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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
James P. Gleason, Chairman
Dr. Jerry R. Kline
Mr. Glenn O. Bright

In the Matter of	Docket Nos. 50-440-OL 50-441-OL
CLEVELAND ELECTRIC ILLUMINATING COMPANY, <u>et al</u>.	ASLBP No. 81-457-04 OL
(Perry Nuclear Power Plant, Units 1 & 2)	September 3, 1985

CONCLUDING PARTIAL INITIAL DECISION ON EMERGENCY PLANNING, HYDROGEN CONTROL AND DIESEL GENERATORS

<u>Appearances</u>

On behalf of the Cleveland Electric Illuminating Company, <u>et al</u>., Applicants: Jay E. Silberg, Esq., Harry H. Glasspiegle, Esq., and Rose Ann C. Sullivan, Esq.

On behalf of the United States Nuclear Regulatory Commission: Colleen Woodhead, Esq.

On behalf of the United States Federal Emergency Management Agency: Brian P. Cassidy, Esq.

On behalf of Ohio Citizens for Responsible Energy, Intervenor: Susan L. Hiatt

On behalf of the Sunflower Alliance, Inc., <u>et al</u>., Intervenors: Terry J. Lodge, Esq.

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I. OPINION

This decision concerns an application for a license to operate a nuclear power plant with two boiling water reactors, Units 1 and 2, each rated at 1625 megawatts, and located at the Applicants' Perry plant site in Lake County, Ohio. The Applicants (Cleveland Electric Illuminating Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company and Toledo Edison Company) received permits to construct the facility in 1977, which is on Lake Erie approximately 35 miles northeast of Cleveland,

In this proceeding, held pursuant to the Atomic Energy Act of 1954, as amended, the parties in addition to Applicants and the NRC Staff, are the Ohio Citizens for Responsible Energy (OCRE), and Sunflower Alliance Inc. et al. (Sunflower). Admitted as government representatives under 10 CFR § 2.715(c) of the Commission's rules, were the Lake County Board of Commissioners, Lake County Disaster Services Agency, the Ashtabula County Commissioners and Ashtabula County Disaster Services Agency.

During a prehearing phase, the Board admitted 16 issues (contentions) to the proceeding. Twelve contentions were dismissed as a result of motions filed for summary disposition or pursuant to Commission rule-making or policy statements.[1] The Board has previously

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1 LBP-82-114, 16 NRC 1909 (1982); LBP-82-119, 16 NRC 2063; LBP-83-18, 17 NRC 501 (1983); LBP-83-48, 18 NRC 218 (1983); LBP-84-40, 20 NRC 1181 (1984); Memoranda and Orders, February 27, March 13, April 9, 1985 (unpublished). Also Order of April 28, 1982 (unpublished);
(Footnote Continued)

issued a partial initial decision on a quality assurance contention in favor of Applicants. This has been affirmed by the Appeal Board. (ALAB-802, 21 NRC 490 (1985)).

Contentions were litigated during hearings on April 9-12, 1985 (emergency planning and diesel generator reliability issues) and April 30-May 3, 1985 (hydrogen control). The Applicants and Staff submitted proposed findings and conclusions in the form of partial initial decisions as has OCRE and Sunflower on their particular issues. Our decision here resolves those three contentions remaining. Limited appearance opportunities were provided nonparty members of the public during both phases of the hearing proceedings.

The decisional record of the proceeding consists of the Commission's Notice of Hearing (46 Fed. Reg. 12372), petitions and findings filed by the parties, transcripts of the hearing and the exhibits received into evidence. In preparation of this decision, the entire record has been reviewed and considered. The proposed findings of fact and conclusions of law that are not incorporated directly or by reference in this initial decision are considered to be unsupported by the record of the case or as being unnecessary to the rendering of this decision.

(Footnote Continued)
LBP-82-83A, 16 NRC 208 (1982). "Contention" and "Issue" are used interchangeably in this Opinion.

The Board's jurisdiction is limited in this proceeding to issues placed in contention by the parties and to those concerns where the Board has found a serious safety, environmental or common defense and security matter exists. The Board has made no additional determinations of this nature. (10 CFR § 2.760(a)).

A. EMERGENCY PLANNING (Issue 1)

This issue was admitted by the Board prior to the development of offsite emergency plans and expressed a general deficiency in the adequacy of preparedness plans. After emergency plans became available, Sunflower was required to specify the inadequacies alleged to exist. Seven contentions and parts of two others, of 18 specified, remained for litigation after Board rulings on motions for summary disposition. During the hearing, testimony was provided by witnesses for the Applicants, Staff, Federal Emergency Management Agency (FEMA), and Sunflower.

1. Contention A: "Evacuation time estimates (ETE) have not been reviewed by State or local organizations."[2]

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2 Applicants' witness: Scott T. McCandless, V.P., HMM Associates, Inc.; Staff witness: Donald J. Perrotti, NRC Emergency Prep. Specialist; FEMA witness: Robert O. Shapiro, Emergency Management Specialist.

The Intervenors, Sunflower, cite herein an inadequacy in Applicants' emergency plan on grounds that State and local officials had not reviewed time estimates proposed for evacuation during an emergency.

The adequacy of State, local and Applicants' emergency plans is evaluated by guidance and criteria imposed by a joint NRC/FEMA document, NUREG-0654 FEMA-REP-1 Rev.1. The time estimates for evacuation within a plume exposure zone are recommended to be in accordance with Appendix 4 of that document. The criteria call for the ETE draft to be reviewed by State and local organizations and comments received from such reviews are to be included in the plans when submitted for evaluation and appraisal to the NRC. (Finding 1).

The evacuation time estimate study for the Perry emergency planning zone (EPZ) was performed by HMM Associates, Inc., a firm that developed ETEs for other nuclear power facilities: many of these were for plants with EPZs similar to Perry's. (Finding 2).

The Applicants' testimony indicated that officials from the three counties located within the EPZ were, in fact, consulted concerning the study and concurred with its scope. These officials, who were County Directors of Disaster Service Agencies and County sheriffs, and the cognizant State Agency were subsequently forwarded copies of the draft ETE. (Findings 3-4). The evidence reflects, contrary to allegations of the contention, that comments from State and local officials were included in a February 1985 revision of the Perry emergency plan and this document was forwarded to the Nuclear Regulatory Commission on February 20, 1985. (Findings 5-6).

The Intervenors directed much of their cross-examination on an alleged failure by HMM Associates to consult with and receive comments on the ETE from County engineers. In Intervenors' view, the holders of those offices should have been solicited for comments since County engineers have detailed responsibilities during an evacuation incident: these include road repairs, providing traffic control apparatus, dispatching school buses for carless population and similar duties. (McCandless, Tr. 2795-2802; Shapiro, Tr. 3122-27).

Applicants' witness testified that in preparing ETEs, coordination work is accomplished with principal local offsite emergency response officials and their assistance is requested--as it was at Perry--to identify any other local officials to be involved. This approach has been accepted by FEMA. The guidance from NUREG-0654 does not indicate what individuals or offices within State and local organization should review and comment on the proposed ETEs, but the intent is to have knowledgeable officials perform these tasks. (Findings 7-8).

The Board concludes that Applicants have complied with the guidance and criteria of NUREG-0654, Appendix 4, in the preparation and review of its ETE study as there has been no evidence that the interests of State and local governments have been ignored: the opposite has been the case. (Finding 9). Testimony by Applicants' witness indicated that, in response to Sunflower's concerns, a meeting was ultimately held with County engineers from the three counties and their concurrence to the basic assumptions, methodology and results of the ETE was in fact received. (Finding 10). If NUREG-0654 required review and comments

from County engineers--which it did not--this meeting accompanied by the written responses from the engineers would have sufficed to remedy the omission. The Board finds that evacuation time estimates have been sufficiently reviewed by State and local response organizations and that comments from these officials have been submitted as recommended by NUREG-0654. We find no merit in this contention.

2. Contention J: "Emergency Action Level indicators are incomplete in Applicants' emergency plan."[3]

This contention rests on an allegation that indicators in Applicants' emergency action level (EAL) scheme were incomplete. NRC regulations establish four classes of action levels according to severity, any of which can be initiated depending on the existence of certain plant conditions. Each of the four emergency classifications is characterized by a set of initiating conditions with corresponding emergency action levels which are observable and measurable indicators of plant status and condition. (Finding 11; see also NUREG-0887, SSER-4 at 13-6).

In its third revision of the Perry emergency plan, 13 of more than 200 individual indicators were noted as incomplete. Applicants' testimony revealed that technical data was not available for appropriate

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3 Applicants' witness: Daniel D. Hulbert, PNPP Emergency Planning Coordinator; Staff witness: Donald J. Perrotti, NRC Emergency Prep. Specialist; Intervenors' witness: Ernest J. Sternglass, Professor Emeritus of Radiological Physics, University of Pittsburgh.

values to be included in the 13 indicators but comparable values were inserted in their place (Finding 12); however, by the time Revision 4 of the plan was issued, the missing values were either determined and specified or an alternate indicator for the EAL had been chosen instead. Both Applicants and Staff concluded that this action completed the regulatory requirement for EAL indicators. (Findings 13-14).

