-6, 11.)

E-90. The data reviewed during the EQE investigation included the type and design characteristic of the surveyed cranes, the type and size of the crane's supporting structures, the peak ground motions, earthquake acceleration time histories and related spectra, site intensities, and the data necessary to assure that the critical parameters of the Big Rock Point overhead crane were enveloped by the data on the surveyed cranes. EQE evaluated Code criteria to which the surveyed cranes were built. Generally, the cranes were either built to the AISC Code or comparable criteria. (*Id.* at 8-9; Yanev, Tr. 3733, 3737-39.)

E-91. Mr. Yanev testified that the buildings housing the surveyed cranes were subjected to seismic motions that were estimated to be equal to or greater than the ground motion postulated in the interim criterion for the Big Rock Point site. Dr. Reiter testified that caution should be observed in utilizing the specific ground motion estimates, since most of Mr. Yanev's estimates were based upon extrapolation techniques which have not een laid out in detail and for which the uncertainty has not been sufficiently emphasized. Richter magnitudes for these seismic events varied from 5.5 to 7.4. (Yanev, ff. Tr. 3598, at 9; Reiter, Tr. 4906B.)

E-92. The surveyed cranes were housed by structures more flexible, equally stiff, and stiffer than the Big Rock Point reactor building. The support height for the Big Rock Point overhead crane is typical for the cranes surveyed. (Yanev, ff. Tr. 3598, at 10-11.)

E-93. Mr. Yanev's figures indicated that none of the approximately thirty cranes investigated were damaged during estimated peak ground accelerations of less than $0.35g$ or about 3 times the $0.12g$ assumed in the interim criterion. (*Id.* at 12; Reiter, Tr. 4906-08.)

E-94. The Pasadena and the Humbolt Bay crane are both single-leg gantry cranes very similar to the Big Rock Point overhead crane. The highest seismically induced stresses in the overhead crane would be caused by the tendency of the crane to twist due to this single-leg configuration. (Yanev, ff. Tr. 3598, at 14-15; *see* Licensee Exhibit 15, Figures 3.19 and 6.4, at 28, 53.)

E-95. The Pasadena and Humbolt Bay single-leg gantry cranes experienced estimated ground accelerations of $0.20g$ and larger, with site horizontal response spectra, at the primary resonant frequency of 2.0 to 3.0 hz, bounding the Regulatory Guide 1.60 ground response spectra assumed for the Big Rock Point site, and suffered no damage. (Yanev, ff. Tr. 3598, at 14-15; Licensee Exhibit 15, Figures 3.18 and 6.1, at 28, 50.)

E-96. None of the cranes surveyed suffered buckling of the gantry legs. (Yanev, ff. Tr. 3598, at 15.)

E-97. Several of the surveyed cranes had rail anchorages similar in size and design to the anchorages for the Big Rock Point overhead crane. None of the surveyed crane anchorages which were properly constructed were damaged by earthquakes. (*Id.*)

E-98. Only three cranes of the approximately thirty cranes surveyed by EQE were damaged during earthquakes. These incidents were unique to factors particular to these cranes. (*Id.* at 16.)

E-99. The Burbank crane at the Burbank/Magnolia power plant had several lower crane leg to lower truck bolts broken when it experienced an earthquake with an estimated peak ground acceleration of $0.35g$. The crane did not fail. The damage was due to the cantilevered design of the crane, which was less stable, and therefore experienced more torsional stress, than would the Big Rock Point overhead crane. (*Id.*; Licensee Exhibit 15, Figures 3.12 and 3.14, at 24-25; Yanev, Tr. 36, 38, 39, 3717.)

E-100. The overhead crane's monorail hoist will not create the cantilevered effect which resulted in damage to the Burbank crane. (Yanev, Tr. 3719-22.) [Change (*see* Tr. 3683).]

E-101. The rail anchors of the Pleasant Valley Pumping Station were damaged during the recent Coalinga earthquake because of poor welding of the bolts to the supporting steel girder and the design of the supporting rail over a building expansion joint. The Pleasant Valley crane experienced ground acceleration of 0.54g, and did not fail despite damage to the rail anchorage. (Yanev, ff. Tr. 3598, at 16; Yanev, Tr. 3681-84.)

E-102. The small bridge crane in the ENALAF power plant during the 1972 Manaqua earthquake collapsed when it experienced ground motion estimated by Mr. Yanev at 0.70g. However, the highest nearby ground acceleration actually recorded was 0.4g. The crane apparently collapsed because the building housing the crane flexed excessively, causing larger than allowable relative deformation between the two crane support rails. (Yanev, ff. Tr. 3598, at 17; Yanev, Tr. 3651.)

E-103. In all other cases, surveyed cranes subjected to ground accelerations up to 0.70g were undamaged and remained in service. (Yanev, ff. Tr. 3598, at 17.)

E-104. The method Mr. Yanev used for collecting data is primarily journalistic in nature, in that he was generally collecting slides of earthquake damage without special concern for cranes. (Tr. 3667.) Consequently, his data tend to be incomplete (Tr. 3678, 3651-52 and 57-58, 3639, 3660, and 3665-66.) Under the circumstances, we accept Mr. Yanev's testimony that he has not discovered evidence contradictory to the engineering analysis of the crane, but we place little weight on his conclusion that cranes are less susceptible to earthquakes than other structures.

CONCLUSIONS OF LAW

A-1. The reliability of the makeup water system has been established based on the single-failure criterion of 10 C.F.R. Part 50, Appendix A and sound engineering judgment.

A-2. There is no realistic possibility that zircaloy fuel cladding will react with steam in the spent fuel pool.

A-3. There is no credible potential for the occurrence of a criticality accident in association with a loss-of-coolant accident.

A-4. In accordance with 10 C.F.R. Part 50, Appendix A, General Design Criterion 61, Licensee's fuel storage and handling systems have been designed to provide adequate safety under normal and postulated accident conditions; in particular, they have been designed to prevent any significant reduction in fuel storage coolant inventory under postulated accident conditions.

A-5. The technical specifications of the Big Rock Point Plant are modified in accordance with Licensee's "Proposed Technical Specifications Change," dated October 25, 1983.

B-1. Licensee's analysis admitted into evidence assures the adequacy of the pool structure under the accident conditions assumed in the contention.

B-2. Licensee's analysis validly applies to the actual pool structure as built.

B-3. The spent fuel pool structure complies with applicable Staff guidance and the relevant industry codes incorporated therein by reference.

B-4. The spent fuel pool structure complies with the relevant portion of the General Design Criteria of Appendix A to 10 C.F.R. Part 50, which, though not strictly applicable to Big Rock Point, are helpful in guiding the Board's decision.

C-1. The Licensing Board concludes that radiation exposure of the public will not exceed 10 C.F.R. Part 100 limits in the event of an accident that causes a substantial release of radioactivity into containment from the spent fuel pool.

C-2. The Licensing Board concludes that the maximum credible accident in the spent fuel pool that could lead to a substantial release of radioactivity into the pool and the containment is the drop of a spent fuel assembly in the pool.

D-1. The probability of occurrence of an aircraft crash at Big Rock Point leading to potential consequences in excess of 10 C.F.R. Part 100 limits is sufficiently low within the guidelines of Standard Review Plan § 2.2.3 that such events need not be considered in the design of the plant.

E-1. The Board finds that the site-specific spectra anchored at 0.105*g* developed for the Systematic Evaluation Program for the Big Rock Point site are the appropriate seismic motion for evaluating the seismic structural adequacy of the overhead crane.

E-2. The record demonstrates that, with the modifications to the overhead crane, the overhead crane will not deform permanently, become unstable, nor will any affixed components become dislodged and fall due to either the site-specific spectra or the Regulatory Guide 1.60 ground response spectra anchored at 0.12*g* used by Licensee as the interim criterion during the development of the site-specific spectra.

E-3. The record demonstrates that, with the modifications Licensee has made to the steel support structure, the steel support structure and crane stops are structurally sound and able to withstand the loadings and stresses associated with the site-specific spectra anchored at 0.105*g*.

E-4. The record demonstrates that the crane rail anchorages are structurally sound and adequate to withstand the loadings associated with the site-specific spectra anchored at 0.105*g*.

E-5. The Board concludes that there is reasonable assurance that the overhead crane is designed to withstand the effects of an earthquake without loss of capability to perform its function, consistent with 10 C.F.R. Part 50, Appendix A, Criterion 2.

E-6. The Board concludes that, with respect to earthquake-induced loadings, there is reasonable assurance that the crane can be operated without endangering the health and safety of the public, consistent with 10 C.F.R. § 50.57(a)(3).

F. Christa-Maria Contention 2 and O'Neill Contention II.A — South Wall

F-1. Christa-Maria Contention 2 states:

> The increase in fuel stored in the Big Rock pool will result in an increase in the amount of radiation released to the environment at the south wall of the storage pool where there is less shielding, according to the Licensee's Description and Safety Analysis. This increment in the level of radiation released to the environment enhances the risks to the health and safety of the public in the vicinity of the plant.

O'Neill Contention II.A states:

> The routine releases of radioactivity during the installation of new racks, the loading of those racks, and storage of fuel in the racks will exceed the exposure of workers, as will the releases of radioactivity through the south wall of the pool exceed the limits imposed by Appendix I to 10 C.F.R. Part 50 on exposure to the general public.

F-2. Based on the affidavits presented by the NRC Staff and Licensee in support of their motions for summary disposition, the Licensing Board found the contentions to raise eight genuine issues of fact. (*See* Memorandum and Order (Concerning Motions for Summary Disposition), dated February 19, 1982, LBP-82-8, *supra,* 15 NRC at 321-22.)

F-3. Licensee presented three witnesses on these issues: Mr. Roger Sinderman, Director of Licensee's Radiological Services Department (Sinderman, ff. Tr. 5023); Mr. Charles Axtell, who until June 1983 was the Plant Health Physicist for Big Rock (Axtell, ff. Tr. 5025); and Mr. Edward Benz, an engineer with NUS Corporation (Benz, ff. Tr. 5021.) The NRC Staff presented one witness, Mr. Seymour Block, a senior Health Physicist with the NRC. (Block, ff. Tr. 5028.) The NRC Staff witness and Licensee's witnesses appeared as a panel. Intervenors withdrew their testimony on this contention (with the exception of one exhibit, which was marked as Consumers Power Company Exhibit 21.) (Tr. 5121-22.)[20]

F-4. The NRC Staff's statements (made in affidavits in support of motion for summary disposition) regarding the thickness of the south wall in their radiation dose calculations were in error. The wall is correctly 3 feet 6 inches thick at its thinnest part. The correct thickness was assumed by Licensee's witness in his calculations of the radiation dose. (Block, ff. Tr. 5047, at 2; Axtell, ff. Tr. 5025, at 2; Affidavit of William Bell, dated September 29, 1981, at 4.)

F-5. Licensee has committed not to store spent fuel at the thinnest portion of the south wall. At that location, Licensee intends to keep channel racks that cannot receive spent fuel assemblies. The thinnest section of the south wall at which Licensee does intend to store spent fuel is 4 feet 5 inches thick. The outer three rows of the rack located there will be used to store spent fuel that has decayed at least 1 year. The radiation dose from the south wall is thus calculated to be about 2.7 mrem/hr. The radiation dose from the spent fuel pool filter sock tank located near the south wall is 30-40 mrem/hr. Thus, the combined radiation dose from the pool and filter sock tank will be between 32.7 and 42.7 mrem/hr. (Axtell, ff. Tr. 5047, at 2-5; Benz, ff. Tr. 5021, at 6; Affidavit of Charles Axtell, dated October 2, 1981, at 8-9.)

F-6. In the calculation of the dose estimates that was presented in support of Licensee's motion for summary disposition, the reference point of the south wall used was the point where the wall is 3 feet 6 inches thick. In the more recent calculations presented with the testimony at the hearings, the point where the wall is 4 feet 5 inches thick was used as the reference point. (Affidavit of William Bell, dated September 29, 1981, at 4; Benz, ff. Tr. 5021, at 3-6.)

F-7. In his affidavit filed in support of Licensee's motion for summary disposition, Mr. Sinderman misstated that he used "mass absorption coefficients" in radiation estimates. He actually used linear absorption coefficients and thus his calculations were correct. (Sinderman, ff. Tr. 5023, at 1-2.)

F-8. The location and reference level to which the NRC Staff applied the inverse square rule to calculate the offsite dose from the south wall was based on calculations performed by Mr. William Bell of NUS

[20] Intervenors filed proposed findings of fact based on extra-record evidence. Accordingly, these findings (which also suffer from a lack of specificity) are not considered by the Board in this portion of its decision. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Unit 2), ALAB-254, 8 AEC 1184 (1975). Further, the proposed findings were not relevant to the shielding characteristics of the south wall.

Corporation. The NRC Staff used distance from coordinate 0.0.0 to center of fuel assembly which is equal to about 4.1 feet. This distance and 2900 feet to the site boundary were used to determine offsite dose by the inverse square rule. (Block, ff. Tr. 5028, at 5; Affidavit of William Bell, dated September 29, 1981, Bell Exhibit 1, at 9.)

F-9. Licensee will take several measures to reduce the radiation level above the spent fuel pool prior to the pool modification, including: filtering pool water through spent fuel pool filter, cycling pool water through the radwaste demineralizer, minimizing movements which would stir up crud from the pool floor, vacuuming in the pool to remove crud, and decontamination of areas near the spent fuel pool. (Affidavit of Charles Axtell, dated October 2, 1981, at 14.)

F-10. The total estimated man-rem dose for the spent fuel rack addition is about 18.2 man-rem. (Axtell Affidavit, dated October 2, 1981, at 16; Licensee's Exhibit 25, Attachment M.)

F-11. The Big Rock Point Plant has averaged 290 man-rem per year over the last 6 years. Based on the estimated 18.2 man-rem, the spent fuel pool modification will deliver only about 6% of a yearly average dose to workers at Big Rock Point. (Sinderman, ff. Tr. 5023, at 2-3; Sinderman, Tr. 5094, 5096, 5118.)