Sunflower's witness complained that there had been insufficient time and technical information available to permit an adequate evaluation of the 13 EALs. (Sternglass, ff. Tr. 2566 at 3-4). Inasmuch as the missing indicator information was available in the February 1985 Revision 4 of Perry's emergency plan, the Board fails to comprehend a greviance concerning a lack of evaluation time. And, on the issue of needing additional information, Intervenors' attorney did not take advantage of the opportunity of examining Applicants' and Staff's witnesses on the technical foundation for the EALs submitted. (Hulbert, Tr. 2966-76).

The Board, in a broad interpretation of Sunflower's contention, permitted its witness' testimony to be admitted over objections of the Applicant and Staff. <u>See</u> Tr. 2547-48. In doing so, Sunflower was able to raise an issue that nomograms were not included in the Perry Nuclear Power Emergency Plan: nomograms are a graphic device containing a series of assumptions and variables and are recommended for use by EPA's Manual of Protection Action Guides and Protective Actions for Nuclear Incidents. (Sternglass, Tr. 2648, 2701-04). The Intervenors' witness was also able to advance testimony that the Applicants' emergency plan

included erroneous assumptions on the sensitivity of the fetus to radioactive iodine. (Sternglass, ff. Tr. 2566 at 5-6). During the interrogation on nomograms, Intervenors' witness acknowledged that methods other than nomograms were recognized by the EPA guide. (Sternglass, Tr. 2702-04; Finding 15). In fact, the Applicants utilized a different and, in its judgment, a better method for dose calculations. (Hulbert, Tr. 2971-72). On Sunflower's claim that the Perry plan included inaccurate information on the sensitivity of fetuses to radiation exposure, cross-examination demonstrated that Intervenors' information was derived from a Food and Drug Administration recommendation which relates only to ingestion pathway protective action guides. Although Sunflower's witness made an effort to blur the distinction in the regulations, it is clear that this contention deals with EALs that concern the inhalation pathway area alone. (Finding 16).

The Board finds the Applicants' EALs complete. A Staff witness indicated that conformity of the EALs to the initiating guidance of NUREG-0654, Appendix 1, is still under NRC review. Accordingly, the correction of any deficiencies forthcoming in such a review will be referenced as a condition in any license approval granted herein. (Finding 17). In its reply to proposed findings of other parties, Applicants recommend that the failure of Sunflower to file proposed findings in this instance, as mandated by the Board, was grounds for treating the contention as uncontested. Accordingly, the reply recommends its dismissal on that basis. We are not inclined to adopt this advice here since the parties have not been previously warned that

failure to file would warrant such action. The authority of the Board to dismiss contentions under 10 CFR § 2.754(b) is, of course, discretionary.

3. Contention M: "Independent Data Monitoring Systems should be installed within all counties in the Emergency Planning Zone (EPZ)."[4]

Intervenors' witness on this contention presented arguments that a system of fixed electronic detectors was necessary for radiological accidents in order to provide an instantly available picture of the plume's shape, intensity and motion. It was alleged that monitoring deficiencies without such a system during the TMI accident handicapped public officials in making proper decisions. A report by Dr. Jan Beyea, stated that car-mounted or helicopter-mounted instruments failed to provide adequate dose information in the "shifting winds" during that accident. The witness also asserted the cost of a fixed system with 100 detector instruments would be minimal compared to the cost of the plant. It was further stated that C. H. Pelletier, an AEC expert, had reported that only many air samplers and fixed detectors could adequately characterize the extent of radioactivity in a timely fashion. The witness also testified that a mobile system could not adequately measure a plume that was close to inaccessible land surfaces or over water such as nearby Lake Erie. (Sternglass, ff. Tr. 2566 at 7-9).

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4 Applicants' witnesses: Richard R. Bowers, CEI Corporate Health Physicist, Kenneth B. Cole, State Nuclear Operations Officer; FEMA witness: Shapiro; Intervenors' witness: Sternglass.

The Board denied a motion by Applicants, which was supported by Staff, to strike most of Intervenors' direct prefiled testimony on this issue. The Board ruled that consideration of the necessity for an independent monitoring system should include consideration of the adequacy or inadequacy of the system proposed. See Tr. 2549-53, 2559.

However, subsequent cross-examination throughout the proceeding raised serious questions concerning the probative and relevancy aspects of Sunflower's direct evidence. See Tr. 2667-76, 2702-09. The Intervenors' witness, Dr. Sternglass, whose qualification as an expert witness in the field of health physics was accepted by the Board, was questioned extensively for impeachment purposes by the Applicants. See Tr. 2596-2645. Although conceding the existence of an extensive list of adverse comments presented by Applicants' attorney on his prior testimony and writings before various scientific and governmental bodies, Dr. Sternglass' reply to these sallies was generally couched in terms that subsequent investigations or events confirmed his prior conclusions or that various governmental agencies deliberately suppressed important scientific data and information on radiation hazards. See Tr. 2612, 2615, 2635, 2637, 2638, and 2642. The Board notes that Sunflower's attorney made a limited but not convincing effort to rehabilitate the witness. See Tr. 2709-2725. In view of the inability demonstrated to support important parts of his testimony, the Board concludes on this and several other contentions that Dr. Sternglass' credibility has been substantially impeached.

The evidence here shows that Applicants, State and local governments have available independent monitoring services capable of responding in a timely manner to nuclear radiation emergencies around the Perry facility site. Offsite monitoring capability is recommended by NRC/FEMA regulations in NUREG-0654 and planning standards call for the Licensee and the State to have monitoring capabilities and that State or federal resources be available to track airborne radioactive plumes. (Findings 18-20). When federal guidance and standards on emergency response plans were initially being addressed by governmental agencies, fixed monitors were considered and rejected due to that system's complexity and cost. (Finding 21). As a consequence, there is no regulatory requirement for a fixed system. At Perry, such a system was estimated to cost about $2.7 million for installation alone. (Findings 22-23). A mobile monitoring system is considered superior to a fixed system because of its capability to identify the critical center line of a plume while data from fixed systems cannot be extrapolated to project doses at other locations. (Findings 24-25).

The State of Ohio has an independent capability of establishing a mobile monitor response within 3-3½ hours and before that, can rely, if necessary, on Applicants' monitoring team and other resources. Releases over Lake Erie can be tracked by a Department of Energy helicopter and mobile teams can be dispatched to cover shorelines for returning plumes. See Bowers, Tr. 2958-59; Cole, Tr. 2901. The State has the primary responsibility for independent monitoring, and even though Lake County also will have independent monitoring teams, all counties within the EPZ

will rely on the State which has demonstrated an effective capability during eight emergency planning exercises. (Findings 26-34).

The Board concludes that an effective independent monitoring system, which meets all regulatory guidance and standards, has been programmed for the Perry facility. Contention M has no merit.

4. Contention P: "Emergency plans are deficient with respect to hospital designations and medical services as well as procedures required to assist contaminated individuals."[5]

In support of Sunflower's contention, testimony was received from Dr. Sternglass that studies showed a serious radiation accident at Perry could contaminate large numbers of people which local medical services could not accommodate. See Sternglass, ff. Tr. 2566 at 13-14. A statement by Dr. Robert L. McTrusty, Chairman of the Ashtabula County Medical Center's Disaster Committee, averred that the Medical Center (one of four hospitals listed in local emergency plans) lacked proper equipment and facilities to decontaminate even minimal numbers of radiation victims. Based on a Sandia National Laboratory report (NUREG/CR-2239), Dr. McTrusty concluded that worst-case accidents could number in the thousands at Perry. His testimony stated that personnel

5 Applicants' witnesses: Roger E. Linneman, Vice Chm., Radiation Management Corp., Deborah Hankins, Principal Engineer, General Electric Co.; FEMA witness: Robert O. Shapiro; Intervenors' witnesses: Ernest J. Sternglass, Robert L. McTrusty. Dr. McTrusty did not appear at the hearing due to a schedule conflict, but his testimony was stipulated by the parties for admission.

at the medical center received only two hours of training and indicated that no requirement in the State's accreditation manual for the emergency handling of radiation victims could be found. See McTrusty, ff. Tr. 3149.

Under cross-examination, Dr. Sternglass admitted that the studies his testimony cited, and on which his predictions of radiation casualties were based, were not related to the Perry reactor design and he acknowledged that he was unaware of the specific type of containment used at Perry. (Sternglass, Tr. 2685-90). Rebuttal testimony by the Applicants demonstrated the inapplicability of the Sandia study--which was for the purpose of developing NRC siting criteria--to the Perry facility. That study assumed a design substantially different from Perry and does not represent the risk of a severe nuclear plant accident at any particular site. The source term (SST 1) used was an estimate of the largest possible release of fission products from a pressurized water reactor and assumes worst-case conditions. The Perry design has mitigating features not considered in the Sandia study, which reduce fatalities and injuries to a minimum in the event of a core-melt accident. See Findings 45-46.

NRC's regulatory requirements call for arrangements being made for medical services for contaminated injured individuals. See 10 CFR § 50.47(b)(12). In NUREG-0654, the applicable criteria recommend arrangements for local and backup hospital services capable of evaluating radiation exposure and handling contamination and transportation of the injured. (Finding 35). The testimony of

Applicants' witness, the Chief Medical Officer of the Radiation Management Corporation (RMC), demonstrated that the four county hospitals within the EPZ listed in local emergency plans have trained personnel and equipment for handling contaminated injured individuals; letters of agreement for their services are being obtained. (Finding 36). The State plan also lists an additional 26 hospitals and counties around the EPZ, all of which have diagnostic and/or therapeutic radioisotopic facilities which can provide medical support for contaminated injured individuals. (Finding 37; see also Linneman Tr. 2979, 2981). RMC has trained 85 personnel in the four hospitals within the EPZ, the State has provided additional training, and each of the County hospitals has equipment which allows detection of high and low levels of radiation. (Findings 38-39). The hospital, with which Sunflower's witness, Dr. McTrusty, is affiliated, has a designated radiation emergency room with decontamination equipment and emergency personnel who received 24 hours of training for nuclear emergencies. The Ashtabula Hospital is also accredited by the State and has been licensed by the NRC for handling contaminated individuals. (Finding 40).

The Applicants' witness, Dr. Linneman, is a radiologist with 15 years' experience in the treatment of contaminated injured patients. He testified that radioactive injuries seldom required hospitalization and that the hospitals within the EPZ and the surrounding areas were adequate to accommodate the unlikely case of large numbers of casualties. (Findings 41-42). County hospitals all have the capability

of detecting radiation overexposure and local and State plans have provisions for the transportation of radiological accident victims. (Findings 43-44; Shapiro, ff. Tr. 3111 at 7).

The Board found the evidence presented by the Applicants and Staff to be unchallenged and credible. We conclude that the training, personnel and equipment at the designated hospitals within the EPZ as well as the medical resources available outside of the EPZ are adequate to comply with the Commission's regulations and standards on hospital designations, medical services and procedures. The arrangements for medical services for contaminated individuals meet the requirements of 10 CFR § 50.47(b)(12). In accordance with the Commission's policy statement of 50 Fed. Reg. 20892, however, we will require the Applicants to fully comply with any additional requirements that may be forthcoming in the Commission's response to the U.S. Court of Appeals' decision in Guard v. NRC, 753 F.2d 1144 (D.C. Cir. 1985). In that decision, the Court vacated the Commission's interpretation that a mere listing of hospitals capable of caring for victims was a sufficient arrangement and the Commission is considering what additional medical service requirements it should impose.

5. Contention Q: "There are no letters of agreement regarding the availability of school buses."[6]

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6 Applicants' witness: John Baer, Project Mgr., Emergency Management Services; Staff witness: Robert Shapiro.

This contention was founded on the fact that letters of agreement have not been obtained for the use of school buses during emergency evacuations. (Finding 47). School buses in Ohio are owned and controlled by local school districts and, consequently, their utilization, during an emergency evacuation, requires the consent of officials from those districts. (Finding 48).

The Applicants and Staff provided evidence, which was unchallenged by Intervenors, that letters of agreement were in the process of being obtained, no difficulty was anticipated in obtaining them, and it was expected that their acquisition would be completed by the time for fuel load. There is a total of 24 school districts within and outside the EPZ where needed transportation resources might be available and no problem was foreseen in obtaining signed agreements. The Applicants are supplying radios for the buses and agreements are expected to follow the installation of the radios. (Findings 49-52).

We conclude that since no evidence has been submitted to cast doubt on the letters of agreement being received prior to fuel load, there is no basis to support this contention. Negotiations are underway, no objections to the letters have been reported, and Intervenors submitted no witnesses on this contention. The Board does believe, however, that these letters should be obtained prior to issuance of an operating license.

6. Contention U: "Reception centers do not have the means or facilities for handling contaminated property."[7]

The Applicants and Staff witnesses provided testimony that local emergency plans and procedures include standard action levels in the three counties in the EPZ for monitoring and decontamination of clothing, isolation of vehicles, and other property. Each county also has standard operating procedures which provide specific directions to fire departments who are responsible for carrying out decontamination procedures. (Finding 53).

An adequate number of fire department personnel are currently being trained for monitoring and decontamination duties and equipment and supply kits are being assembled for monitoring and decontamination purposes. This activity will be completed prior to fuel load. (Findings 54-55). The County plans erroneously list the Cleveland Electric Illuminating Company as the entity to handle the disposal of decontaminated property. The Applicants' witness stated that the State plan accurately provides that the Ohio Environmental Protection Agency will have that disposal responsibility and local plans are being changed to reflect that arrangement. (Finding 56).

Intervenors' cross-examination brought out the fact that the Ohio Disaster Service Agency's radiological training manual, which is used to

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7 Applicants' witness: John Baer, John Wills, Radiological Analyst, Ohio Disaster Services Agency; FEMA witness: Robert O. Shapiro.

train firemen for decontamination procedures, implies that firemen and policemen would have to decontaminate their own vehicles. However, Applicants' rebuttal testimony from a DSA official corrected this impression by indicating that the manual reference did not apply to contamination received from accidents at nuclear power stations. (Finding 57). The emergency plans and procedures call for using areas adjacent to reception centers for decontaminating vehicles. (Finding 58).

The Board finds that the reception centers around PNPP will be adequately supplied with equipment and supplies for implementing monitoring and decontamination procedures in handling property and that trained fire personnel will be available to carry out this assignment. This contention is without merit.

7. Contention Z: "The plants do not provide decontamination protection for bus drivers during an emergency."[8]

Intervenors' witness on this contention urged the necessity of providing respirators and goggles to bus drivers alleging that repeated trips by buses into contaminated areas would make such equipment necessary. In support thereof, the EPA manual of protective action guides which recommends respirators for emergency workers was placed in

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8 Applicants' witnesses: John Baer, John Wills; FEMA witness: Robert Shapiro; Intervenors' witness: Ernest J. Sternglass.

evidence. There are NRC regulatory requirements to control radiation exposures to emergency workers. (Finding 59).

Applicants and Staff witnesses testified that bus drivers would be exposed to little, if any, radiation since their duties would be completed prior to any serious contamination being experienced. In the possible event of some exposure, however, the plans do call for the issuance of dosimeters to bus drivers who will be trained in their use. Additionally, 2-way radios are being supplied to buses by the Applicants which will facilitate the transmission of radiological information on evacuation routes to bus drivers. (Findings 60-61).

Although the Ohio Department of Health currently requires that respirator equipment be provided to emergency workers, a rebuttal witness for the Applicants from the State's Disaster Services Agency testified that the applicability of that provision to bus drivers was being eliminated. (Finding 62). The Board concludes that the duties of bus drivers during an emergency evacuation do not require the issuance of respiratory equipment and that the decontamination equipment that is being provided is adequate to conform to federal regulations. This contention is found to be without merit.

8. <u>Contention BB</u>: "Offsite emergency plans are inadequate due to the planning deficiencies set forth in the Federal Emergency Management Agency Interim Report of March 1, 1984."[9]

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9 Applicants' witness: John Baer; FEMA witness: Robert Shapiro.

The planning deficiencies referenced in this document were the result of a review of the draft local emergency plans by the FEMA Regional Assistance Committee (RAC). Of the 145 deficiencies listed in the interim report, more than half were corrected and accepted by FEMA in the report itself; the remainder have since been corrected or are being addressed. (Finding 63). The few remaining deficiencies primarily involve an emergency information handbook which is to be available prior to fuel load. An NRC witness testified that his review of the information handbook found the handbook adequate to meet NRC regulatory standards. (Finding 64).

All local plans have been revised since the 1984 Interim Report to reflect the action taken on the planning deficiencies and Applicants submitted into evidence a list of actions on each of the deficiencies noted. (Finding 65). In a full participation exercise conducted in November 1984, no Category A deficiency (affecting public health and safety) was noted by FEMA. (Finding 66).

The Board concludes that all deficiencies in the March 1984 Interim Report are remedied or are in the process of being corrected. Accordingly, local emergency plans are not considered inadequate due to these deficiencies and we find this contention to lack merit.

9. <u>Contention CC</u>: "The resolution items set forth by the staff in its Safety Evaluation Report, NUREG-0887, Supp. 4 (February 1984), pages 13-1 to 13-22, are uncorrected deficiencies in the emergency plans."[10]

There were 35 items identified in SSER-4 as requiring resolution in the PNPP emergency plan. The Applicants have since made several revisions to its plan in which the deficiency items were resolved. (Finding 67). The Staff is currently reviewing the latest plan revision (Revision 4) to confirm that the Applicants have complied with the planning commitments previously identified. (Finding 68).

The Board finds that the deficiencies noted by Intervenors, who conducted no cross-examination on this contention, have been satisfactorily resolved. The contention is found to be without merit.

Conclusions

The Board has reviewed each of the emergency planning deficiencies submitted by the Intervenors and on the evidence of record found them to be unsubstantiated. This is not to state that certain activities do not require completion. Progress still has to be made in regard to EALs, letters of agreement on the use of school buses, training of fire personnel, equipment at reception centers and a formal commitment to the Commission's response on the availability of hospital facilities and

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10 Applicants' witness: Daniel D. Hulbert; Staff witness: Donald J. Perrotti.

services. The final accomplishment of these emergency plan activities the Board believes will be concluded but to assure this result, conditions will be attached to the issuance of a license. Except for these items, the Board concludes that Applicants have met their burden of proof on Intervenors' allegations, the emergency plans are not inadequate based on those allegations and the contentions are accordingly dismissed.