F-12. Licensee is almost certain that it will perform the pool modification with plant personnel. However, if temporary workers are employed, it is anticipated that only a small number will be needed. (Sinderman, Tr. 5055; Axtell, ff. Tr. 5025, at 7, 9.)

F-13. Licensee has had considerable experience in the hiring, training, and supervising of plant workers in the 21-year history of the Big Rock Point Plant. (Axtell, ff. Tr. 5025, at 7.)

F-14. All visitors and contractors who may enter restricted areas receive a 6-hour training course which includes coverage of radiation protection, respiratory protection, nuclear plant industrial safety, fire protection, and site alarms and responses. (*Id.* at 8.)

F-15. The Health Physics Department provides training in respiratory protection mask fitting for workers who may need such protection. (*Id.*)

F-16. Maintenance Department employees have received about 40 hours of intensive training in radiation protection and receive further training at monthly safety meetings. (*Id.* at 9.)

F-17. Chemistry and Radiation Protection Technicians, who provide radiation protection and monitoring services receive extensive training in numerous radiation topics, including ALARA, in a 12-week basic course and an 11-week advanced course. They also receive training under a "Practical Factors" program which assures their ability to perform radiation protection tasks. (*Id.* at 9, 14-19.)

F-18. Licensee has developed procedures that will be followed in the pool modification which will ensure that work will be carried out efficiently and thereby minimize the radiation exposure of personnel. Experienced personnel will participate in the execution of the procedures. Eighty percent of the maintenance department now employed at Big Rock Point participated in a major spent fuel pool task in 1973. Two first-line supervisors respectively have been in the Big Rock Point Maintenance Department for about 21 and 13 years. (*Id.* at 9-10; Axtell, Tr. 5064-65; Licensee Exhibit 25, Attachments E, F, G, H, I; Licensee Exhibit 26.)

F-19. The procedures emphasize the use of proper protective clothing and respiratory equipment. (Axtell, ff. Tr. 5025, at 10.)

F-20. Qualified Health Physics Technicians will continuously supervise workers during the entire pool modification. (*Id.* at 11; Axtell, Tr. 5073-74.)

F-21. All personnel involved in the spent fuel pool modification will be issued a pocket dosimeter, which will be read and recorded at appropriate intervals, and a TLD, which records total accumulated radiation exposure. Workers will also be given a "whole-body count" before and after the work. (Axtell, ff. Tr. 5025, at 11-12.)

F-22. The foregoing finding is not necessary to determine the adequacy of the south wall as a radiation shield, which was the subject of the admitted contention. (*See* note 20, *supra.*)

F-23. Based on his review of the steps being taken to reduce exposure of workers during the pool modification, Mr. Sinderman concludes that the exposure from the operation will be ALARA. (Sinderman, ff. Tr. 5023, at 6.)

F-24. In response to criticisms in the 1981 INPO Report, Licensee has endeavored to provide more thorough instruction in the use of friskers, through lessons and exhibits, and to relocate or shield frisking stations from high-background-radiation areas. (Axtell, ff. Tr. 5025, at 26-29.)

F-25. The radwaste demineralizer is used to cycle spent fuel pool water several weeks prior to shutdown and before personnel spend any significant amount of time over or near the spent fuel pool surface. The radwaste demineralizer is used to process other plant water streams at other times when few man-hours were spent in the area. This is in keeping with the ALARA principle. (*Id.* at 29-30.)

F-26. Cycling pool water through the radwaste demineralizer reduces the dose levels over the pool from approximately 25-30 mrem/hr to 12 mrem/hr. During the spent fuel pool modification, the ALARA goal is to maintain the dose rate at 12 mrem/hr. (*Id.*)

F-27. It would be possible to cycle pool water continuously through the radwaste demineralizer except when needed for some other purpose, thus providing more time for cycling pool water than is presently used. However, constant changeover of the radwaste demineralizer to cycle pool water would expose personnel to relatively high radiation doses in the valving operation which would easily exceed the dose savings from the spent fuel pool. (Axtell, Tr. 5049; Sinderman, Tr. 5049-51.)

CONCLUSIONS OF LAW

F-1. Based on Licensee's commitment to refrain from storing spent fuel along the thinnest portion of the south wall of the spent fuel pool, the Board concludes that reasonable assurance exists that the health and safety of the public and plant workers will not be endangered by the proposed expansion due to radiation emanating through the pool wall.

G. Christa-Maria Subcontention 9(1) — Size of the EPZ

G-1. Christa-Maria subcontention 9(1) states:

> The increased inventory of the fuel pool requires that the emergency plan be based on an inhalation pathway of 10 miles rather than 5 miles and on a 50-mile rather than a 30-mile ingestion pathway.

G-2. To address this subcontention, Licensee submitted the testimony of Roger W. Sinderman, who is employed by Licensee as Director of Radiological Services. (Sinderman, ff. Tr. 2758.)

G-3. The NRC Staff presented the testimony of Monte Phillips, an emergency preparedness analyst and section chief. (Phillips, ff. Tr. 2859, at 2-8; Tr. 2860-2918.)

G-4. The greater the distance of a person from the plant during a release of radiation, the smaller the risk of severe health effects. The radius of the plume exposure pathway EPZ represents a distance beyond which early severe health effects from a spectrum of accidents would not be expected. The outer radius of the ingestion pathway EPZ is based on the minimal potential for significant contamination of food supplies from similar accidents. (NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Plants," EPA-520/1-78-016, at 15-17, Appendix I; NUREG-0654, at 12-13.)

G-5. The NRC Staff has concluded that small water-cooled reactors (less than 250 MWT) may use a plume exposure pathway emergency planning zone of about 5 miles in radius and an ingestion pathway emergency planning zone of about 30 miles in radius. This conclusion is based on the reduced hazard from these smaller facilities. (Phillips, ff. Tr. 2859, at 3-5; Sinderman, ff. Tr. 2758, at 3, Phillips, Tr. 2906; NUREG-0654, at 11.)

G-6. The authorized power level at Big Rock Point is 240 MWT. (Sinderman, ff. Tr. 2758, at 4; Sinderman, Tr. 2766.)

G-7. On April 24, 1980, Licensee submitted an analysis to the NRC Staff justifying a 5-mile plume exposure pathway EPZ for Big Rock Point. (Licensee Exhibit 14, ff. Tr. 2805.)

G-8. Licensee analyzed a postulated containment failure and release of fission products from the core (100% of the noble gas inventory and 25% of the halogens) over a 24-hour period following containment failure. Source terms consistent with Regulatory Guide 1.3 were assumed. Licensee corrected for decay of short-lived noble gases and assumed no decay of radioiodines. This accident approximates one of the worst core melt sequences referenced in NUREG-0654. The results show a whole-body dose at the 5-mile EPZ boundary of 34 rem to a person in continuous occupancy at that location for the entire 24-hour period. This dose is well below a life-threatening dose of 100 rem. (Sinderman, ff. Tr. 2758, at 2, 6-7; Phillips, ff. Tr. 2859, at 4-5; Phillips, Tr. 2877, 2855-86.)

G-9. Licensee's core melt and containment failure evaluation assumed worst-case meteorological conditions. (Sinderman, Tr. 2801; Phillips, Tr. 2877.) During average meteorological conditions the dose rate would be approximately 100 times less than that calculated by Licensee. (Phillips, ff. Tr. 2859, at 5.)

G-10. The NRC Staff reviewed Licensee's analysis and concluded that its method was appropriately conservative. (Phillips, ff. Tr. 2859, at 4, 5.) On June 13, 1980, the Staff informed Licensee that 5- and 30-mile EPZs are appropriate for Big Rock Point. (Sinderman, ff. Tr. 2758, at 3, Attachment 1.)

G-11. Licensee's analysis did not take into account the plutonium stored in the spent fuel pool and the reactor. This plutonium is in a non-volatile oxide state. (Sinderman, Tr 2802.) In fact, plutonium oxide is considered a ceramic material, as are dishes, bones, and bricks. (Sinderman, Tr. 2828.) To become dangerous, the plutonium oxide must be transformed to a respirable state. (Sinderman, Tr. 2809, 2828.) However, no process of combustion could cause such a transformation.

(Sinderman, Tr. 2830.) Therefore, plutonium will not be released in the event of an accident. (Sinderman, Tr. 2802; Phillips, ff. Tr. 2859, at 7.)

G-12. The reactor core constitutes about 93% of the total fission product inventory at Big Rock Point. Only about 7% is provided by the spent fuel, and of this 7%, less than one-fifth is attributed to the proposed expansion of the Big Rock Point spent fuel pool. (Sinderman, ff. Tr. 2758, at 5; Sinderman, Tr. 2807.)

G-13. Licensee testified that the corresponding distances from Big Rock Point that would receive the same dose rates as would be received 10 and 50 miles from a typical 3800 MWT reactor under identical accident and meteorological conditions are 1.4 and 7 miles, respectively. This determination is based on meteorological dispersion data provided in Regulatory Guide 1.3 and also takes into account the smaller radiation source term at Big Rock Point, including the proposed expanded spent fuel pool. (Sinderman, ff. Tr. 2758, at 5-6.)

G-14. Licensee postulated criticality in the spent fuel pool at the maximum power and temperature levels that could occur. Licensee assumed that this condition would continue for the time necessary to achieve an equilibrium concentration of radionuclides similar to that of the reactor core, i.e., 3 years of continuous criticality (which could not occur because the pool would boil dry, interrupting or terminating criticality). This worst-case scenario results in an increase in the total inventory available for release of only 1.6%. This requires an increase in the calculated 5-mile plume exposure EPZ of only 64 meters, from 1.40 miles to 1.44 miles. The Licensee concluded that, even assuming this incredible criticality scenario occurred for purposes of analysis, the 5- and 30-mile EPZs remain more than adequate. (Sinderman, Turski, ff. Tr. 4346, at 4.)

G-15. Certain rain or snow conditions could conceivably cause a substantial amount of fission product inventory leaving the reactor to be deposited in a localized area near the plant. Such concentrations are known as "hot spots." (Sinderman, Tr. 2825-26.)

G-16. A most extreme hot-spot scenario entails literally smearing all of the semi-volatile particulates and halogens from the reactor core over a 22½° sector 3 miles from the plant site. Under such a scenario, a person standing in the middle of the sector would receive a dose rate of 5½ rem per hour. (Sinderman, Tr. 3201-02.)

G-17. The likelihood of such a meteorological phenomenon occurring concurrent with both core melt and containment failure is extremely low. (Phillips, Tr. 2892-94.)

G-18. Hot spots are disregarded for purposes of determining the size of the EPZs. However, protective actions are contemplated, on an *ad hoc* basis, if hot spots occur. (Phillips, Tr. 2910-12.)

H. Christa-Maria Subcontention 9(6) — Radiation Monitoring

H-1. Christa-Maria Contention 9, subpart (6) states:

> Applicant should comply with regulations requiring adequate radiation monitoring.

H-2. Charles E. Axtell is a health physicist who has had responsibility at Big Rock Point for radiation protection of plant personnel and the general public, as well as the chemistry aspects of plant operation. His testimony addresses effluent monitoring, in-plant iodine instrumentation, and containment radiation monitoring at Big Rock Point. (Axtell, ff. Tr. 2924, at 1, 2.)

H-3. Robert M. Marusich is a staff engineer in Licensee's Radiological Services Department. His testimony addresses Licensee's capability to promptly assess the degree of core damage following an accident. (Marusich, ff. Tr. 2924, at 1-3.)

H-4. Donald L. Swem is a general engineer at Big Rock Point. His testimony addresses the power sources of the high-range containment monitors in use at Big Rock Point, as well as their calibration. (Swem, Tr. 2982.)

H-5. Monte Phillips, an NRC emergency preparedness analyst, evaluated Licensee's radiation monitoring systems in the context of emergency planning, and found them to comply with currently applicable NRC regulations. (Phillips, ff. Tr. 2859, at 8-10; *see* Emch, Tr. 3111.)

H-6. Radiological monitors are placed throughout the Big Rock Point Plant to provide both local and control room annunciation and readouts. The effluent process monitoring system measures gross radioactivity levels of all airborne and liquid effluents released from the plant via the liquid and gaseous radwaste systems and the plant ventilation systems. (Phillips, ff. Tr. 2859, at 10.)

H-7. A wide range of radiation monitors are used at Big Rock Point, including process monitors, area monitors, emergency effluent monitors, wound monitors, portable monitors and sampling equipment. These monitoring systems comply with the guidance of subpart b of Criterion II.H.5 of NUREG-0654. (Axtell, Tr. 3044-47.) (*See also* findings regarding noble gas effluent monitor.

H-8. The noble gas effluent monitors at Big Rock Point measure all radioactivity released from the plant's gas stack during operation or shutdown, either in the form of noble gases, iodines or particulates. (Axtell, ff. Tr. 2924, at 3.)

H-9. During accident conditions, effluent monitors would enable Licensee to measure the radiation dose to the public at the site boundary. (*Id.*)

H-10. The Big Rock Point Plant has had a noble gas effluent monitor in operation since startup in 1962. This monitor met or exceeded all applicable regulations in effect at that time. (*Id.*)

H-11. The events at Three Mile Island caused both the Commission and Licensee to reevaluate the adequacy of effluent monitors. Investigation results indicated that the effluent monitor used at Big Rock Point did not have the capability to monitor stack releases during severe accident conditions. (*Id.*)

H-12. The NRC Staff issued NUREG-0578, recommending installation of effluent monitors of greater range and reliability. (*Id.* at 4.)

H-13. Because such monitors were not yet commercially available, the Commission required installation of an interim high-range noble gas effluent monitor. (*Id.*; Phillips, ff. Tr. 2859, at 9-10.)

H-14. In January 1980, an interim high-range effluent monitor was installed at Big Rock Point and approved by the NRC Staff. (Phillips, ff. Tr. 2859, at 9.)