B. HYDROGEN CONTROL (Issue 8)

Ohio Citizens for Responsible Energy (OCRE) and the Sunflower Alliance, Inc. (Sunflower) originally submitted hydrogen control contentions that were rejected for not meeting admission standards. Specifically, Intervenors failed to specify a credible accident scenario as required by a Commission Order. Metropolitan Edison Company (Three Mile Island, Unit 1), CLI-80-18, 11 NRC 674 (1980). Sunflower subsequently renewed its motion and the contention was approved on the basis that the Commission was considering a proposed rule which specifically addressed additional hydrogen control protection to be required in MARK III BWRs. The contention proceeded through a tortuous history which included at least one rewording and a later designation of OCRE as its lead intervenor. We need not recount that history here since it has been adequately discussed in previous rulings.

The Commission adopted a new hydrogen rule for BWRs with Mark III containments on January 17, 1985. (50 Fed. Reg. 3498). OCRE filed a

motion requesting another rewording of Issue 8 so as to conform its language to that of the new rule. OCRE's motion was opposed by the Applicants and the Staff who sought summary disposition of the issue. The Board granted OCRE's motion, reworded Issue 8 in a Memorandum and Order of March 14, 1985 and dismissed the Staff's motion for summary disposition as moot.

The Contention

The contention reads:

> The Perry hydrogen control system is inadequate to assure that large amounts of hydrogen can be safely accommodated without a rupture of the containment and a release of substantial quantities of radioactivity to the environment.

The wording basically alleges that Applicants' hydrogen control system does not conform to the new regulatory requirements of 10 CFR 50.44 and challenges both the hydrogen control system and the ability of the containment to withstand the consequences of large amounts of hydrogen release.

The Hydrogen Control Rule Controversy

The new rule requires BWR power plants having Mark III containment designs to install a hydrogen control system capable of handling an amount of hydrogen equivalent to that generated from a 75% metal-water reaction of the active fuel cladding without loss of containment integrity. A license for full power operation can be granted on the basis of a satisfactory preliminary analysis of the effectiveness of the

system which is approved by the Staff. The analysis may reference previously approved analyses for plants of similar design. A final analysis, addressing applicable provisions of Section 50.44(c)(3)(iv), (v), (vi) of the Rule, must be provided by the Applicants after commencement of reactor operation at full power on a schedule to be established by the NRC Staff. (50.44(c)(3)(vii)(A), (B)). The rule deals only with recoverable degraded core accidents. The Applicants' hydrogen control system also includes a combustible gas control system dealing with design basis accidents that was not litigated in this proceeding. (Finding 74).

Pursuant to the new regulation, Applicants' analysis must (1) evaluate for Perry the consequences of the generation of large amounts of hydrogen, and include consideration of hydrogen control measures, as appropriate; (2) include the period of recovery from the degraded condition; (3) use accident scenarios that are accepted by the NRC Staff; (4) support the design of the hydrogen control system selected; and (5) show that (i) containment structural integrity will be maintained; (ii) that systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. The effects of local detonations are to be considered unless such detonations can be shown as unlikely to occur.

The new hydrogen rule is unclear as to the boundaries that distinguish a preliminary from a final analysis. It is clear that the

rule permits the issuance of licenses for full power operation on the basis of a satisfactory preliminary analysis. Not surprisingly, the parties dispute the scope of a satisfactory preliminary analysis. OCRE argues that the preliminary analysis must be all but complete before the Board can make a finding of reasonable assurance of safety. (OCRE Proposed Opinion at 9-14). The Staff and Applicants, on the other hand, believe that a number of open or unresolved items can be deferred until the final analysis and have agreed to the scope of a preliminary evaluation and analysis for Perry. (Finding 71). The Board finds no need to resolve the dispute in this instance. We find that the Applicants' preliminary analysis as described in the preliminary evaluation report and in Applicants' testimony, and as approved by the Staff and discussed in Staff's testimony, does address in detail the substantive provisions of the hydrogen rule.

In the rewording of the contention Order, supra, the Board defined the scope of the hearing to permit OCRE to challenge only the adequacy of the Applicants' preliminary analysis. The Applicants have no obligation under the rule to present a final analysis until some time after reactor startup and operation at full power. We did not, however, attempt to define the boundaries of the preliminary analysis in advance. During the hearing we permitted OCRE broad latitude in its cross-examination over the objections of Applicants and Staff who asserted that the questioning went beyond the scope of a preliminary analysis.

Our standard of acceptance in this case is simply that we will determine whether there is reasonable assurance of safety during

operation at Perry based on our assessment of the record that was actually developed. The Board takes account in findings that, if a license to operate is granted, the final analysis will follow in a relatively short time after reactor startup. (Finding 73).

Witnesses

Both the Applicants and the Staff presented the testimony of expert witnesses in panels.[11] The Board found the Applicants' and Staff's witnesses to be competent technical experts and qualified to testify on the issues presented. The Board rejects as not substantiated all assertions that one or more witnesses were incompetent, inexperienced, evasive or untruthful. OCRE presented no witnesses on this contention and relied on extensive cross-examination to make its case.

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11 Applicants' witnesses: Eileen M. Buzzelli, Senior licensing engineer at CEI; John D. Richardson, Vice President of Enercon Services, Inc.; Kevin W. Holtzclaw, principal licensing engineer in the General Electric Safety and Licensing Operation of the Nuclear Energy Business Operation; Roger W. Alley, Manager of the Structural Engineering Nuclear Section of Gilbert/Commonwealth, Inc., Architect-Engineer for PNPP; Dr. Bernard Lewis, President of Combustion and Explosives Research, Inc. (Combex); Bela Karlovitz, Secretary-Treasurer of Combex; Dr. G. Martin Fuls, President of FMF Associates; and James H. Wilcox, Supervisor of the Test Support Group at PNPP. Staff witnesses: Allen Notafrancesco, NRC Containment Systems Branch, (written testimony in two parts, designated I and II); Hukam Garg, NRC Equipment Qualification Branch; Li Yang, NRR Structural and Geotechnical Engineering Branch; and W. Trevor Pratt, Brookhaven National Laboratory.

THE HYDROGEN IGNITION SYSTEM

The evidence shows that the Applicants have installed a distributed igniter system in the Perry containment and analysis of its reliability has been performed. The igniter system, consisting of 102 glow plug igniters distributed throughout the containment, wetwell, and drywell, is designed to control large amounts of hydrogen that could be released in a degraded core accident. The system is powered by AC Class IE power systems which are backed by diesel generators, and is designed to burn off hydrogen at low concentrations to prevent accumulation of high concentrations. The igniters are manually actuated when the reactor water level drops to the top of the active fuel. Intervenors do not contest the basic features of the igniter system. (Findings 69-84).

Written procedures for operation of the igniter system were not available at the time of the hearing but will be before Perry exceeds 5% power. (Finding 85). Intervenors urge the Board to find that the absence of final operating procedures constitutes a deficiency in the Applicants' preliminary analysis. (OCRE Proposed Opinion at 15). OCRE believes that the procedures should not only be finalized prior to operation but that they should be available for scrutiny in our hearing because, in its view, these procedures are of sufficient complexity to warrant our thorough evaluation. Without this scrutiny, according to OCRE, we would have no way of assuring that the procedures used would be appropriate, that instruments relied upon by the operators are available, or that 10 CFR Part 100 guidelines will be met for containment venting.

The Board concludes that the Applicants have installed and performed technical analyses of a distributed igniter system in the Perry containment and that the system is designed to cope with large amounts of hydrogen that could be released in a degraded core accident. Since Applicants have committed to providing written procedures prior to exceeding 5% power, this is not an element that will be deferred to the final analysis. OCRE has not alleged or pointed to any equipment that would not be available nor has it shown that NRC's new regulations on hydrogen control are the vehicle by which we test compliance with Part 100 in licensing proceedings. We need not address OCRE's concerns about releases of radioactivity because Perry will not be allowed to operate if the hydrogen control system proves to be inadequate. Thus, we reject these arguments as a basis requiring our scrutiny of written procedures. OCRE further bases its argument on the fact that the Applicants have not prescribed a vent path for containment in the event one is needed for a degraded core accident in conjunction with a station blackout event. We disagree with OCRE that the identification of a vent path from containment is a matter so complicated that it requires our scrutiny at hearing. It is sufficient to know that one will be chosen. The Board concludes that the Applicants' preliminary demonstration with regard to the installation, design and operation of igniters is adequate. Its commitment to produce written procedures prior to operation at full power is also adequate but it will be made a condition of licensing. If, when completed, the procedures are deficient in

OCRE's judgment, it can seek relief through other NRC proceedings. See 10 CFR § 2.206.

CONTAINMENT INTEGRITY

The evidence shows that an analysis of PNPP containment integrity was performed according to the requirements of the ASME Code service level C. The internal pressure capacity of the PNPP containment was evaluated by Applicants' consultant, Gilbert Commonwealth (G/C). Results are set forth in a report entitled "Cleveland Electric Illuminating Company, Perry Nuclear Power Plant, Units 1 and 2, Ultimate Structural Capacity of Mark III Containments" (Ultimate Capacity Report). This analysis presents one method pursuant to the rule by which Applicants can demonstrate the pressure-containing capacity of the Perry containment vessel and it disclosed that the most limiting penetration in containment, P414, had a pressure capacity of approximately 50 pounds per square inch (psig). The dome knuckle region and containment cylindrical shell all had pressure containing capacities considerably above that limiting penetration. (Findings 86-94).

OCRE criticizes Applicants' analysis because it did not include a consideration of dead load as required by the ASME code, it neglected the effect of elevated temperatures on material properties due to hydrogen combustion and it neglected the stresses resulting from the as-built, out-of-tolerance conditions of the PNPP containment vessels. The evidence shows, however, that Applicants considered all three of the

foregoing factors. They concluded that the factors were insignificant and would not affect their analysis. (Findings 90, 91, 95).

We do not find it impermissible to consider each of these quantities, to find them of small quantitative import and therefore to neglect them. We also cannot accept OCRE's invitation to consider these effects as additive. (OCRE Proposed Opinion at 21). The three factors mentioned by OCRE do not converge in additive fashion on the limiting penetration. The out-of-tolerance condition was acknowledged by the Applicants but has its effect on the vessel shell and not the limiting penetration. Stresses during elevated temperatures were considered and shown to remain conservative, and the dead load contribution constitutes only 4.7% of the stress. It is properly neglected in an analysis having substantial conservative margins. OCRE's assertion that Applicants' containment vessel capacity analysis is inadequate or erroneous is not supported by the evidence.

We also cannot accept OCRE's assertion that containment penetration P205 was analyzed in an improper manner because Applicants took credit for actual material properties instead of lower-bound material properties. The analysis of that penetration adequately demonstrates that the conclusions regarding the limiting penetration in containment should not be altered. (Findings 94, 110).

Applicants' containment analysis shows that the strength of containment is significantly greater than its design strength, that it can withstand negative pressures and that containment strength exceeds

the pressure expected from hydrogen burning by a large margin. (Findings 96-98).

Defective Welds

Some defective welds remain in the lower weld courses of the containment vessels in both Perry Units 1 and 2. Both Applicants and Staff are aware of the defects and a detailed analysis of their importance has been performed by the Applicants. (Finding 99). OCRE challenges Applicants' weld flaw analyses on the basis that stress margins used were not conservative, that measurements of weld flaw size used some radiographs of poor quality, and that a faulty calibration was made of the radiographic analysis technique used by Applicants' consultant, APTECH Incorporated.

The evidence reflects that the Staff has accepted the digital radiographic enhancement technique used by APTECH and only has reservations on the depth-of-flaw measurement using this technique. Calibrations show that flaw depth measurements are conservative in that the digital enhancement technique projects deeper flaws than actually exist in the material. (Findings 100, 101). OCRE's assertion that inadequate margins were used in the stress analysis represents nothing more than a difference of opinion with experts and it ignores throughout the conservative aspects of the analysis. (Findings 100-106).

OCRE criticizes Applicants' evaluation of stresses at these welds for 50 psig and its finding that the stresses at the higher internal pressures were less than at 15 psig as analyzed by APTECH. Contrary to

OCRE's assertions, however, we do not find Applicants' witness to be noncredible on this matter. It is clear that Applicants conducted an additional analysis of hydrogen deflagration stresses that went beyond the analysis performed by APTECH. While APTECH's analysis did not consider the annulus concrete, the Applicants in their additional work did evaluate additional stiffening in the containment vessel due to it. (Finding 103). We see nothing improper in this analysis. Clearly the annulus concrete now exists in the as-built structure and consideration of its additional strengthening characteristics seems to us appropriate. APTECH's choice not to consider the annulus concrete in its analysis adds to our confidence that the overall analysis is conservative relative to the final as-built containment.

Applicability of ASME Service Level C Limits

We reject OCRE's invitation to challenge the use of ASME service level C limits to insure a leak-tight containment. (OCRE Proposed Opinion at 25). The newly promulgated hydrogen rule permits the use of service level C limits as a means of demonstrating containment shell integrity and that of its penetrations. The evidence shows that Applicants' analysis of the equipment hatch and other containment penetrations has been adequate and that all penetrations meet service level C requirements. (Findings 107-111).

Penetration Seals

The Applicants performed an analysis of springback of the O-ring seals in the equipment hatch. (Findings 107-108). Contrary to OCRE's assertion, we find that the Applicants did not neglect the effect of compression set which could result from thermal and radiation aging. (Finding 108). Further, our analysis of the adequacy of the hydrogen control system is limited to the adequacy of the Applicants' preliminary analysis. We are responsible for making findings of reasonable assurance of safety during the interim between reactor startup and the submission of Applicants' final analysis. Nothing in the record would cause us to conclude that radiation or thermal aging of materials could create serious compression set during this interim period. The Applicants have ample time to complete their commitment to prescribe what appears to be a purely routine procedure for maintenance and inspection of seals.

Manufacturers and contractors have brought to the Applicants' attention potential defects in inflatable seals. This notification caused a re-evaluation of the integrity of seals and their possible degradation during design basis accident conditions. We do not accept OCRE's assertion that the discovery of such potential problems and their reporting--even though followed by assessment and corrective action--is a cause for concern regarding the integrity of penetrations. (Finding 112).

OCRE argues that the ASME code may not be the most conservative method for all parts and conditions encountered in PNPP containment.

However, the new hydrogen rule permits the use of ASME service level C stresses in a demonstration of containment integrity, and the Applicants have presented even additional conservatism by providing an analysis based on service level D stresses, which the code permits. (Finding 92). Additionally, we see no reason for challenging Applicants' assertion that their techniques are applicable for the linear elastic stress ranges under consideration. Neither can we find a basis for challenging the finite element analysis used by Applicants for consideration of the stress ranges in the Perry containment. (Finding 93).

We conclude that the Applicants have identified the controlling penetration in containment and that such penetration has an internal pressure capacity of 50 psig. We find that pressure limit is conservative and that in fact the pressure limits of the controlling penetration are higher than those stated. OCRE criticizes Applicants' analysis of negative pressure capacity of the containment because Applicants rely on the action of vacuum breakers during design basis events to establish that design negative pressure is not exceeded. No basis has been established in this record for challenging the actions of redundant vacuum breakers to relieve negative pressures in containment. (Finding 97).

Drywell Capacity

The Applicants relied on the analysis of the Grand Gulf drywell to establish pressure-retaining capabilities of the Perry drywell. OCRE

considers this inadequate because the Grand Gulf analysis did not consider the effect of voids in the drywell. Voids were found in the Perry drywell concrete. However, they have been repaired and analysis has demonstrated that they have no effect on the pressure-retaining capacity of the Perry drywell. The concrete of the drywell has been inspected and the Applicants are confident that no further voids exist. (Findings 114, 115).

We have no basis to challenge Applicants' analyses on concrete voids in the drywell. The hydrogen rule permits the referencing of previously accepted analyses for similar plants as part of the overall analysis for Perry. Thus, the Applicants' referencing of Grand Gulf drywell integrity is not inappropriate to establish, on a preliminary basis, the integrity of the Perry drywell.

CONTAINMENT RESPONSE ANALYSIS

Applicants have analyzed two accident scenarios to describe the behavior of the reactor system during and following a degraded core accident. These scenarios consist of (1) a small steam line break in the drywell with extended ECCS failure and (2) a transient with a stuck-open relief valve with extended ECCS failure. Recovery of coolant flow was assumed to occur at the point of a 75% metal-water reaction. Steam and hydrogen release during these accidents was analyzed using a computer code termed MARCH. Intervenor does not challenge the appropriateness of the two scenarios used nor does it challenge the use of the MARCH code in spite of its recognized shortcomings. (OCRE

Proposed Opinion at 16). We have found that an improved MARCH code would show less hydrogen release than the one used in the Applicants' preliminary analysis. Therefore that analysis is conservative. (Findings 116-121).

OCRE is concerned that the scenarios chosen do not represent the most severe challenge to containment from hydrogen combustion. The Intervenor urges that Applicants should also have considered a station blackout degraded core accident, which assumes loss of both offsite and emergency electrical power. In this scenario, the igniters would not be available and hydrogen would accumulate to a concentration of 28% in containment. Deflagration of this amount of hydrogen when power is restored to the igniter system would produce high pressures in containment. (Findings 144, 145).

Applicants argue that consideration of this scenario should be deferred to the final analysis stage. Staff agrees and will require evaluation of the station blackout in the final analysis unless the Applicants make an adequate demonstration as to why it should not be considered. (Finding 158).

The Applicants, however, have in testimony given preliminary consideration of the station blackout situation. A degraded core situation is unlikely to arise during a station blackout because core cooling would be available up to 9 hours after the blackout occurred. As long as there is core cooling, there would be no hydrogen release. (Finding 146). The Board concludes that the core cooling capability during station blackout gives the station substantial amounts of time to

recover either offsite power or emergency power before any hydrogen release can be expected. If hydrogen is released during a station blackout event, however, the Applicants have options for venting containment rather than permitting a global deflagration to occur. (Finding 145).

The hydrogen rule is silent on the question of whether station blackout must be analyzed in the preliminary analysis or, indeed, whether it must be analyzed at all as one of the scenarios employed by the Applicants. A judgment must be rendered on the basis of whether a finding of reasonable assurance of safety can be made in the absence of the analysis at the preliminary stage. We conclude that such finding can be made on the basis of the low probability of the event and the fact that although the formal analysis has not been made the Applicants are able to outline the general dimensions of its response. We find this to be adequate for a preliminary analysis.