H-15. NUREG-0737 recommended that permanent wide-range monitors be installed by January 1, 1982. Because a large number of orders for this equipment were placed with a small number of manufacturers within a relatively short time period, there was a problem with equipment availability, and the installation date at Big Rock Point had to be deferred several times. (Axtell, ff. Tr. 2924, at 4, 5.)

H-16. The new permanent noble gas effluent monitoring system was delivered to Big Rock Point in the Fall of 1982. (*Id.* at 5; Axtell, Tr. 2933.) However, spare parts for the equipment were not yet available, so that the system could not be quickly restored to operation if it broke down and required repair. This necessitated a further postponement of the in-service date of the permanent system. (Axtell, ff. Tr. 2924, at 5; Axtell, Tr. 2933.)

H-17. The NRC Staff approved this further postponement on condition that the interim high-range noble gas effluent monitor remain in operation until the permanent monitor is placed in service. (Axtell, ff. Tr. 2924, at 5; Axtell, Tr. 2933.)

H-18. On February 16, 1983, Licensee committed to placing the permanent monitor in service by December 31, 1983. This commitment was confirmed by Commission Order Confirming Licensee Commitments on Post-TMI Related Issues, dated March 14, 1983. (Axtell, ff. Tr. 2924, at 5; NRC Exhibit 4, ff. Tr. 3068, at 4.)

H-19. Spare parts were expected to become available for the permanent monitor beginning in late November or early December 1983. (Axtell, Tr. 3058.)

H-20. The permanent wide-range effluent monitoring system consists of two pieces of equipment: a low-range monitor to replace the monitor that has been used since 1962, and a high-range monitor to replace the interim high-range monitor. (Axtell, Tr. 3051-52; 3058.) This new effluent monitoring system complies, generally, with the recommendations contained in NUREG-0737, Item II.F.1, Attachment 1. (Axtell, Tr. 3029.)

H-21. The NRC Staff has inspected the new monitoring equipment to determine that it has been installed and calibrated correctly. (Phillips, Tr. 3086-88.) This equipment was the subject of a supplementary evidentiary proceeding. Findings on this subject are footnotes in the text of the Conclusions section of this Decision.

H-22. In-plant iodine sampling methods have recently been improved as a result of lessons learned from Three Mile Island. The filter medium silver zeolite was found to enable accurate sampling with only negligible interference from noble gases, something not possible with previous methods. (Axtell, ff. Tr. 2924, at 5, 6.)

H-23. Permanent iodine monitoring equipment using silver zeolite filters was installed at Big Rock Point in the Fall of 1982. The major components of this system are inspected at least weekly and recalibration is performed as necessary, but at least annually. (*Id.* at 6.)

H-24. In-plant iodine is sampled by a high-volume air sampler through which room air is drawn, located in the air compressor room. The sampler contains both a particulate filter and a silver zeolite filter, which are regularly removed and analyzed. (Axtell, Tr. 2932, 2957.)

H-25. High-range containment radiation monitors at Big Rock Point directly measure the radiation level inside containment. (Axtell, Tr. 3022.) They will be used to follow the course of a core damage accident. (Axtell, ff. Tr. 2924, at 6.)

H-26. These monitors were installed in April 1982, and were approved by the NRC Staff in a Safety Evaluation Report, dated October 18, 1982. (Axtell, ff. Tr. 2924, at 6.)

H-27. These monitors are located just outside containment, in the cable penetration room. (Axtell, Tr. 2935.) This placement, which is possible because of the lack of concrete containment shielding, is preferable because it avoids subjecting the monitors to hostile conditions which might be present in containment during an accident, such as steam, high humidity, high temperature and high pressure. (Axtell, ff. Tr. 2924, at 7; Axtell, Tr. 2938-39.)

H-28. The containment radiation monitors activate an alarm in the control room at a reading of approximately 12 rem per hour. (Axtell, Tr. 2942.)

H-29. Plant operators have been adequately trained to perform the straightforward task of reading the containment radiation monitors. (Marusich, Tr. 3004-05.)

H-30. Commission regulations do not require a backup system for monitoring. (Phillips, Tr. 3067.) NUREG-0737 does suggest that two separate high-range containment radiation monitors be installed. (Phillips, Tr. 3067.) Big Rock Point does have two such monitors, each of which is capable of backing up the other. If one fails, the other will be sufficient to provide a readout. (Axtell, Tr. 3039-40.)

H-31. The two containment radiation monitors are powered by an emergency AC power bus, which normally receives electricity from off site. (Swem, Tr. 2983-85.) The monitors are connected to the power bus by individual circuit breakers. (Swem, Tr. 2984.)

H-32. If offsite power is lost, an emergency diesel generator automatically powers the AC power bus. If the emergency diesel generator fails, a second emergency diesel generator is available as an additional power source. (Swem, Tr. 2983-91.) Licensee tests both the emergency bus and the diesel generator on a regular basis. (Swem, Tr. 2988.)

H-33. The containment radiation monitors were originally calibrated by the vendor over their entire range and are certified to remain in calibration for 18 months. (Axtell, ff. Tr. 2924, at 7.) They are recalibrated every year during the maintenance and refueling outage. (*Id.*; Tr. 2941; Swem, Tr. 3006.) In addition, an electronic calibration check is performed monthly. (Swem, Tr. 3006-07.)

H-34. The containment radiation monitors at Big Rock Point have error bands similar to those on other such monitors in use in the nuclear industry. (Swem, Tr. 2986-87.)

H-35. The manufacturer of the containment radiation monitors at Big Rock Point specifies an error band of plus or minus 36%. Licensee's own calibration procedures specify an accuracy of plus or minus 45%. During calibration at the time of the last refueling outage, the accuracy of the monitors was determined to be within 20%. (Swem, Tr. 2986.) Staff considers a margin of error within an order of magnitude to be acceptable. (Phillips, Tr. 3071.)

H-36. NUREG-0737, Item II.B.3, recommends post-accident sampling and analysis of reactor coolant and containment atmosphere as a means for determining, on a continuing basis, the degree of core damage following an accident. (Marusich, ff. Tr. 2924, at 3.)

H-37. Licensee has developed a calculational procedure, based upon data from the containment radiation monitors, as an alternative method of assessing core damage during an accident. (Marusich, ff. Tr. 2924, at 4, 5; Phillips, ff. Tr. 2859, at 10.)

H-38. The NRC Staff agrees that this procedure is an adequate alternative to post-accident sampling for estimating the degree of core damage. (Phillips, Tr. 3098.)

H-39. Most core damage scenarios include the release of core coolant into containment. Containment radiation monitors measure the radiation level in containment generated by the coolant. The extent of damage to the core then may be estimated by comparing the actual radiation level with the level which would be present following a 100% core meltdown. (Marusich, ff. Tr. 2924, at 5.)

H-40. This calculational procedure is part of Licensee's Site Emergency Plan Implementing Procedure. It is Procedure 5D, entitled "Proceuure to Determine Extent of Core Damage for 0 to 100% Meltdown." (*Id.*, Attachment 2.)

H-41. The radiation field associated with a 100% meltdown is determined by assuming that the radionuclide release to containment is instantaneous and that the source term is that set forth in Regulatory Guide 1.3 (as recommended by NUREG-0737). Credit is taken for radioactive decay and removal of radionuclides by containment sprays and surface deposition. From this data, a curve associated with 100% core melt as a function of time is developed. (Marusich, ff. Tr. 2924, at 5, 6; Marusich, Tr. 2967.)

H-42. The curve is used to postulate the radiation level which would be present within containment at any particular time following a 100% core melt. The actual radiation level following an accident, as measured by the containment radiation monitors, is then divided by the radiation levels which would be present following a 100% core melt. This calculation would reveal the approximate percentage of the core which has actually been damaged. (Marusich, ff. Tr. 2924, at 6.)

H-43. This calculation is simple, straightforward, and probably will take less than 1 minute to perform. (Marusich, Tr. 2947.) Moreover, the mathematical models used to estimate core damage based on containment radiation levels have been reviewed and accepted by the NRC Staff. (Emch, Tr. 3090-92.)

H-44. There may be situations in which a 100% core melt would not look like the core melt postulated in Regulatory Guide 1.3, and core damage estimates based on this calculation may be inaccurate by as much as a factor of 2. (Marusich, Tr. 2969, 3024-25.) However, Regulatory Guide 1.3 remains the NRC Staff's best judgment of the appropriate source terms absent revision based upon ongoing studies. (Phillips, Tr. 3103-05.)

H-45. Precise core damage information is of little value in determining whether to take protective actions. (Marusich, Tr. 2966, 3040; Phillips, Tr. 3108.) Rather, the status of the core, whether it is intact or insignificant cladding failure has occurred, is really what is of concern. The readings from the monitor corresponding to these kinds of core status changes differ by several orders of magnitude. (Phillips, Tr. 3071-72.)

H-46. The margin of error for these instruments is sufficient for the purpose of estimating core damage. (Marusich, Tr. 2966; Phillips, Tr. 3071, 3108.)

H-47. The Licensee has developed protective action recommendations that would be implemented prior to a release based on the high-range monitor and containment status. (Marusich, Tr. 3043; Phillips, Tr. 3075-76.) These readings have been predetermined, placed in the Licensee's procedures, and are considered adequate by the NRC Staff. (Phillips, Tr. 3075-76.)

H-48. If an accident damages the core but does not release coolant into containment, the containment radiation monitors will not provide the information necessary to estimate core damage. In such a case, any release will be detected and measured by the high-range effluent monitors, continuous air monitors and by surveys performed by radiation protection technicians. These readings, coupled with the Emergency Plan Implementing Procedures, which address releases outside containment, provide sufficient information to determine adequate public protective action. (Marusich, ff. Tr. 2924, at 6, 7; Axtell, ff. Tr. 2924, at 3-7; Marusich, Tr. 3017-18; Phillips, Tr. 3075-76.)

H-49. For an accident with no release of coolant into containment, post-accident sampling can be conducted inside containment for accident sequences resulting in less than 10% core melt. (Emergency Preparedness Appraisal, ff. Tr. 3065, at 13.)

H-50. A preliminary report by the NRC Staff finds that the containment radiation monitors are adequate to assess the degree of core damage, and that additional coolant or atmosphere sampling is not necessary. (Emch, Tr. 3111.)

H-51. The Board finds that the containment radiation monitoring system, in conjunction with the calculational procedure presented by Licensee, provides an adequate alternative means to determine the extent of core damage under accident conditions and therefore satisfies the principal purpose of NUREG-0737, Item II.B.3.

I. Christa-Maria Subcontention 9(8) — Summer and Winter Emergency Plans

I-1. Christa-Maria subcontention 9(8) states:

> Applicant should have separate emergency plans appropriate for summer and winter.

I-2. On April 20, 1982, the Board determined that a genuine issue of fact existed as to whether the emergency plans adequately accommodate the difficulties associated with winter weather and the complications caused by large numbers of summer visitors.

I-3. Licensee presented the testimony of six witnesses. Messrs. Roger W. Sinderman, the Director of Licensee's Radiological Services Department, and Robert D. Klimm, a Project Manager and Senior Transportation Engineer for HMM Associates, appeared as a panel. Mr. Sinderman testified as to the division of emergency planning responsibilities between the Licensee, the State and local units of government; the protective measures available; and the factors, including seasonal conditions, that Licensee considers in making protective action recommendations. Mr. Klimm testified as to the purposes, assumptions and limitations of the Evacuation Time Study. Messrs. Earl Muma, John F. Hess, Fred Welch, and Sheriff George T. Lasater, members of the Charlevoix County emergency response organization who will man the Emergency Operations Center ("EOC"), appeared as a panel. Mr. Muma, the Emergency Services Director, testified as to the mobilization of the county EOC and the circumstances under which the county, and not the State of Michigan, would determine the appropriate protective measure. Mr. Hess, who is designated the county's Radiological Defense Analysis Section Official, described how he would evaluate the protective action recommendation to be provided to the county by Licensee. Sheriff Lasater's testimony addressed the difficulties to an emergency evacuation posed by adverse winter weather and the influx of transient population during the summer. Mr. Welch, who is designated as the county's Public Works Official, testified as to the County and City of Charlevoix's road-clearing capabilities.

I-4. The NRC Staff presented the testimony of Mr. Monte Phillips, an emergency preparedness analyst and section chief.

I-5. Intervenors presented written testimony of one witness. Ms. Christa-Maria testified concerning the traffic effects of severe weather conditions during the winter and of the number of tourists who attend rock concerts and festivals in and near the City of Charlevoix during the summer. In addition, Ms. Liane Christiansen appeared on behalf of Intervenors and related her recollection of an incident where the drawbridge in the center of the City of Charlevoix was inoperable.

I-6. At the request of Licensee, the Board called Mr. Joseph M. Hennigan as a Board witness. Mr. Hennigan is the Chief of the Nuclear Facilities and Environmental Monitoring Section of the Radiological Health Services Division of the Michigan Department of Health. Mr. Hennigan testified concerning the State's protective action decisionmaking process.

I-7. Responsibilities and actions during nuclear power plant emergencies are divided among the utility, the State and local units of government. (Sinderman, ff. Tr. 3134, at 2.)

I-8. Licensee performs various actions to monitor plant conditions, assess and classify the accident and implement protective actions for people on site. Licensee also provides State and local units of government with information and recommendations so that they may take the appropriate action to reduce or eliminate consequences to the public. (Sinderman, ff. Tr. 3134, at 2; Sinderman, Tr. 3196-97.)

I-9. The State of Michigan, unless the State Emergency Operations Center has not become operational and the accident is developing rapidly, is responsible for making the ultimate decision whether or not public protective actions are required and, if so, what action is appropriate. (Sinderman, ff. Tr. 3134, at 2-3; Muma, ff. Tr. 3235, at 4-6; Hennigan, ff. Tr. 3296, at 1-2; Hennigan, Tr. 3302, 3338.)