We also note that the station blackout scenario has been addressed in the Staff SER as a generic unresolved safety issue (Perry SER). The station blackout scenario represents a safety problem, all aspects of which are not directly before us and the full dimensions of which are not on our record. Clearly our principal task under the admitted contention is to assess the adequacy of the hydrogen igniter system regarding its functional capability of coping with a 75% metal-water reaction without loss of containment integrity. Since the Staff intends to require that the Applicants analyze the station blackout with regard

to hydrogen generation in its final analysis, we conclude that adequate assurance of safety at Perry is provided.

The Applicants used the CLASIX-3 computer model, originally developed by Westinghouse, for calculating its containment response analysis. That model predicts temperatures and pressures resulting from the combustion of hydrogen and it tracks concentrations of hydrogen, oxygen, nitrogen and steam in the containment. The model uses standard equations and assumptions and predicts, for both scenarios analyzed, a large number of hydrogen burns taking place primarily in the wetwell region of the Perry containment. (Findings 122-124).

OCRE challenged the input parameters used in the CLASIX-3 model and the validity of the code itself. In doing so, it urged Board reliance on a document produced by Sandia National Laboratory labeled NUREG/CR-2530 (OCRE Exhibit 21). That report contains Sandia's comparative assessment of a number of computer models designed to calculate containment response during hydrogen combustion. The analysis was performed for the NRC Staff to assist it in the assessment of the hydrogen control system at Grand Gulf. One of the models advocated by OCRE that was compared in the Sandia analysis was termed HECTR. Results from HECTR consistently showed higher temperature and pressures from hydrogen burning in reactor containment than CLASIX-3. However, Staff claimed that the HECTR model was crude and did not model a MARK-III containment. (Notafrancesco, Tr. 3688, 3724, 3733). We have examined OCRE Exhibit 21 and conclude that nothing therein gives us any more basis for relying on HECTR than on CLASIX. (OCRE Exhibit 21 at 12,

15-16). The Sandia report does establish that several models used in early attempts to model containment response produced variable results. Sandia's report, however, does not indicate which of the models was the more reliable although it accepts the CLASIX results with reservations. (Id. at 11).

The Staff testified that the modeling of containment response has improved since the Sandia report was completed. (Pratt, Tr. 3700). Indeed, even the HECTR model of later vintage shows lower containment temperatures and pressures than the earlier version. (Notafrancesco, Tr. 3733-34). The Board concludes that the modeling of containment response is a rapidly changing and improving research and development effort. We do not put our reliance on the model comparisons of the Sandia report for assessing containment response in Perry since we accept that later improved versions of models have been developed and that later experiments tend to confirm the validity of CLASIX-3. (Fuls, Tr. 3621). Additional subjective considerations also support a conclusion that CLASIX results are conservative. (Finding 133).

The Sandia report identified numerous sensitivities of all the models to variations in input parameters, which we believe remains valid. One of these was flame speed during combustion of hydrogen. Sensitivity analyses established that high flame speed is accompanied by more rapid release of energy and correspondingly high peak pressures and temperatures are attained. Thus accurate modeling results depend upon accurately known flame speeds. (OCRE Exhibit 21 at 16-17).

The flame speed used in CLASIX analysis was 6 feet per second. The Staff believes this is conservative and that the flame speeds used in HECTR were overestimated. Similarly, Applicants' experts on combustion believe that flame speeds could be lower than 6 feet per second and that the assumption used in CLASIX is conservative. However, some experiments have shown higher flame speeds and Sandia thought they should be higher as well. (Finding 132; OCRE Exhibit 21 at 16-17).

We accept that OCRE has established by its cross-examination and Exhibit 21 that flame speed is an uncertain quantity not known with high accuracy. The most we can deduce from the testimony presented is that the flame speeds used in CLASIX are not deliberately biased. We conclude that true flame speeds, if and when finally known with precision, could be lower or higher than the values used. (Finding 132). This conclusion does not arise from any evidence of faulty craftsmanship on the part of those engaged in the construction of models. Rather, it appears to be an inherent uncertainty in the ability to measure a parameter that varies with environmental conditions.

In view of the uncertainty of the parameters which apply as well to HECTR as they do to CLASIX-3, we put no reliance on OCRE's detailed comparison of the results of CLASIX-3 with HECTR. (OCRE Proposed Opinion at 33 and Appendix C).

OCRE also challenged the values for the ignition limit of hydrogen in air. The evidence shows that hydrogen has geometry-dependent ignition limits in air. (Finding 126). Eight percent hydrogen concentration in air corresponds to the downward propagation limit.

Sandia used 9% for a downward propagation limit and Applicants' expert thought that the downward propagation limit was in the range of about 8-1/2 to 10% Upward propagation limit is about 4% and horizontal propagation limits occur at around 6-1/2% hydrogen by volume. The uncertainty in the ignition limits of hydrogen in air appears to be considerably smaller than the uncertainty for flame speeds and a lingering dispute as to the correct values appear to be insignificant to the containment response analysis. Citing the Sandia report, however, OCRE claims that the placement of igniter assemblies close to ceilings and the spray shield of the igniter housing would inhibit combustion effectiveness. Sandia concluded, however, that ignition would reliably occur at the downward propagation limit but not at lower concentrations. We fail to see any important disagreement between OCRE and the Applicants since Applicants' witnesses also state that hydrogen combustion will not occur in downward propagation at concentrations less than about 8%. Eight percent hydrogen concentration is the concentration assumed for all flame propagation in the CLASIX model. It is, therefore, a conservative assumption since upward propagation or horizontal propagation will in reality occur at considerably lower concentrations. (Findings 125, 126, 131).

Considering the undisputed fact that hydrogen tends to rise from lower to upper regions in the containment, we conclude that considerable amounts of hydrogen will burn at the upward propagation limits before accumulating at higher concentrations at ceilings or spray shields on igniters. We agree, therefore, with Applicants that variations in the

location of hydrogen igniters in containment is not critical to the overall analysis of hydrogen combustion. (Finding 84). The igniters are placed in containment anywhere that hydrogen might go and there is no need to consider an igniter-by-igniter location analysis as performed by OCRE. (OCRE Proposed Opinion at 35). Given the potential for combustion at lean concentrations during upward propagation or horizontal propagation, we see little hazard in the possibility that hydrogen might somehow escape these ignition sources and accumulate at higher concentrations in the containment dome or near ceilings. (Findings 83-84). Even if hydrogen did accumulate, however, it would burn in downward propagation at approximately 8% concentration just as the CLASIX model assumes and we see no error in this aspect of modeling the behavior of hydrogen by CLASIX. (Finding 81).

OCRE challenges the combustion completeness of hydrogen in air of 85% utilized in the CLASIX model. Other researchers have used 100% combustion of hydrogen at concentrations above 7.7%. There is no basis in the record for us to decide with any certainty on a correct value for combustion completeness, although we have found that as a practical matter much hydrogen will be burned at lower concentrations than 8%; this lends credence to the estimate of 85% completeness. We consider this to be a minor part of the intrinsic uncertainty in modeling.

We conclude that even current models have considerable parameter sensitivity and that at least some parameters are poorly known. Although the Staff acknowledged this in its testimony, it was not clearly highlighted. However, as already indicated by the Staff, we

conclude that it is essential that Applicants in their final analysis perform appropriate parameter sensitivity analyses in its determination of containment response. (Finding 158).

We reject OCRE's assertions that ionizing radiation will increase flame speed from deflagration to detonation. There is no evidence in our record that the specific chemical radicals needed to accelerate flame speed would be present in Perry containment or that ionizing radiation in any event could create enough such radicals to accelerate flame speed. OCRE's assertion was based on old and outdated evidence and there was no recent corroborating evidence to suggest that ionizing radiation in containment could have any effect. (Finding 135). The evidence is clear that no detonation of hydrogen from any cause is likely in the Perry containment. (Findings 127, 134).

Containment Spray Availability

OCRE also challenged the availability of containment spray operation during a degraded core accident. The evidence shows containment sprays are an important heat transfer mechanism that, during a hydrogen burn, would result in a significant reduction of pressure and temperature. (Finding 129; OCRE Exhibit 21 at 12). It is assumed in CLASIX-3 that containment sprays are automatically actuated after the first burn. No factual basis exists in the record, however, for challenging the availability of containment sprays. We cannot accept OCRE's assertion that containment spray unavailability can be deduced from the scenarios used by Applicants that postulated loss of core

cooling. (Finding 154). That assumption was necessary in order to produce a degraded core in the models and to produce a 75% metal-water reaction. That hypothetical is not improper since the hydrogen rule postulates the need to accommodate hydrogen generated from a 75% metal-water reaction without specifying the mechanism of generation. It is fruitless to compound hypotheticals, however, by deducing spray unavailability from a hypothetical loss of coolant. OCRE's assertion concerning spray unavailability is without merit and outside the scope of a test of compliance with the Commission's new rule on hydrogen control. The proper vehicle for challenging spray availability is by a separate admitted contention.