I-10. In the event the accident is developing rapidly and the State Emergency Operations Center is not operational, the Chairperson of the County Board of Commissioners, using the county EOC staff's assessment of Licensee's recommendations, will determine which, if any, protective measures should be implemented. (Muma, ff. Tr. 3235, at 4-5.)

I-11. The two primary protective measures which can be implemented in response to an accident at the Big Rock Point nuclear plant are sheltering and evacuation. (Sinderman, ff. Tr. 3134, at 3; Phillips, ff. Tr. 2859, at 11; Hennigan, ff. Tr. 3296, at 2.)

I-12. The objective of public protective measures is to minimize the public's exposure to radiation. Evacuation is the preferred protective measure and is appropriate when the expected radiation risk to the public from sheltering exceeds the expected risk from evacuation. (Sinderman, ff. Tr. 3134, at 3-5; Hennigan, ff. Tr. 3296, at 2-3; Phillips, ff. Tr. 2859, at 11-12.)

I-13. Several factors are taken into account in making the choice between sheltering and evacuation. These factors include the amount of time before the expected onset of the release, current meteorological conditions, weather forecast, an estimate of the magnitude and duration of the release, road conditions, the presence of a large transient population, and the estimated time necessary to accomplish an evacuation. (Sinderman, ff. Tr. 3134, at 6; Hennigan, ff. Tr. 3296, at 3; Hess, ff. Tr. 3235, at 3; Sinderman, Tr. 3138, 3144, 3191; Hennigan, Tr. 3303, 3313-14, 3326; Hess, Tr. 3236-37; Phillips, Tr. 3487-88.)

I-14. Licensee's methods for monitoring weather provide information on current meteorological conditions as well as 6- and 12-hour weather forecasts. (Sinderman, Tr. 3143, 3145-47, 3149.)

I-15. Notification, communication and assessment actions are not significantly affected by seasonal conditions. (Phillips, ff. Tr. 2859, at 11; Lasater, ff. Tr. 3235, at 5-6.)

I-16. Seasonal conditions, including severe winter weather, have little impact on the effectiveness of sheltering. (Sinderman, Tr. 3184, 3187-90, 3197, 3209-13.)

I-17. However, seasonal conditions, such as adverse winter weather and summer transient population, may lengthen the travel times necessary for evacuation. (Sinderman, ff. Tr. 3134, at 4.) Adverse winter weather may also make evacuation impractical or increase the risks associated with evacuation due to the increased likelihood of accidents. (Phillips, ff. Tr. 2859, at 11-12; Sinderman, Tr. 3195-98, 3205.)

I-18. The evacuation time estimates which have been developed by HMM Associates reflect seasonal conditions and are contained in Licensee's, the State's, and the County of Charlevoix's emergency plans in order to guide their choice between protective actions. (Sinderman, ff. Tr. 3134, at 7-9; Klimm, ff. Tr. 3137, at 3; Muma, ff. Tr. 3235, at 5; Hess, ff. Tr. 3235, at 3-4; Phillips, ff. Tr. 2859, at 11-12; Hennigan, Tr. 3306-09.)

I-19. HMM Associates has updated the evacuation time study performed in 1980. The 1980 evacuation study was performed prior to the installation of the prompt public notification system and defined adverse winter weather in a manner no longer consistent with the guidance provided in NUREG-0654, which requires an assessment incorporating a reduction in both roadway capacity and travel speeds. The updated study reflects these two developments and was completed in February 1984. (Klimm, ff. Tr. 3137, at 4-5, 8, 10; HMM Associates, Inc., "Evacuation Time Estimates for Areas Near the Big Rock Point Plant," HMM Document No. 83-600, February 1984.)

I-20. The updated evacuation time study considers peak summer population in two ways. First, the study considers summer transient populations resulting from summer residences and recreational facilities in the Big Rock Point area. Second, the updated study evaluates the impact of special events, such as local festivals occurring within the City of Charlevoix, on evacuation time estimates. The updated study also assesses the potential impact of rock concerts, held outside the EPZ, on the flow of evacuation traffic from the EPZ. (Klimm, ff. Tr. 3137, at 6; Klimm, Tr. 3162, 3165-66, 3168, 3179-82.) However, the updated study does not adequately discuss the potential for reducing delays attendant to rock concerts.

I-21. The adverse winter weather condition considered in the updated study will not be a worst case. The scenario is intended to reflect conditions under which evacuation is feasible but more difficult due to adverse weather. The adverse weather condition to be studied by HMM Associates will be consistent with NUREG-0654 and is expected to assume a reduction in roadway capacity and travel speeds on the order of 30%. (Klimm, ff. Tr. 3137, at 6; Klimm, Tr. 3171-73, 3176-79, 3219-22.)

I-22. The updated study evaluates the effect of adverse winter weather conditions on preparation and mobilization times. (Klimm, Tr. 3218.)

I-23. HMM Associates prepared interim evacuation time estimates for use by the Licensee, State, and the County of Charlevoix until the updated study is completed. (Klimm, ff. Tr. 3137, at 9; Sinderman, ff. Tr. 3135, at 8-9.)

I-24. Interim estimates were developed for three scenarios: summer weekend, fair weather condition; winter weekday with characteristic winter weather (light wind and snow); and winter weekday with adverse weather. (Klimm, ff. Tr. 3137, at 12-13; Sinderman, ff. Tr. 3134, at 8-9.)

I-25. The summer weekend scenario included vehicle demand associated with permanent residents, seasonal residents and peak summer transient population, including campers. (Klimm, ff. Tr. 3137, at 11; Klimm, Tr. 3165-66.)

I-26. The winter condition included vehicle demand associated with permanent residents, winter seasonal residents and employers. Adverse winter weather conditions were accounted for by reducing roadway capacity and travel speeds by 30%. (Klimm, ff. Tr. 3137, at 11-12; Klimm, Tr. 3171-73, 3176-79, 3219-22.)

I-27. The interim time estimates developed by HMM Associates were provided to Licensee, the County of Charlevoix, and the Nuclear Facilities and Environmental Monitoring Section of the Radiological

Health Services Division of the Michigan Department of Health. (Sinderman, ff. Tr. 3134, at 8; Hess, ff. Tr. 3235, at 3; Hennigan, Tr. 3306, 3311.) The results of the updated time study will be provided to and reviewed by Licensee, the County of Charlevoix, and the Michigan Department of Health, and the emergency plans and procedures will be revised whenever appropriate. (Sinderman, ff. Tr. 3134, at 9; Hess, ff. Tr. 3235, at 3; Hennigan, Tr. 3318.)

I-28. The times necessary to clear roadways during a heavy snowfall, the effects of automobile accidents or a broken drawbridge, and the characteristics and location of "whiteouts" as well as other highly specific roadway incidents, are not factored into the evacuation time estimates. As conditions depart from the circumstances considered in the evacuation time studies, the decisionmakers will use their experience and judgment to factor such conditions into their evaluation of the appropriate protective measures. (Klimm, Tr. 3169, 3171-75, 3217; Hess, ff. Tr. 3235, at 3; Hess, Tr. 3239-40; Phillips, Tr. 3491-93.)

I-29. The State and county officials responsible for determining the appropriate protective measures will obtain information concerning local conditions. The local officials responsible for road clearing will estimate the time required to make the roads passable and will convey this information to the County Radiological Defense Analysis Section Official and Michigan Department of Health. (Welch, Tr. 3374-75, 3382-83, 3385-86; Hess, Tr. 3276; Hennigan, Tr. 3326-27.) Similarly, other information regarding the condition of the roadways and presence of a large transient population will be provided to the Radiological Defense Analysis Section Official by the Law Enforcement Coordinator. (Hess, ff. Tr. 3235, at 3; Hennigan, Tr. 3326-27.)

I-30. The County of Charlevoix is responsible for coordination and implementation of public protective actions, including public protective actions for the City of Charlevoix. (Sinderman, ff. Tr. 3134, at 3; Welch, ff. Tr. 3235, at 6; Muma, Tr. 3259-61, 3264, 3397-3401; Lasater, Tr. 3253-55.)

I-31. The county emergency plan and procedures are designed to be sufficiently flexible to, among other things, accommodate seasonal conditions. The plan delegates authority to provide emergency services to several individuals, with each individual free to make the appropriate decision and allocate the resources necessary to handle special or unusual circumstances. (Muma, Tr. 3261-62, 3266.)

I-32. The agencies with emergency response responsibilities, including, among others, law enforcement, fire operations, public works and the health department, will be represented at the operations table at the County Emergency Operations Center. Special circumstances, such as special events and adverse weather conditions, will be evaluated, discussed and factored into the county's implementation of the public protective measure by the officials manning the operations table. (Muma, Tr. 3261-62, 3267.)

I-33. The evacuation pattern will be altered to accommodate variations and shifts in wind patterns. (Sinderman, Tr. 3147-48.)

I-34. The choice of evacuation routes will be made by the Law Enforcement Coordinator in conjunction with the Emergency Services Coordinator and the Public Works official. The county emergency plan identifies the primary routes which could be used for evacuation. The routes will be chosen, however, to reflect the local seasonal conditions. (Lasater, ff. Tr. 3235, at 3-4.)

I-35. Should an evacuation be required during or immediately following a snowstorm, and the sector to be evacuated has been identified, the primary roads for evacuation out of the sector will be determined and given priority for any necessary plowing. (Lasater, ff. Tr. 3235, at 3.)

I-36. The roads to be plowed will be communicated to the public works official, who in turn will communicate this information to the Charlevoix County Road Commission. (Welch, ff. Tr. 3235, at 7-8; Welch, Tr. 3244.)

I-37. The Charlevoix County Road Commission and the City of Charlevoix Street Department have snow-plowing equipment, and extensive snow-plowing experience. (Welch, ff. Tr. 3235, at 2-3, 8-9; Welch, Tr. 3244, 3388-89.)

I-38. The Charlevoix County Road Commission will aid in the removal of stalled vehicles. (Lasater, ff. Tr. 3235, at 4.)

I-39. The County Sheriff's Department has four-wheel-drive vehicles capable of responding to emergencies under severe winter weather conditions. These vehicles will be dispatched under the direction of the Law Enforcement Coordinator. (Lasater, Tr. 3246-48.)

I-40. The County Sheriff's Department will be particularly conscientious during the winter months about warning people who are not close to the public siren system. (Lasater, ff. Tr. 3235, at 6.)

I-41. The county emergency plan and procedures provide flexibility in the choice of access and traffic control points. Suggested access and control points are included in the county emergency plan. (*Id.* at 4.)

I-42. During the summer, additional access control points may be established outside the campgrounds to direct existing traffic. Traffic control points will be established to aid the flow of traffic at intersections and stoplights. (*Id.*)

I-43. The Law Enforcement Coordinator will be aware of special events in the area surrounding the Big Rock Point Plant. The Law Enforcement Coordinator will make assignments and delegate manpower, vehicles and equipment to accommodate these special events. (Muma, Tr. 3266-67; Lasater, Tr. 3249.)

I-44. The County Sheriff's Department and City of Charlevoix Police Department have had experience with mass gatherings and crowd control, including rock concerts and local festivals. They have handled their tasks in a professional manner. (Lasater, Tr. 3249-52, 3281.)

I-45. Traffic from the local rock concert will not prevent law enforcement personnel from responding to an emergency. (Lasater, Tr. 3252.)

I-46. The County emergency plan provides for the warning of boaters on Lake Michigan and Lake Charlevoix. (Lasater, ff. Tr. 3252, at 5.)

J. Christa-Maria Subcontention 9(9) — Special Evacuation Measures for Children and Pregnant Women

J-1. Christa-Maria subcontention 9(9), states:

> Appropriate emergency plans should be made for children and pregnant women to evacuate at appropriate levels of radiation, considering their special susceptibility.

J-2. Roger Sinderman, a health physicist with a Masters of Public Health degree from the University of Michigan, testified for Licensee. (Sinderman, ff. Tr. 3511.)

J-3. Monte Phillips, an NRC emergency preparedness analyst, testified on behalf of the NRC Staff. (Phillips, ff. Tr. 2859, at 13-15.)

J-4. Children at or below the age of puberty (12 years of age) and the developing human fetus are more sensitive to radiation than the public-at-large. (Sinderman, ff. Tr. 3511, at 2-3; Sinderman, Tr. 3515-16, 3548; Phillips, ff. Tr. 2859, at 13.)

J-5. Emergency plans for nuclear power plants should provide guidance for the evacuation of children and pregnant women. (Sinderman, ff. Tr. 3511, at 3-5; Phillips, ff. Tr. 2859, at 13-14; NUREG-0654; "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, September 1975.)

J-6. The implementing procedures for the Big Rock Point Plant site emergency plan recommend that pregnant women and children be evacuated at a lower whole-body dose level (1 rem) than that of the general public (5 rem). (Sinderman, ff. Tr. 3511, at 6-7 and Attachment 1; Phillips, ff. Tr. 2859, at 14.)

J-7. The State of Michigan Emergency Preparedness Plan and the Charlevoix County Basic Plan recommend that pregnant women and children be evacuated at a lower whole-body dose level (1 rem) than that of the general public (5 rem). (Sinderman, ff. Tr. 3511, at 6-7, Attachments 2 and 3; Phillips, ff. Tr. 2859, at 14-15.)

J-8. The early evacuation guideline in the State of Michigan and Charlevoix County emergency plans was recently revised from 0.5 rem to coincide with the 1-rem EPA guideline. (Sinderman Tr. 3507, 3514-15; Phillips, Tr. 3565.)

J-9. The 1-rem guideline for the early evacuation of children and pregnant women was established by the U.S. Environmental Protection Agency, based on its balancing of the risks of radiation exposure against the risks of the fiscal, nonradiological health and other societal costs of evacuation. (Sinderman, ff. Tr. 3511, at 4; Sinderman, Tr. 3525, 3536-38.)

J-10 (Deleted.)