Suppression Pool Bypass

OCRE raised the question through its cross-examination of the possibility of hydrogen releases entering the containment or wetwell without first traversing the suppression pool. OCRE cites its Exhibit 21 at 197-198 correctly for the proposition that severe challenges to containment integrity could occur if hydrogen bypasses the suppression pool and leaks directly into the wetwell. OCRE calls for added analysis of the effects of drywell leakage on containment responses to a degraded core accident. However, the Sandia report goes on to say that the possibility of significant hydrogen releases directly into the wetwell/containment without first traversing the suppression pool is apparently extremely unlikely.

Applicants agree that leakage of hydrogen through the drywell leak paths could occur and that on the order of 14 to 19% of the total hydrogen could bypass the suppression pool. They conclude, however, that hydrogen leakage through the drywell leak paths in that amount could not affect the operation of the hydrogen control system or their conclusions. It is apparent that the scenario proposed by OCRE has been analyzed. The Applicants also considered an analysis that was performed by General Electric for its assessments in a small break LOCA of the effects of hydrogen bypass through the drywell. That analysis found that drywell bypass leakage is of no concern to the operation of the PNPP distributed igniter system. Even if hydrogen were to leak out through the drywell wall, the hydrogen transport and combustion characteristics analyzed in Perry's preliminary evaluation would not change. Richardson, Buzzelli, Tr. 3499-3501, 3615-16; Fuls, Holtzclaw, Tr. 3628-29; Pratt, Tr. 3726-27.

EQUIPMENT SURVIVABILITY

OCRE urges the Board to find Applicants' analysis of equipment survivability to be inadequate based on its proposed finding that the CLASIX-3 analysis itself was inadequate. (OCRE Proposed Opinion at 45). Equipment survivability analysis depends on containment environmental temperatures and pressures calculated by CLASIX due to hydrogen burning. These quantities are used in subsequent codes to calculate equipment temperatures which are then compared with equipment qualification temperatures. (Findings 136-137). We cannot, however, reject the

equipment survivability analysis since we do not find that the CLASIX-3 analysis is so flawed as to be rejected outright.

Applicants' principal method for demonstrating equipment survivability is by referencing the Grand Gulf equipment survivability analysis. As we have indicated, such referencing is permitted by the newly promulgated hydrogen rule. Confirmatory calculations for specific pieces of equipment in Perry have been made which show that the Applicants' conclusion that the Grand Gulf results will bound the Perry results is well founded. (Findings 136-138). OCRE has not developed any basis for challenging the HEATING 6 computer code that is used for computing equipment temperatures at Perry. We conclude, therefore, that Applicants, in their preliminary analysis, have made an adequate demonstration of the survivability of equipment in the Perry containment.

The Board understands, as do the Applicants and Staff, that computations produced by the CLASIX-3 code have uncertainties in the results. We conclude that these uncertainties do not constitute fatal flaws. We are satisfied that the Applicants and Staff are making substantial efforts to improve computer codes for the prediction of containment response in hydrogen release scenarios and we expect that these will be used in the final analysis.

The existence of uncertainty dictates that we cannot find reasonable assurance of safety for operation in the interim based on model calculations alone. We conclude, however, that it is appropriate to also rely on the fact that the analyses were consistently performed

in conservative manner and that margins of safety exist. Thus, we conclude that CLASIX-3 computations, while not completely accurate, bound the conditions that would actually prevail during a hydrogen burn event in Perry containment. (Findings 133-143).

The Staff has found that seals for locks and hatches and the transformer for the igniter assembly are not yet temperature qualified. CEI is required by 10 CFR § 50.49(i) to justify interim operation of this equipment and provide qualification for it by November 30, 1985. (Staff Proposed Finding 23).

We have found, however, that the Staff has acknowledged in its SER that certain components in containment have very low qualification pressures but that it has an inadequate explanation on the record for its acceptance of that fact. (Findings 139-142). We are not convinced from the Applicants' own analysis that compressors having a 24 psig qualification pressure are safe from failure in an environment calculated by CLASIX to have projected peak pressures of 21.2 psig. If we knew these numbers exactly, of course, the reasoning would be correct and we understand that Staff and Applicants believe the cited pressures are upper bound estimates of containment pressures. However, our assessment is necessarily subjective and we conclude that in this case the actual survival pressure approaches with small margin a calculated pressure having unknown bounds of uncertainty. Neither are we comfortable with the assertion that the active components of unqualified equipment will not be exposed to peak pressures. Thus, we conclude that the Applicants should make further confirmatory analyses of the

equipment which has not been qualified for pressure survivability in containment and of equipment that has, in our view, inadequate margins of pressure survivability. These should be reviewed and approved by the Staff prior to exceeding 5% power at Perry.

Diffusion Flames

OCRE asserts that the possibility of diffusion flames occurring in a degraded core accident pose another form of thermal threat to equipment and penetration seals. Diffusion flames are continuously burning standing flames that could occur at the surface of the suppression pool. The evidence shows that penetration seals would not reach a high temperature during hydrogen deflagration; there is a significant margin between the temperature from hydrogen burning and the qualification temperatures of the seal materials. This is because the seals are next to a large mass of metal which acts as a heat sink. In addition, the sealing material for the equipment hatch is between the flange materials, which is essentially outside of the containment structure, and would not be exposed directly to the hydrogen burning environment. Additionally, the personnel hatch has two doors, one inboard and one outboard, and only the inboard door would be exposed to the possibly high temperatures from hydrogen combustion. (Finding 113).

The Owners Group plans to conduct quarter scale testing of diffusion flames and to submit the results in its final analysis. The Group, however, will not consider diffusion flames resulting from a release history corresponding to a 75% metal-water reaction in its

tests. The reason is that the rates of release and the volume of release required by that scenario would produce an unrecoverable accident. (Findings 149-152). This is not realistic for conformance to the rule which is limited to consideration of a recoverable degraded core accident scenario. OCRE urges that we cannot ignore this matter and that we cannot defer its consideration to the final analysis. In its view, Applicants' plans for research constitute a defiance of the regulations which require evaluation of equipment survivability under a 75% metal-water reaction scenario.

We do not accept OCRE's argument that the scenarios that are going to be adopted for future research constitute a violation of NRC regulations. All the hydrogen rule requires is that a 75% metal-water reaction be considered. It does not specify particular scenarios and is silent on the question of diffusion flames. We do not expect that Applicants will perform unrealistic research. There is no evidence to indicate that a hydrogen release accident is likely to proceed through a 75% metal-water reaction with nothing but diffusion flames produced. Indeed, the evidence shows that hydrogen will escape for the most part to the containment atmosphere where it will be burned by multiple deflagrations. We consider the diffusion flame scenario to be an investigation of one among many possible occurrences during a hydrogen release event. Nothing in the hydrogen rule requires the Applicants to commit to evaluating the thermal environment and equipment response resulting from diffusion flames using a hydrogen release history

resulting from a 75% metal-water reaction. (OCRE Proposed Opinion at 47).

OCRE raises the issue of whether other effects of hydrogen control system operation will aggravate the course of an accident. For this proposition it cites three possible events that would have adverse effects on the course of an accident and could arise because of hydrogen combustion: these are excessive drywell pool loads, loss of decay heat removal capability and secondary fires in containment. (OCRE Proposed Opinion at 47-50).

Drywell Pool Loads

OCRE's challenge concerning drywell pool loads is based on a paper presented at a scientific conference by one of the Applicants' witnesses. The paper outlines a scenario by which differential pressures between containment and drywell could result in a violent overflow of the suppression pool into the drywell. The analysis cited no adverse effects on essential equipment from such an occurrence but indicated that the effects would be evaluated. That scenario was evaluated as part of a number of sensitivity studies using conservative assumptions in the CLASIX model. These studies involved analyses beyond the design base case. The design base case evaluated the potential consequences of any reverse or forward pool swell. OCRE brought out on cross-examination that the Hydrogen Control Owners Group has a plan for long-term investigation of pool-swell loading from hydrogen combustion. However, in the case of Perry, the preliminary evaluation had considered

differential pressures and those pressures were found to be less than the ones analyzed in the design basis case. (Finding 153). The Board concludes that drywell pool loading is a design basis consideration which, for the most part, is beyond the scope of matters relevant to hydrogen control. It is analyzed separately in the Applicants' overall safety analysis. There is linkage with the hydrogen control problem because hydrogen combustion during the degraded core event could produce differential pressures between containment and drywell. However, those differential pressures have been considered and have been found to be less than the design basis case.

The Board does not find it unusual or alarming that sensitivity analyses utilizing very conservative assumptions might well show pool loading conditions as described in Applicants' paper. We disagree with OCRE, however, that such analyses need be accepted as literal descriptions of likely events during a degraded core accident at Perry. We conclude that the possibility of pool swell during a degraded core hydrogen release incident has been considered in Applicants' preliminary analysis and that it does not hold significant potential for further aggravating the course of an accident at Perry.

Decay Heat Removal

OCRE asserts that in a degraded core accident, hydrogen combustion adds heat to the containment atmosphere in addition to the decay heat being added to the suppression pool. It is therefore appropriate, in its view, to examine the decay heat removal capability in a situation

where heat from hydrogen combustion must also be removed from containment. The evidence shows that there are redundant safety grade residual heat removal (RHR) loops available for suppression pool cooling. The active components for the RHR system are located outside containment and would survive hydrogen burning. Even if both RHR loops were lost for suppression pool cooling, elevated pressure in containment could be handled by the containment spray which passes through heat exchangers before being sprayed. Thus, long-term decay heat removal would be assured.