J-11. Despite the lack of verified health effects at doses of 1 rem, some risk to the fetus probably exists based on the widely accepted hypothesis that there is a linear relationship between dose and risk. (Sinderman, ff. Tr. 3511, at 9.)

J-12. Mr. Sinderman agrees with the views of Dr. Robert L. Brent, a medical doctor, that the overall risk to the fetus from a dose of 1 rem is about 1 in 1000 to 1 in 10,000. (Sinderman, ff. Tr 3511, at 9; Sinderman, Tr. 3543-44.)

CONCLUSIONS OF LAW

G-1. With respect to the plume exposure pathway EPZ for Big Rock Point, the evidence demonstrates that early severe health effects from whole-body doses will not occur at distances from the plant site greater than 5 miles, even for the worst accidents referenced in NUREG-0654. Similarly, with respect to the ingestion pathway EPZ, there is a minimal potential for significant contamination of food supplies at distances greater than 30 miles from the plant site. Accordingly, the Board concludes that the 5- and 30-mile EPZs for Big Rock Point are sufficient.

G-2. The Board further concludes that the incremental impact of the proposed spent fuel pool expansion on the amount of radioactive inventory. will be insignificant. Therefore, the proposed expansion does not warrant any increase in the size of the EPZs.

H-1. The Board concludes that Licensee's monitoring systems are adequate to assess and monitor the actual or potential offsite consequences of possible radiological emergency conditions, consistent with 10 C.F.R. § 50.47(b)(9). *See also* NUREG-0654, Items II.H.5 and II.12.

I-1. The record demonstrates that the interim evacuation time estimates consider site winter weather conditions in compliance with 10 C.F.R. § 50.47(b)(10) and the guidance of Appendix 4 to NUREG-0654.

I-2. The record demonstrates that the updated time estimate study considers the impact of seasonal conditions, including the peak summer tourist season and adverse winter weather conditions, in compliance with 10 C.F.R. § 50.47(b)(10) and the guidance of Appendix 4 to NUREG-0654.

I-3. The Board finds that Licensee, State and local planners have adequately evaluated the impact of seasonal conditions on the effectiveness of various protective actions and that seasonal conditions are adequately taken into account and accommodated in the choice between protective actions in compliance with 10 C.F.R. § 50.47(b)(10) and the guidance provided by § II.J of NUREG-0654.

I-4. The record establishes that seasonal conditions have been adequately considered in the allocation of emergency responsibilities among the various supporting organizations.

I-5. The Board concludes that the Licensee, State and County of Charlevoix emergency plans and response organizations are adequately prepared and appear to be sufficiently flexible to accommodate seasonal conditions in the implementation of public protective measures in compliance with 10 C.F.R. § 50.47(b)(10) and the guidance provided by § II.J of NUREG-0654.

I-6. The record provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency in the summer or winter, consistent with 10 C.F.R. § 50.47(a)(1), and that separate summer and winter emergency plans are neither necessary nor required.

J-1. The emergency plans of the Licensee, State of Michigan and Charlevoix County provide adequate specific provisions for the early evacuation of small children and pregnant women.

J-2. The EPA guideline of 1 rem, that has been adopted as the trigger for the consideration of the early evacuation of small children and pregnant women, is adequate.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

John H Frye, III, Chairman
Gustave A. Linenberger
Dr. Frank F. Hooper

In the Matter of **Docket No. 50-358-OL**
(ASLBP No. 76-317-01-OL)

CINCINNATI GAS & ELECTRIC COMPANY, ***et al.***
(William H. Zimmer Nuclear Power Station, Unit 1) **August 29, 1984**

Licensing Board grants Applicants' unopposed motion to withdraw their application for an operating license for the Zimmer Station and to terminate this proceeding, subject to the condition that Applicants implement, with Staff verification, their site restoration plan. The Board refuses to impose a condition, consented to by Applicants, that the grant of the motion be with prejudice to any future application by these Applicants for a nuclear reactor at this site on the ground that such a condition is unnecessary.

LICENSING BOARDS: DISMISSAL OF PROCEEDINGS

Dismissal of an operating license application with prejudice is a severe sanction which is reserved for unusual situations where it is necessary to prevent substantial prejudice to a party who opposed the application.

MEMORANDUM AND ORDER
(Ruling on Applicants' Motion to Withdraw Application)

On March 20, 1984, Applicants moved for an Order authorizing withdrawal of their application for an operating license for this facility and dismissing this proceeding. In support of their motion, Applicants represented that:

(1) All fuel would be removed from the site by August 31, 1984;

(2) The nuclear steam supply system would be modified to prevent its operation as a "utilization facility" (defined by § 11(cc) of the Atomic Energy Act) by:

- (a) severing and welding caps on the two main feedwater lines and four main steam lines; and
- (b) removing the control rod drive mechanisms;

(3) The balance of the plant will be used to the extent possible as part of a fossil-fired generating station; and

(4) Applicants have no objection to the dismissal of the application "with prejudice."

Only the NRC Staff responded to this motion. In its April 9, 1984, response, Staff points out that § 11(cc) of the Atomic Energy Act defines a "utilization facility" as one which is capable of making use of special nuclear material. Therefore, according to Staff, because the facility is essentially complete, it must be disabled so that it cannot make use of special nuclear material. Staff found that the modifications which Applicants represented they would make would accomplish this purpose. Staff therefore urged that the motion be granted subject to the condition that these modifications be made and to the condition that the fuel be shipped from the site by August 31, with implementation of the conditions to be verified by Staff.

Staff also noted that it had no objection to dismissal of the application with prejudice and urged that we include such a condition. Staff gave no reasons for this position.

Finally, Staff noted that it was reviewing the site to determine whether conditions for the protection of the environment were necessary. Staff indicated that it would advise the Board of its conclusions in this regard.

On August 2, 1984, Applicants filed certain information with the Board relevant to their motion. In this filing, Applicants advised us that they had shipped their fuel off site and had accomplished the modifications to the nuclear steam supply system which they represented they would make. Applicants therefore renewed the request contained in their motion. On August 7, the Board Chairman wrote counsel for Applicants indicating that the Board would act on the motion promptly upon receiving Staff's conclusions with regard to the need for conditions to protect the environment.

On August 17, the Staff filed a further response to the Applicants' motion. Staff noted that it had conducted an inspection and verified that the feedwater and main steam lines had been severed and capped, and that the Applicants were in the process of removing the control rod drive mechanisms. During the inspection, Staff verified that the fuel had been removed from the site. This inspection was conducted from April 27 through July 16, 1984. Staff attached a copy of Inspection Report 50-358/84-05 to its response.

Staff also advised us that it had reviewed certain additional information relevant to environmental protection which Applicants furnished in response to Staff's request and had visited the site. Staff concluded that, based upon this review, withdrawal of the application should be conditioned on implementation of Applicants' June 1, 1984, restoration plan (which was furnished with the information Staff requested), such implementation to be verified by Staff. Staff furnished its environmental review and the affidavit of Germain La Roche in support of its conclusion.

After receiving Staff's August 17 response, we inquired of Applicants' counsel whether he wished to reply and were informed that he did not.

We agree with Staff that it is necessary that the nuclear steam supply system be modified to prevent its utilization of special nuclear material and that the reactor fuel be shipped off site. We are satisfied that these steps have been accomplished. Having heard no objection from Applicants, we will condition our authorization to withdraw the application on implementation of the June 1, 1984, site restoration plan, such implementation to be verified by Staff.

Applicants do not object to the authorization of withdrawal of the application with prejudice and have included such a provision in the draft order accompanying their motion. That provision states that the authorization is "with prejudice to future reapplication by the Applicants for the construction and operation of any nuclear power facility at the same site." Staff, without elaboration, urges that the authorization be so conditioned. Ordinarily such a condition would only be imposed if substantial prejudice would otherwise result to a party who opposed the application. *See Puerto Rico Electric Power Authority* (North Coast Nuclear Plant, Unit 1), ALAB-662, 14 NRC 1125 (1981) and *Philadelphia Electric Co.* (Fulton Generating Station, Units 1 and 2), ALAB-657, 14 NRC 967 (1981). Here no party has seen fit to attempt to make such a showing. And despite years of consideration of both the construction permit and operating license, no final agency decision has been rendered which disapproves these Applicants, this site, or this reactor. In these circumstances, we view the attachment of such a condition to the authorization to withdraw the application as unnecessary. Therefore we have not included such a condition.

In consideration of the foregoing, it is, this 27th day of August 1984, ORDERED that:

Applicants' motion for authorization to withdraw their application and for termination of this proceeding is granted subject to the condition that Applicants are to implement their June 1, 1984, site restoration plan and Staff is to verify that this has been accomplished within 6 months of the date of this Memorandum and Order.

Dr. Hooper concurs but was unavailable to sign this Memorandum and Order.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

Gustave A. Linenberger
ADMINISTRATIVE JUDGE

John H Frye, III, Chairman
ADMINISTRATIVE JUDGE

Bethesda, Maryland
August 29, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

Herbert Grossman, Chairman
Dr. Richard F. Cole
Dr. Emmeth A. Luebke

In the Matter of **Docket No. 50-244-OLA**
(ASLBP No. 79-427-07-OLA)

ROCHESTER GAS & ELECTRIC CORPORATION
(R.E. Ginna Nuclear Plant, Unit 1) **August 30, 1984**

In this Memorandum and Order the Licensing Board dismisses the proceeding in view of the withdrawal of the sole Intervenor and the consequent removal of all issues requiring hearing.

MEMORANDUM AND ORDER
(Terminating Proceeding)

Memorandum

By Memorandum and Order dated May 25, 1984 (unpublished), the Licensing Board directed that discovery commence in the proceeding which had, in effect, been suspended during the lengthy Staff review under the Systematic Evaluation Program. It also directed that the parties file status reports by August 15, 1984, containing their proposed prehearing and hearing schedules. On July 13, 1984, Rochester Gas & Electric Corporation (Applicant) served its first set of interrogatories on the sole intervenor, Michael A. Slade.

By a pleading dated July 24, 1984, Mr. Slade withdrew all of his outstanding contentions in this proceeding. To date, two intervenors have been admitted to this proceeding: Mr. Slade and the Rochester Committee for Scientific Information (RCSI). RCSI subsequently withdrew from the proceeding pursuant to stipulation with Applicant. The withdrawal of RCSI was accepted by the Licensing Board. The State of New York became, and still remains, a participant in this proceeding, but only as an interested State pursuant to 10 C.F.R. § 2.715(c). On March 12, 1974, Counsel for the State of New York appeared at the only prehearing conference convened to date and indicated that the State was not intervening in this proceeding with contentions and that the State had no position on the licensing of the plant at that time. The State has filed no contentions since that time. Nor, since Michael Slade's notification to the Board that he intends to withdraw his contentions, has New York State indicated that it wishes to file any.

The withdrawal of the only intervenor removes both the need and the occasion for evidentiary hearings in this proceeding. There are no longer any matters which the parties wish to resolve in this proceeding and, consequently, there is no issue to be heard by the Board.

Dismissal of this proceeding would be consistent with the Commission's requirements which do not contemplate a hearing on a application for an operating license in the absence of any matters in controversy or any request for hearing by interested persons (*see* 10 C.F.R. §§ 2.104, 2.105, 2.714, 50.58(b) and 50.91) and is consistent with the general powers of the presiding officer under 10 C.F.R. § 2.718.

Order

For all of the foregoing reasons and based upon the entire record in this proceeding, it is, this 30th day of August 1984,

ORDERED

That this proceeding, begun with the issuance of a notice of opportunity for hearing on December 8, 1972, published at 37 Fed. Reg. 26,144, is hereby terminated.

THE ATOMIC SAFETY AND
LICENSING BOARD

Richard F. Cole
ADMINISTRATIVE JUDGE

Emmeth A. Luebke
ADMINISTRATIVE JUDGE

Herbert Grossman, Chairman
ADMINISTRATIVE JUDGE

Directors' Decisions Under 10 CFR 2.206

DIRECTORS' DECISIONS

Cite as 20 NRC 773 (1984) DD-84-19

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
	(10 C.F.R. § 2.206)
PACIFIC GAS AND ELECTRIC COMPANY	
(Diablo Canyon Nuclear Power Plant, Units 1 and 2)	**August 20, 1984**

The Director of Nuclear Reactor Regulation denies a request by the Government Accountability Project on behalf of two former employees at the Diablo Canyon facility that licensing be deferred until alleged intimidation and harassment on site is neutralized. The Director concluded that, even if the two petitioners were improperly terminated, harassment and intimidation and possible coverup of deficiencies were not such substantial problems at the site as to require deferral of a licensing decision. However, the Director also determined that a final determination on the petitions will be made upon completion of investigations into the harassment and intimidation of the petitioners.

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

By petition and supplemental documents, dated July 27, July 29, July 30 and July 31, 1984, the Government Accountability Project (GAP) on behalf of Timothy J. O'Neill and James L. McDermott filed a request pursuant to 10 C.F.R. § 2.206 of the Commission's regulations that further licensing decisions on the Diablo Canyon Nuclear Power Plant be deferred until alleged harassment and retaliation on site are neutralized, organizational freedom for quality assurance (QA) inspectors is restored and all project personnel are retrained in NRC quality assurance and employee protection requirements. In accordance with the Commission's usual practice, the petitioners' request was referred to the Staff for appropriate action.

Mr. O'Neill asserts that he was not provided adequate organizational freedom to carry out his quality control responsibilities and that he was harassed and threatened with disciplinary action for attempting to identify certain quality assurance deficiencies. Mr. O'Neill states that it was for these reasons he resigned on July 24, 1984, as a quality control inspector at the Diablo Canyon site. Mr. McDermott asserts in his affidavit that his layoff on July 28, 1984, was retaliatory because he had refused to sign forms on retraining which he felt were inaccurate and covered subjects on which adequate training had not occurred.

The NRC Staff has had a continuing concern about assertions of intimidation at the Diablo Canyon site which might inhibit workers from adequately completing their work or identifying deficiencies which could have an impact on safe operation of the facility. In order to determine whether a widespread problem existed, NRC inspectors conducted earlier this year structured interviews with approximately 250 workers selected at random on site. Numerous additional informal inquiries on the specific question of harassment were made. The NRC Staff has been continuing inspections at the site throughout the Summer which have involved numerous contacts with employees. Based on all of these contacts, the Staff has concluded that widespread or pervasive intimidation of employees is not a problem at the site.

That is not to say one way or the other whether Mr. O'Neill and Mr. McDermott were intimidated or improperly terminated. The Office of Investigations (OI) is investigating their assertions. If the petitioners were improperly terminated or harassed, then the NRC Staff will consider appropriate enforcement action against the licensee. However, the Staff has concluded on the basis of its interviews with employees on their working environment, its own inspections and reviews of the Diablo Canyon facility and its investigations of various allegations of specific deficiencies, that the allegations of harassment and intimidation and possible coverup of deficiencies identified in this petition do not raise such significant safety questions that the licensing of the Diablo Canyon facility should be deferred or that the other relief requested is mandated prior to licensing. (*See* Transcript of Commission meeting, August 2, 1984, at 30-31.) Consequently, those aspects of Mr. O'Neill's and Mr. McDermott's petitions are *denied*. A final determination on the petitions will be made on completion of the OI investigations.

A copy of this decision will be filed with the Office of the Secretary, pursuant to 10 C.F.R. § 2.206(c).

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland,
this 20th day of August 1984.

Cite as 20 NRC 776 (1984) DD-84-20

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of	**Docket No. 50-275** **(10 C.F.R. § 2.206)**
PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Unit 1)	**August 20, 1984**

The Director of Nuclear Reactor Regulation denies a series of petitions filed by the Government Accountability Project on behalf of the San Luis Obispo Mothers for Peace which requested deferral of decisions to issue low-power and full-power licenses for the Diablo Canyon Unit 1 facility until a series of specified actions were taken.

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

By petition pursuant to 10 C.F.R. § 2.206, dated February 2, 1984, Thomas Devine of the Government Accountability Project (GAP) on behalf of the San Luis Obispo Mothers for Peace requested that the Nuclear Regulatory Commission defer any decision on whether to grant a low-power operating license to the Diablo Canyon Nuclear Power Plant, Unit 1, until a number of specified actions were taken.[1] Notice of

[1] GAP's petition was filed before the Commission. It was referred to the NRC Staff for action, as were all subsequently filed petitions — supplemental documents dated March 1, March 23, April 12, May 3, June 21, June 22, July 11, July 16 and July 23, 1984. These petitions are addressed herein.

receipt of the petition was published in the *Federal Register* on March 13, 1984 (49 Fed. Reg. 9517).

PETITIONS REGARDING LOW-POWER OPERATION

The actions requested by GAP in its February petition included:

1. Completion of "a comprehensive, third-party reinspection program of all safety-related construction in the plant, with full authority by the independent organization to identify and impose corrective action on any nonconforming condition . . ."
2. "An independent audit of design quality assurance, including the reliability of conclusions from remedial design verification programs imposed since 1981 such as the seismic design review";
3. development of a full record on Pacific Gas & Electric Company's character and competence to operate the Diablo Canyon Plant, including
   a. a management audit by an independent organization,
   b. a full investigation and report by the NRC Office of Investigations to determine the causes of construction and design QA violations at Diablo Canyon, including questions of harassment, subordination of safety to cost concerns, destruction of records and deliberate violations of the Act;
4. a full program of public participation for selection and oversight of the independent organizations and creation of a public oversight committee with authority to obtain all requested information and to conduct legislative-style public oversight hearings.

In support of its request, GAP identified some 170 alleged violations of "legal requirements and relevant specifications," based upon the affidavits and supporting exhibits of six present or former employees at the Diablo Canyon site. The alleged violations involved breakdowns in both construction quality assurance and design quality assurance (QA).

In the construction area, a number of issues concerning the adequacy of welding were raised. These included problems with (1) qualifications of welders, welding procedures and welding inspectors; (2) control of welding equipment; (3) maintenance of welding material; (4) weld inspection program; and (5) weld repairs. Additional constructional problems were alleged in the areas of nondestructive examinations, hydrostatic tests of piping, vendor QA, generic breakdowns in material control, construction procedures and training for quality control (QC) inspectors, suspect inspection acceptance criteria, breakdowns in the system for disclosure of QA violations and in the organizational freedom of QC inspectors, harassment of and retaliation against QC personnel. In the area of design QA the petitioner described alleged violations in the areas of results from the seismic design review and design control. Allegations were also raised concerning design flaws in the residual heat removal system (RHR) of the emergency core cooling system (ECCS).

Finally, GAP asserted that even if specific safety hazards were not created or specific regulations violated, the factual pattern which they have described demonstrates that PG&E does not have the necessary character and competence to operate a nuclear power plant and that the allegations must be resolved prior to any low-power operating decision because they concern issues which could be grounds for denying the license.

GAP filed supplements to its petition, with additional allegations and supporting affidavits on March 1, 1984, March 23, 1984, and April 12, 1984. GAP was joined in its March 1, 1984, supplemental petition by six other organizations.[2] This petition submitted five additional affidavits and interviews with nine present and former plant workers. Additional, specific remedial actions were requested of the NRC based on this information. GAP requested that the reinspection of plant safety-related construction be preceded by a comprehensive review of all potential quality-related documentation, an expansion of the sample program in the seismic design review to cover 100% of relevant, safety-related installations and implementation of definitive corrective action to eliminate a design flaw in the RHR pumps at Diablo Canyon. The March 12, 1984, supplement provided twelve additional affidavits in support of previous allegations made in the February 2 and March 1 petitions.

On April 12, 1984, GAP filed a petition pursuant to 10 C.F.R. § 2.206 before the Commission alleging that the record before it for a decision on a low-power license was inaccurate and requesting (1) provision for the Joint Intervenors to brief the Commission along with the NRC Staff; (2) assumption of responsibility by the Commission to conduct further fact-finding and oversee ongoing corrective action; (3) direction to the NRC Staff to provide transcripts of "whistleblower" interviews to the Atomic Safety and Licensing Appeal Board, and (4) initiation of an

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[2] Christic Institute, Critical Mass, Environmental Action, Friends of the Earth, Fund for Constitutional Government, Greenpeace Pacific Southwest, and Nuclear Information and Resource Services. Their participation is limited to the March 1, 1984, petition.

investigation by the Office of Inspector and Auditor into certain actions by the NRC Staff.[3]

NRC STAFF EVALUATION OF ALLEGATIONS RE: DIABLO CANYON

During the course of the independent design reverification program at Diablo Canyon from 1982 through early 1984, the Commission began to receive allegations from a variety of sources concerning the design, construction and operation of the facility and the Licensee's management of these activities. As a result of the growing number of allegations, the Commission directed the Staff on October 28, 1983, to pursue all allegations and concerns to resolution and requested a status report on the investigation, inspection and evaluation efforts prior to its decision regarding authorization of criticality and low-power testing. In order to assure an adequate and coordinated response to all allegations received concerning the facility, the Staff developed th Diablo Canyon Allegation Management Program (DCAMP), set forth in a document dated November 23, 1983.

Briefly, DCAMP provides for a systematic examination and analysis of allegations and expressions of concern pertaining to design, construction, operation and management of safety-related structures, systems and components at the Diablo Canyon Plant. It provides for procedures to maintain confidentiality where requested, confirmation with the alleger where possible and appropriate and preliminary assessments of allegation significance and programmatic implications prior to Commission consideration of licensing actions. Resolution of allegations may involve site inspections, technical reviews, interviews with site personnel and public technical meetings.

The basic approach for each allegation was to determine if it represented significant new information which suggested that some safety-related structure, system or component necessary for safe operation would not perform its safety function, or whether it identified such weaknesses in Licensee's management or quality assurance that plant safety was called into serious question. The Staff applied the following criteria as set forth in SSER 22 for assessing which allegations and concerns required resolution prior to criticality and ascension above 5% power:

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[3] All allegations received in this petition and subsequent ones which dealt with alleged misconduct by the NRC Staff have been referred to the Office of Inspector and Auditor for handling.

1. Prior to criticality those allegations or concerns must be resolved which offer new information, not previously available to the Staff, and which appear to involve a discrepancy between design criteria, design, construction or operation of a safety-related component, system, or structure of such magnitude so as to cause the operability to be drawn into question. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the Staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to criticality.
2. Prior to criticality, those allegations or concerns must be resolved which offer definitive new information, not previously available to the Staff, and which indicate a potential, significant deficiency in the Licensee's management or quality assurance of safety-related activities. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the Staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to criticality.
3. Prior to exceeding 5% power, those allegations or concerns must be resolved which offer specific new information, not previously available to the Staff, and which may reasonably be expected to involve sizeable failures of systems that contain radioactivity or of the ECCS systems. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the Staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to exceeding 5% power.

In Supplements 21 and 22 to the Safety Evaluation Report for PG&E's application (copies of which are attached (not published)) for an operating license, the Staff reported on the status of its investigation and evaluation under DCAMP of 103 and 219 allegations, respectively, it had received as of December 1983 and March 9, 1984, excluding those received under the 2.206 petitions. The Staff concluded that none of these allegations required resolution prior to a reactor criticality decision, but that eighteen allegations relating to eight subject areas needed to be resolved prior to issuance of a full-power license.

At a Commission meeting on March 26, 1984, the Staff indicated that it had evaluated each allegation in sufficient detail contained in the February 2, 1984, and March 1, 1984, petitions to determine whether they were identical or similar to allegations already dealt with, whether they represented a slightly different twist on an issue already dealt with or whether they were totally new. Approximately 75% of the issues in the 2.206 petitions were found to have been already addressed by the Staff. The remaining items were totally new or contained insufficient information for review. The Staff reviewed the totally new issues against the criteria described above to determine whether resolution of the allegations was necessary prior to making a decision on permitting reactor criticality. The Staff concluded that none of these items met the criteria for an issue which should be resolved prior to a decision on criticality. This conclusion was confirmed by the Staff at the Commission meeting on April 13, 1984 (Tr. 44-45).

On April 13, 1984, the Commission voted to reinstate the operating license to conduct low-power tests up to 5% of rated power for the Diablo Canyon Unit 1 facility. CLI-84-5, 19 NRC 953 (1984). In that decision the Commission described the DCAMP and the criteria used to evaluate allegations to determine if final resolution was necessary prior to reinstatement of the license. The Commission concurred in the Staff's conclusions that none of the allegations received in the 2.206 petitions warranted immediate resolution and directed that evaluation of the allegations under DCAMP should continue both to document reviews completed to that time and to address those matters that need to be resolved prior to licensing at higher power levels.

In addition, the Commission reviewed the specific allegations and actions requested in GAP's April 12, 1984, petition. CLI-84-5, *supra*, 19 NRC at 962-63. It noted that GAP's allegations of false statements by the NRC Staff and PG&E were based for the most part on its own interpretations of the implications of various allegations and that other allegations were based on differences of opinion with members of the NRC Staff. Again, the Commission concluded that nothing in GAP's April 12th submittal required delay in reinstatement of the Diablo Canyon Unit 1 low-power license.

Thus, GAP's request that the specific actions as described above be taken prior to issuance of a low-power license has been denied by the Commission's decision to reinstate the low-power license. The NRC Staff concluded and the Commission agreed that evaluation and resolution of the allegations submitted by GAP in accordance with the DCAMP and the screening criteria are appropriate and sufficient methods for determining that the Commission has reasonable assurance that the Diablo Canyon facility can be operated at low power, and ultimately full power, without undue risk to the public health and safety.

The NRC Staff did conclude that certain issues must be satisfactorily resolved before Diablo Canyon could be permitted to operate above 5% power. One of the issues related to the adequacy of small-bore piping and piping supports which also encompassed some allegations submitted with the GAP petitions.

On April 18, 1984, an Order Modifying License was issued to PG&E requiring completion of specific actions related to piping and supports before the Licensee would be permitted to operate above 5% power. 49 Fed. Reg. 18,202 (April 27, 1984).

PETITIONS REGARDING FULL-POWER OPERATION

On May 3, 1984, GAP filed a new petition on behalf of the San Luis Obispo Mothers for Peace requesting the Commission to defer any decision to permit the Diablo Canyon facility to go above 5% power until after "successful completion" of certain specified actions. These actions consist of:

1. appointment and implementation by an independent third party of corrective action required by the April 18th Order;
2. a comprehensive review of all "Pipe Support Design Tolerance Clarification Program" activities;
3. full public participation in selection and oversight of independent organizations to carry out the first two items;
4. publication of a Construction Assessment Team (CAT) report by non-Region V personnel and people not previously assigned to Diablo Canyon;
5. development of a full record on the character and competence of PG&E based on a management audit, reports of the NRC Office of Investigations and records of Department of Labor hearings;
6. Board Notification of transcripts of whistleblowers; and
7. investigation by the Office of Inspector and Auditor of alleged false statements by the NRC Staff.

As the basis for its request, GAP adopted by reference all the affidavits submitted in its earlier petitions described above. They asserted that the information had not been "seriously reviewed, let alone resolved." They also based their petition on transcripts of "witness" interviews taken since April 3, 1984, draft reports on Diablo Canyon by NRC inspector Yin, and six additional affidavits by a GAP representative and four current and former plant employees. In brief, these various documents allege a widespread breakdown in quality assurance for design of large- and small-bore piping, and that PG&E has demonstrated such a lack of concern in this area through its practices at the plant and false and misleading statements to the Commission prior to the low-power licensing vote that the Commission should not rely on PG&E review and corrective actions for these problems. GAP also asserted that there is a widespread construction quality assurance breakdown as revealed by Pullman Power Products' (a contractor) guidance documents, safety-related bolting and reactor coolant system welds and piping. Finally, GAP expressed dissatisfaction with the role of NRC Staff, particularly that of regional Staff, in reviewing alleged deficiencies and corrective actions at the Diablo Canyon facility.

On June 21, 1984, GAP submitted additional allegations based upon seventeen additional witness statements in support of the May 3 petition.[4] These statements alleged a breakdown in the reporting system for QA violations due to a campaign by management to get inspectors to stop using the formal reporting system, and not write up problems on "old work"; ineffective reinspection and corrective actions including those for cracked welds in the Component Cooling Water System (CCW); poor-quality materials and inadequate hydrostatic tests of piping.

The statements also include allegations of false statements and records falsification by PG&E, increasing reprisals and harassment on site as well as inadequate corrective actions, changing plant design through memoranda, inaccurate drawings and undersized weld design. GAP again expressed dissatisfaction with the manner in which the NRC Staff has been handling its allegations of QA breakdown and "coverup" by PG&E.

On July 16, 1984, GAP filed an additional petition before the Commission requesting that a number of steps be taken before any commercial licensing decision on the Diablo Canyon Plant. The actions requested, including providing "sufficient organizational freedom" to NRC inspector Yin, appointment of an organization other than the Advisory Committee on Reactor Safeguards (ACRS) to review the work by Mr. Yin and other NRC Staff, expansion of the NRC internal investigation into false statements by the Staff, provision of a forum to resolve the various allegations submitted by GAP, a briefing by the Office of Investigations on PG&E's character and competence and an explanation of why some "6000 licensing commitments" have been postponed for the

[4] Six of the witness statements were provided only to the Office of Investigations which subsequently provided them to the NRC Staff.

Diablo Canyon facility. Two additional affidavits, including one by GAP counsel, were submitted with the petition.

DISCUSSION

The NRC Staff has continued to examine all allegations concerning the Diablo Canyon facility received from GAP in its 2.206 petitions (and from elsewhere). All allegations are assessed against the screening criteria described above to determine which allegations required resolution prior to full-power operation.

As stated in Supplement 26 to the Safety Evaluation Report, as of July 8, 1984, 1404 allegations have been received, although many are duplicates or variations on previous allegations. For tracking purposes each allegation received has been assigned a number. To date, 581 of all allegations are resolved and documented. Additionally, approximately 300 have been resolved and are in the process of being documented. The remaining allegations are as yet unresolved.

The allegations have been and continue to be resolved by methods appropriate for the individual allegation. Certain allegations have been assigned for resolution by NRC's Region V office, others to the Office of Nuclear Reactor Regulation.[5] Following appropriate screening by the Staff, a number of allegations have been submitted to the Licensee for evaluation. The Licensee has been required to provide the results of its evaluations and identify any necessary corrective actions to the Staff in writing. The subsequent Staff evaluation of an allegation then also considers the Licensee's response and action. As of July 1, 1984, 177 allegations have been handled in this manner. While thirty-one require additional Staff or Licensee action, none indicate a problem, individually or collectively, sufficient to preclude power ascension or full-power authorization.

All allegations received from GAP have been evaluated against the screening criteria. SER Supplement 26 (which is attached) presents the resolution of those allegations which the Staff has determined in accordance with the screening criteria must be resolved prior to power ascension and full-power operation.

These allegations relate to the following subject areas: (1) operational limit for CCW system; (2) replacement of welded high-strength bolts;

[5] The Office of Investigations has inquiries pending on a number of allegations involving, among other thing., potential false statements and personnel matters. Twenty-two of 121 allegations before OI are resolved. In the Staff's view, those remaining did not require resolution prior to full-power operation.

(3) as-built drawings for operations; (4) completion of systems interaction program and modifications; (5) evaluation of coating concerns; (6) piping and supports and related design issues; (7) RHR low-flow alarm; (8) bolted connections. The issues concerning piping and piping supports were the subject of a special NRC Peer Review Group. The review in the Spring of 1984 resulted in seven license conditions requiring certain actions before operation above 5% power. The Review Group has examined the Licensee's actions regarding the license conditions by means of system walkdowns and onsite inspections, audits and meetings with the Licensee. A draft report by the Review Group was issued on July 13th, which found that these issues should not prevent operation of Diablo Canyon Unit 1 at full power. The final report, after consideration of comments by NRC Inspector Yin and the ACRS, has been issued as SSER 25. A copy is attached. The various allegations received from GAP as part of its 2.206 petitions and in meetings and interviews on this subject have been specifically reviewed to determine if the Staff's evaluation efforts have adequately considered the concerns expressed. The Staff has concluded that none of the allegations require any further evaluation prior to full-power operation of Unit 1.

GAP's July 16th petition described a number of steps which it believes the Commission should take before any licensing decision on the Diablo Canyon Plant. As indicated above, the Staff has concluded that no substantive issues remain unresolved which would preclude the requisite safety findings for issuance of a full-power license for Diablo Canyon Unit 1 at this time. The Diablo Canyon Review Group has concluded that the seven license conditions to be met before full-power operation, which arose out of Mr. Yin's concerns, have been satisfied. The ACRS in its letter to the Commission dated July 16, 1984, has concurred in these Staff findings. (A copy of the letter is included in SSER 25.) With respect to GAP's request for a public forum to address material disputes of fact, it has been clearly established that the holding of hearings in response to the filing of a 2.206 request is not required. *Porter County Chapter of the Izaak Walton League v. NRC*, 606 F.2d 1363 (D.C. Cir. 1979); *Illinois v. NRC*, 591 F.2d 12 (7th Cir. 1979). In any event, as explained above and in the Staff's SSERs, we have concluded that there are no substantial safety issues remaining that would justify the initiation of a proceeding that would provide an opportunity for a hearing.

With respect to GAP's request for a Staff report regarding "postponement for approximately a year of PG&E compliance with some 6000 licensing commitments," the Staff concludes that GAP has not provided any adequate basis for such a request. The matter of "6000 licensing commitments" was discussed at an NRC meeting with the Licensee on

July 2, 1984. A transcript of the meeting was issued on July 11, 1984, as Board Notification 84-128. At the meeting the Licensee informed the Staff that a computerized quality commitments management data base is being developed for internal use to track those commitments that are to be met throughout the life of the plant. At the time of the meeting the Licensee had identified approximately 6000 such quality commitments. As explained further on pages 104-06 of the meeting transcript and based on further discussions by the Staff with the Licensee, the data base will be routinely checked to assure that commitments are being met on their prescribed schedule. The data base will be updated to include new commitments.

It is the Staff's understanding that the two specific examples cited in GAP Exhibit 2 are not included in this commitment list because they did not exist at the time of the meeting, because they are specific commitments to be met only once at a specific time and because they are not directly quality program related.

As indicated on page 105 of the transcript, the Staff has concluded that the Licensee's commitments are to be met at the times specified for such commitments and that no extensions of such commitment dates will be given without proper justification. The NRC has not waived at any time the requirements for any Diablo Canyon commitment, quality related or other, without proper bases.

Finally, Exhibit 2 at page 6 implies that the 6000 line items in the program necessitate repairs. While some of these items relate to specific systems, structures and components, many of them relate to administrative and personnel matters such as training and qualification, reporting, exercises and tests as set forth in the Technical Specifications. The need for repairs resulting from the 6000 line items is expected to be rare.

In summary, the "6000 license commitments" is not a list of open items but rather a tracking system for license commitments to be met throughout the life of the plant. As stated in recent SSERs, in particular SSER 27, the Staff has evaluated those license commitments that must be met prior to issuance of a full-power license amendment and has concluded they have been met.

CONCLUSION

The petitioner bases its request for relief on numerous allegations of inadequate quality assurance in design and construction; construction defects and harassment and intimidation of QA/QC personnel. As discussed above, the NRC has established a program to screen and to evaluate the safety significance and to resolve these allegations and has since

1983 spent thousands of hours under that program investigating, inspecting and evaluating the concerns raised. In the Staff's view, no issues remain unresolved which indicate problems of such a magnitude, either individually or collectively, that preclude authorization for power ascension testing and full-power operation. Therefore, petitioner's request for specific actions to be taken prior to a decision on full-power operation of the Diablo Canyon Unit 1 facility is *denied.* A copy of the Decision will be filed with the Secretary for the Commission's review in accordance with 10 C.F.R. § 2.206(c).

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland,
this 20th day of August 1984.

Attachments: Supplemental Safety Evaluation
Report Nos. 21, 22, 25 & 26

[The Attachments have been omitted from this publication but may be found in the NRC Public Document Room, 1717 H Street, NW, Washington, DC 20555.]

Cite as 20 NRC 788 (1984) **DD-84-21**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

Harold R. Denton, Director

In the Matter of | **Docket No. 50-416**
(10 C.F.R. § 2.206)

MISSISSIPPI POWER & LIGHT COMPANY
MIDDLE SOUTH ENERGY, INC.
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION
(Grand Gulf Nuclear Station, Unit 1) | **August 31, 1984**

The Director of the Office of Nuclear Reactor Regulation denies a request by Cynthia Stewart on behalf of Jacksonians United for Livable Energy Policies that the Commission take action with respect to the Grand Gulf Nuclear Station, Unit 1.

TECHNICAL ISSUE DISCUSSED: OPERATOR QUALIFICATION

The Commission has accepted industry criteria for evaluating the adequacy of on-shift operating experience for near-term operating license applicants.

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

INTRODUCTION

By Petition dated March 29, 1984, Cynthia Stewart, on behalf of Jacksonians United for Livable Energy Policies (hereinafter referred to as JULEP or the Petitioner), requested that the Nuclear Regulatory Commission issue an order to Mississippi Power & Light Company (hereinafter referred to as MP&L) to show cause why the low-power operating license for the Grand Gulf Nuclear Station, Unit 1, should not be revoked and a stay of operation should not be issued. Notwithstanding its request for license revocation, the Petitioner also requested that the operating license be modified to remove management personnel responsible for past problems at Grand Gulf and to ensure implementation and verification of corrective actions associated with Technical Specification discrepancies and other deviations from NRC requirements. Additionally, the Petitioner requests hearings before an Atomic Safety and Licensing Board.[1] As grounds for granting this relief, the Petitioner asserts the following: (1) the Technical Specifications issued for the plant were, and continue to be, erroneous; (2) operator qualifications were falsified; (3) the drywell cooling system was inadequately designed and constructed; (4) the electric power system is inadequate; (5) MP&L had no previous nuclear experience and until recently none of the staff had operated a commercial reactor; and (6) given the history of problems and consistent poor management performance of the Licensee, NRC will be unable to assure compliance by the Licensee with NRC requirements. In accordance with usual NRC practice, the Petition was referred to the Staff for appropriate action in accordance with 10 C.F.R. § 2.206. A notice was published that the Petition was under consideration. 49 Fed. Reg. 22,168 (May 25, 1984).

On May 30, 1984, pursuant to 10 C.F.R. § 50.54(f) and § 182 of the Atomic Energy Act, MP&L was requested to respond to the Petition. On July 5, 1984, MP&L filed its response. As explained in this Decision, a number of actions have been taken to ensure implementation and verification of corrective actions for identified problems at Grand Gulf. In view of these actions, the Staff does not believe that institution of further proceedings to modify or revoke the Grand Gulf license is

[1] The Petitioner also asked for appointment of an "independent panel" to inquire into the propriety and effectiveness of NRC personnel's actions related to Grand Gulf. Although this request is beyond the scope of relief normally contemplated under 10 C.F.R. § 2.206, a copy of the petition was provided to the Commission's Office of Inspector and Auditor for appropriate action.

warranted. Accordingly, I have concluded that Petitioner's request should be denied.

DISCUSSION

Discrepancies in Technical Specifications and Surveillance Procedures

A brief historical review is helpful at this point to place the Petitioner's assertions in proper perspective. On June 16, 1982, a low-power license was issued for the Grand Gulf Nuclear Station, Unit 1. Inspections by Region II with regard to compliance of surveillance procedures with the Technical Specifications were performed from June 16, 1982, to October 8, 1982, and discrepancies in the surveillance procedures and Technical Specifications were identified. *See* Inspection Reports 50-416/82-55, 50-416/82-58, 50-416/82-60, 50-416/82-65, and 50-416/82-67. Based on these inspections, a Confirmation of Action letter was issued on October 20, 1982, confirming the Licensee's commitment to restrict the next criticality (the plant was then shut down for other reasons) until the identified discrepancies were resolved. At the conclusion of this phase of the Licensees' review, in late August 1983, another inspection was held to evaluate operational readiness. *See* Inspection Report 50-416/83-38. The plant returned to criticality on September 25, 1983, and low-power tests were conducted until November 8, 1983. The plant was shut down after completion of testing which was followed by an extensive licensed operator recertification program, during which time MP&L and the Staff again reviewed the Technical Specifications as issued through Amendment No. 12 to the Operating License. Further problem areas were identified, resulting in a complete review of the Technical Specifications by MP&L beginning on March 2, 1984. This review was completed in April 1984. As a result of these actions, Technical Specification problem areas were identified by MP&L. The Staff determined that changes to the Technical Specifications needed to be made. The Staff performed a safety evaluation in order to determine which changes were required for 5% power operation.[2] On April 18, 1984, the Director of the Office of Nuclear Reactor Regulation issued an Order Restricting Conditions for Operation, effective immediately, which provided:

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[2] The Staff determined that operation at a power level of up to 5% power did not require all problems with the Technical Specifications to be resolved at that time.

> MP&L shall not operate the Grand Gulf plant under the terms of License No. NPF-13 unless such operation is in conformance with the revised Technical Specifications appended to this Order and MP&L, prior to entry into mode 2, certifies to the Regional Administrator, Region II, that MP&L's procedures have been modified and training [illegible] to reflect the revised Technical Specifications.

49 Fed. Reg. 17,832, 17,833 (April 25, 1984).

This order implemented some twenty-three changes in the Technical Specifications that were required to restore the safety margins for low-power operation. Resolution of the remaining items identified by the NRC Staff and MP&L has been accomplished. *See* Supplement No. 6 to the Safety Evaluation Report. MP&L has submitted amendment requests to the Technical Specifications in order to make these changes. These Technical Specifications have been included in the amendment authorizing full-power operation. Operation in accordance with the amended Technical Specifications provides reasonable assurance that the plant can be operated at full power with no significant hazard to the health and safety of the public.

Falsification of Operator Qualifications

Discrepancies in the documentation of operator training were identified during a special training inspection conducted in February 1983 and a special safety inspection conducted by Region II during August, October and November 1983. *See* Inspection Reports 50-416/83-06, 50-416/83-38, 50-416/83-53. An evaluation of these inspections by Region II concluded that these discrepancies were not limited to documentation errors and that some information submitted to the NRC on applications for operators' licenses was inaccurate. At Region II's request, an investigation was conducted by the Office of Investigations (OI) from October 18, 1983, through February 10, 1984.

To ensure that the individuals granted licenses had the requisite qualifications to retain their licenses, the Staff has taken a number of actions. During the week of October 31, 1983, Region II conducted a second training assessment inspection. This inspection was to follow up on problems identified during the February 1983 assessment with particular attention to the training of licensed operators. In this assessment, Region II conducted walkthrough-type evaluations on selected systems for thirteen licensed operators. These operators were identified as being deficient in knowledge level and were removed from licensed duties by MP&L.

The issuance of further licenses at Grand Gulf was suspended. In the Staff's judgment, the indicated weaknesses in operator training were not of such significance to warrant revocation of the licenses.

On November 11 and November 18, 1983, Region II and MP&L met concerning a recertification program to be conducted for all licensed operators, shift advisors and shift technical advisors. MP&L agreed to implement an extensive program to recertify licensed operators which would include areas of identified weaknesses. This special program began in November 1983 and was essentially completed in February 1984. The recertification program included an individual examination of each licensed operator on each of sixty-eight systems listed on a licensed operator qualification card acceptable to Region II. These examinations were monitored by MP&L, representatives of two other utilities, by the nuclear steam supply vendor (General Electric), and by NRC. At the completion of this examination process, the records of the operators were reviewed by a Grand Gulf Operator Training Evaluation Committee (OTEC) consisting of representatives of plant management. The Committee examined operator training records and the results of the examinations and conducted additional oral examinations as necessary. Out of twenty-seven individuals examined by the Committee, one was found to be unqualified and was removed from licensed duties. The NRC conducted an independent recertification examination of these twenty-six individuals. The results of the independent NRC recertification examination were that twenty-three of the twenty-six operators passed. The remaining three who failed were removed from licensed duties. Following retraining, these three were reexamined by the NRC. Two passed, and one who failed the reexamination is no longer employed at Grand Gulf.

Region II also examined the training and qualification of shift technical advisors. The training was reviewed against FSAR commitments, and previous exams were reviewed for weak areas. Retraining was provided by the utility to strengthen weak areas, and exams were given. The exams were prepared and administered by the utility and reviewed by NRC examiners. OTEC reviewed the training and exam records, gave each advisor an oral exam, and recertified the shift technical advisors.

These actions to review operator qualifications provide reasonable assurance that the operating staff at Grand Gulf have met the NRC requirements for training and obtaining a license. While revocation of the Grand Gulf operating license is not warranted, enforcement action will be taken with regard to the applications for operators' licenses.

Lack of Experience of MP&L Staff

Petitioner raises as an issue the lack of operator experience similar to Diablo Canyon[3] and the inexperience of MP&L as an operator of nuclear facilities. Improvements in MP&L's management are discussed in the latter portion of this Decision. With respect to operator experience, the Commission has expressed similar concern about the limited prior operating experience possessed by members of the operating shifts at certain plants including Grand Gulf. An industry working group was formed to respond to this concern. The working group developed proposed criteria for shift operating experience and presented these criteria to the Commission in February of this year. These criteria require the four operators on each shift to possess at least 13 years of power plant experience, at least 6 of which must be nuclear. Weighting factors are used in assessing experience. The criteria further require at least one senior reactor operator with 6 months "hot" participation at the same type plant on each shift or a qualified shift advisor until such time as the plant meets this participation requirement. With a few improvements, the industry criteria were recently accepted by the Commission. *See* Generic Letter 84-16, "Adequacy of On-Shift Operating Experience for Near-Term Operating License Applicants" (June 27, 1984). Region II has conducted an assessment and has concluded that the operating experience at Grand Gulf exceeds the Commission-approved criteria.

During the startup phase, Grand Gulf has enhanced operating experience by use of contract personnel in an advisory capacity. In addition to the normal shift technical advisor, a nuclear shift advisor has been assigned to each shift to participate in shift training. The shift advisors previously held senior reactor operator licenses at other BWR facilities. Although they do not hold licenses at Grand Gulf, they have been certified by MP&L, and a specific training program was developed to provide each of these individuals with training on the differences between Grand Gulf, a BWR-6, and earlier boiling water reactor designs. This program involved training lectures on plant systems, procedures and Technical Specifications followed by an OTEC examination. Each shift advisor also received 2 weeks of simulator training in power ascension and emergency operating procedures and an examination on that training.

The operating staff at Grand Gulf has gained experience in systems operations and surveillance testing during the low-power testing program. These activities were monitored by NRC inspectors. The

[3] *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Unit 1), CLI-84-5, 19 NRC 953, 960-61 (1984).

planned, deliberate power ascension program will add to this experience. Region II Staff has conducted broad-based operational readiness inspections at Grand Gulf prior to the plant's exceeding 5% power and will inspect again prior to its exceeding 50% power.

Design and Construction of the Drywell Cooling System

Petitioner alleges that the drywell cooling system was inadequately designed and constructed. Grand Gulf Nuclear Station, Unit 1, is the first of the General Electric BWR-6/Mark-III reactor/containment designs to be built in the United States. As such, a new product line and a prototype reactor sometimes experience some difficulties in going from design to actual operation. In fact, the purpose of preoperational testing and the startup test program is to identify anomalies during the initial phases of operation. An inadequacy in the performance characteristics of the drywell cooling system was identified during the nonnuclear heatup as part of the preoperational testing. The problem resulted from inadequate insulation which led to higher-than-anticipated heat losses to the drywell. MP&L solved this problem by making several modifications to the plant. The modifications included repairs and rework to existing reflective insulation, the addition of insulation in certain areas, modifications and additions to the air distribution systems, and the addition of a 1200-ton chiller capacity to the drywell cooling. With these modifications, MP&L is capable of meeting the requirement in the Technical Specifications to limit the temperature in the drywell to 135°F to protect the structure and safety-related equipment. While the operational mode of the plant was restricted by this deficiency, the anomaly did not pose a risk to the public health and safety.

Adequacy of the Electric Power System

On August 12, 1983, the main crankshaft on one of the three emergency diesel generators at the Shoreham Nuclear Power Station, which were manufactured by Transamerica Delaval, Inc. (TDI), broke during a load test. During the course of the evaluation of the failure, information related to the operating history of TDI engines was identified which called into question the reliability of all TDI diesels, including the TDI diesel generators installed at the Grand Gulf facility. As a result, an Owners Group was organized with all plants utilizing TDI engines in order to resolve this problem.

Subsequently, NRC Staff conducted an evaluation of the effect of failure of TDI diesel generators at Grand Gulf at the maximum power level of 5% then authorized by the license. The Staff concluded that the total failure of the Delaval diesels at Grand Gulf would not significantly increase the risk of low-power operation and that the risk of low-power operation was acceptably small. Nevertheless, some very low probability environmental events were contributors to that risk, and that risk would be reduced if the reliability of the TDI diesel generator is enhanced. Consequently, the Staff determined that it would be appropriate to have increased assurance as to reliable onsite power. Moreover, for full-power operation, a high degree of reliability is required for the diesel generators. The Staff found that the most appropriate method to obtain information about the specific conditions of the diesel generators at Grand Gulf would be to disassemble and inspect the diesel generator which had been operating the longest. In view of these findings, the Director of NRR issued an Order Requiring Diesel Generator Inspection, effective immediately, on May 22, 1984, which provided that the Division I TDI diesel generator be disassembled for inspection, all defective parts be replaced prior to declaring the engine operable (the engine block and engine base could be excepted if indications were not significant), and that preoperational testing be performed on the inspected engine prior to declaring it operable. 49 Fed. Reg. 22,582 (May 30, 1984).

MP&L has completed the teardown and inspection as required by the Order. The only significant finding involved the failure of some capscrews in the turbocharger. Subsequently, the turbochargers for both diesels were refurbished by the manufacturer. As a result of its review of the diesel generator issue, the NRC Staff has concluded that the TDI diesel engines at Grand Gulf will provide a reliable standby source of onsite power. This finding is based upon the reviews of (1) the current status of the TDI Owners Group Program in resolving the TDI diesel engine issue; (2) actions taken by MP&L to verify the reliability of the Division I and II engines, including those actions taken in response to the NRC Order dated May 22, 1984; (3) the Augmented Engine Maintenance and Surveillance Program to which MP&L committed in letters dated July 20 and 22, 1984; and (4) changes to the Technical Specifications to limit future testing of the engines to 185 psig brake mean effective pressure. In addition, certain license conditions have been imposed to provide future assurance that the diesel generators will be acceptable at Grand Gulf. The results of the Staff's review and the basis for its findings are contained in Supplement No. 6 to the Safety Evaluation Report for Grand Gulf. Certain exemptions have been issued with respect to the onsite power supply, but the Staff believes that full-power operation with the exemptions will not pose an undue risk to public health and safety. These exemptions are not related to the TDI diesel engine performance.

Since the Licensee's inspection and the NRC Staff's review, on July 26, 1984, a cylinder head on the Division I diesel was found to be leaking water into the cylinder from the jacket water cooling system. The leak was found during a surveillance check specifically intended to identify such leakage. The source of the leak was identified as a crack located in a region of the head which had not previously been identified as a potential problem area and which has not been subject to the inspections performed under the Order. The leaking cylinder head has been replaced. Surveillance checks for cylinder head leakage will be performed periodically while the engines are in an operable standby mode. The Staff believes that this surveillance provides adequate assurance that any future leaks will not impair the operability of the engines and that this event does not modify the Staff findings as stated above.

Assurance That Licensee Will Meet NRC Requirements

The Petitioner argues that, in view of the Licensee's past difficulties in meeting NRC requirements and the consistent poor performance of the Licensee's management, NRC can have no assurance that the Licensee will operate the facility competently in the future. In support of this charge, the Petitioner cites the Licensee's failures to meet regulations in the case of employee training, the discrepancies between the physical plant and technical specifications, and the fact that, in the NRC's annual Systematic Assessment of Licensee Performance (SALP) reviews, MP&L management has consistently scored poorly.

To put this issue in perspective, it is important to consider that, beginning on October 1, 1983, a new organizational structure was implemented at the Grand Gulf site. The major thrust was to establish more managerial control over plant operations. Three parallel assistant plant managers reporting to the plant manager were established. One has the operations superintendent and the operating crews reporting to him along with health physics and chemistry. One has all maintenance personnel reporting to him. The third has training and security reporting to him, along with various administrative functions. Management changes within the training area included elevating the training function to report directly to an assistant plant manager, consolidating the training staff, assigning additional personnel to the training department, initiation of a special financial incentive program to improve the staff retention rate, and the addition of a corporate nuclear human resource manager directly responsible for increasing the number and level of competence of personnel entering the training program.

A number of other management and personnel changes have also been made in an effort to augment the previous limited commercial nuclear experience available on the Grand Gulf staff. The former assistant plant manager for operations was made plant manager. His experience includes service as an instrumentation engineer at TVA's Browns Ferry facility, assistant plant manager at Watts Bar and assistant plant manager at Sequoyah. A former Director of nuclear power for TVA who has extensive nuclear experience, particularly in the operation and maintenance of BWRs, has assumed the position of Technical Advisor to the Vice President. It is also important to note that MP&L has a new President who has direct experience in the operation of commercial nuclear power plants and a new Senior Vice President with extensive nuclear Navy and corporate experience.

These changes represent a significant improvement in the experience and capability of the Licensee's management. Moreover, as noted in earlier portions of this Decision, appropriate measures have been taken to review deficiencies in operator qualifications and plant Technical Specifications and to ensure appropriate remedial action. In connection with the resolution of these issues, MP&L management has demonstrated marked improvement in its control of licensed activities. Thus the Staff believes that Grand Gulf can be operated in compliance with the Commission's requirements and with reasonable assurance that the health and safety of the public will not be endangered.

CONCLUSION

The Petitioner bases its request for relief upon past difficulties with the Grand Gulf facility. As discussed above, NRC has taken actions to resolve these difficulties. The Staff believes that the actions taken with regard to these problems are sufficient to provide reasonable assurance of the safe operation of the plant. Therefore, Petitioner's request for revocation of the license for Grand Gulf Nuclear Station, Unit 1, is denied. As the Commission has taken actions to resolve the problems with the diesel generators, Technical Specification discrepancies, and falsification of operator qualifications and because management has changed, Petitioner's requests to replace management and ensure implementation and verification of corrective actions for identified problems at Grand Gulf have been essentially satisfied. However, institution of further proceedings to implement these actions is unnecessary.[4]

A copy of this Decision will be filed with the Secretary for the Commission's review in accordance with 10 C.F.R. § 2.206(c) of the Commission's regulations.

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Dated at Bethesda, Maryland,
this 31st day of August 1984.

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[4] Consequently, the Petitioner's request for initiation of hearings before an Atomic Safety and Licensing Board is also denied. The holding of hearings on the Petitioner's § 2.206 request is not required. *Porter County Chapter of the Izaak Walton League v. NRC*, 606 F.2d 1363 (D.C. Cir. 1979); *Illinois v. NRC*, 591 F.2d 12 (7th Cir. 1979). Because appropriate actions have been taken or will be taken in connection with the authorization of a full-power license, initiation of further enforcement proceedings, which might result in the holding of the adjudicatory hearings that the Petitioner requests, is not warranted.