The Board concludes that active heat removal systems exist which would remove decay heat and the heat added by hydrogen combustion from the suppression pool. Beyond that, the PNPP evaluation shows that hydrogen burning occurs early in the scenarios and that peak suppression pool temperature occurs after several hours. Thus, hydrogen burning would have an insignificant effect on the overall suppression pool temperature. (Finding 155).

Mr. John M. Humphrey is a former GE engineer who expressed some safety concerns regarding Mark III containment. OCRE referenced a so-called Humphrey analyses which indicated the containment spray operation might significantly reduce suppression pool mixing effectiveness and lead to pool stratification. The Board, however, is satisfied that the so-called Humphrey concerns were evaluated for Grand Gulf and that they have been adequately considered not only by the Applicants but by the NRC and the Advisory Committee on Reactor

Safeguards. These concerns were determined not to raise significant safety issues. (Finding 156).

We conclude that problems associated with decay heat removal would not be further aggravated by hydrogen combustion in a degraded core accident.

Secondary Fires

OCRE asserts that Applicants have not evaluated the potential for secondary fires in containment that could be initiated by hydrogen burning. Such fires, according to OCRE, would affect containment pressure and temperature profiles. There is no evidence in the record in support of these assertions. The potential for secondary fires, particularly cable fires, has been considered in tests at Fenwal and the Nevada test site. Cable burning has been shown only for hydrogen concentrations above 10%; although some burning was shown in other tests, they were not of the type of wire used in the Perry containment. Evaluations conducted for Grand Gulf have shown that there is no potential for secondary fires for the temperatures predicted by CLASIX-3. The Applicants are confident that secondary fires will not occur because the temperature peaks predicted for PNPP are intermittent, of short duration, and not uniform throughout containment. The Board concludes that the risk from secondary fires initiated by hydrogen combustion in containment is low and that its consideration for the preliminary analysis by the Applicants has been adequate. (Findings 147-148).

The Scope of the Preliminary Analysis

We stated at the outset that we did not intend to define the precise boundaries that separate the preliminary analysis from the final analysis because the Applicants submitted an extensive and detailed analysis of hydrogen combustion during degraded core events at Perry. We permitted OCRE considerable latitude in its cross-examination during the hearing because these boundaries were undefined. OCRE took advantage of the opportunity and included many subjects in its cross-examination that may well have gone beyond the bounds of a reasonable preliminary analysis on the comparatively narrow safety issue of hydrogen control. Now, having reviewed the entire record, we conclude that the basic questions that have to be answered at the preliminary analysis stage are whether an igniter system has been installed, whether it will function as designed, whether hydrogen will burn as predicted, and whether the containment and essential equipment will retain their integrity under the pressures and temperatures predicted. From this we conclude that it was appropriate to permit cross-examination on the CLASIX code which predicted the temperatures and pressures in containment. Cross-examination on defective welds was also appropriate since this appears to be directly related to containment integrity. We also believe it was appropriate to inquire into equipment survivability and the basis for the Applicants' confidence that essential equipment in containment would survive the temperatures and pressures produced by hydrogen burning.

The hydrogen rule for Mark III containments, however, is not the vehicle by which we test the comprehensive safety analyses of other systems that are performed by Applicants and Staff. Many of the systems that OCRE wanted to include within the scope of the hydrogen control issue are analyzed elsewhere in other parts of the overall safety analysis. Even though linkages exist we do not believe that an analysis of hydrogen control raises each and every other aspect of the Applicants' overall safety analysis. For example, we conclude that the issue of hydrogen control does not raise the question of containment spray availability during a hydrogen event. This is not because spray availability is unimportant, but because that issue is analyzed elsewhere in the overall safety analysis. Similarly, the broad issue of station blackout, while important, is not raised by the hydrogen rule, even though linkages exist for which the staff may require additional analysis.

We do not similarly analyze each of the other systems that OCRE challenged in this case since the foregoing is sufficient to highlight our concerns. The hydrogen control issue can become unnecessarily complex because of the number of systems and components in containment which could be brought into play during a degraded core event. We do not believe that safety is well served by conducting fragmented analyses of complex systems and we conclude separate contentions should be the means for challenging separate engineered safety systems and that the reliability of these systems should not be treated as collateral to a hydrogen control contention.

The analysis of hydrogen combustion during degraded core events is a new task in overall reactor safety assessment. New cases will produce new thinking on the issue which should be integrated with past analyses as provided for in the rule. In an area of actively developing technology, however, we are not comfortable in relying alone on references to similar reactors that have been licensed in the past. It would have been helpful had the Applicants and Staff referenced other portions of the Perry FSAR or SER for its confidence that other containment systems that could be linked to the hydrogen control issue would function and that the linkages to hydrogen control had been considered. We conclude, however, that the Applicants' analysis of hydrogen control at Perry was thorough and reasonably within the scope of the preliminary analysis prescribed by the hydrogen rule for BWR Mark-III containments.

Conclusions

Based on the entire record in this matter the Board concludes that the Applicants have made an adequate demonstration in its preliminary analysis that: a hydrogen ignition system has been installed at Perry; that the system will ignite hydrogen in lean mixtures, which will prevent the accumulation of large amounts of hydrogen in containment; that the Perry containment will survive the controlled combustion of hydrogen without failure; that essential equipment in Perry containment is qualified to withstand the temperatures and pressures likely to be generated during hydrogen combustion.

Our confidence that the Perry containment will survive hydrogen combustion is not based on computer analysis of containment response alone. We agree with OCRE that modeling of containment response is a process having some uncertainty. We are satisfied, however, that work is progressing on the problem of containment response analysis and that progress has been made since the time the Sandia report for Grand Gulf was published. We do not accept CLASIX-3 results regarding temperatures and pressures as exact representations of what would take place during hydrogen burns in containment. We do accept that these analyses are based on conservative assumptions, and are more likely to over-estimate than under-estimate the response. Succeeding versions of other models analyzed in the Sandia report also show that earlier models erred on the side of conservatism and that more recent results tend to show less severe conditions than earlier efforts.

In the face of analytical uncertainty, however, which we believe still lingers, we must also find that other aspects of the analysis are conservative or that margins of safety exist which could add to confidence that containment will not fail during hydrogen combustion. We find that substantial margins do in fact exist on containment strength. The ASME standards for analyses of stresses are conservative and show that there is likely added strength of PNPP containment and its penetrations beyond that performed under the service levels prescribed by the rule. Furthermore, there is considerable margin between the peak pressures shown by CLASIX and the ability of the containment to withstand internal pressure. The first yielding of containment

conservatively could occur in a penetration at approximately 50 psig while the CLASIX-3 model shows that peak internal pressures would occur at slightly over 21 psig. The yield strengths are likely higher and the pressures generated are likely lower than those stated. Thus there exists a substantial, though not precisely measured, margin between the peak pressure likely to be generated and the pressure-containing capacity of the most limiting penetration in containment.

The overall shell of the containment is considerably stronger than the limiting penetration. We conclude that there is virtually no likelihood that the containment shell itself would fail catastrophically during a degraded core accident at Perry when the hydrogen ignition system functions as designed.

Based on the foregoing considerations, the Board finds reasonable assurance that the hydrogen igniter system at Perry will function as designed and that the containment will retain its integrity during a degraded core accident which generates up to 75% of the equivalent metal-water reaction.

The Applicants' analysis of its system is in compliance with 10 CFR 50.44(c)(3) as it addressed the preliminary analysis, provided applicable requirements stated herein are met. The items that the Applicants and Staff have agreed to defer to the final analysis were technically reasonable and were not selected, as alleged by OCRE (see OCRE Proposed Opinion at 52), for the mere convenience of Applicants and Staff. (Findings 157-158). The deferred items represent events of remote likelihood or refinement of existing analyses and we see no basis

for deferring reactor operation pending completion of those analyses. The Board finds that Applicants have carried their burden of proof on issue 8.[12]

C. DIESEL GENERATORS (Issue 16)

OCRE and Sunflower filed contentions concerning the PNPP diesel generators. By order of July 28, 1981, the Board dismissed both contentions for lack of specificity. OCRE submitted a motion to resubmit its Contention #2 on September 26, 1983. By Order of December 23, 1983 the Board granted OCRE's motion in simplified form. As accepted by the Board the new Contention 16 reads as follows:

> Applicant has not demonstrated it can reliably generate emergency on-site power by relying on four Transamerica Delaval diesel generators, two for each of its Perry units.

The diesel generators installed at the Perry plant (PNPP) are meant to supply emergency power for the plant safety systems in case offsite power is not available. (Finding 159). This requirement is set forth in General Design Criterion (GDC) 17. Various models of Transamerica Delaval, Inc. (TDI) diesels are in use at a number of nuclear plants. When a TDI diesel at the Shoreham nuclear plant suffered a catastrophic failure of its crankshaft in 1983, a "TDI Owners Group" was formed for

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12 While our findings are adverse to OCRE, we compliment its representative for her efforts in making a substantial contribution to the record which aided materially in clarifying complex issues that surround the hydrogen control problem.

the purpose of developing a program to qualify the reliability of the units. The proposed program, which included design review, inspection and testing, was submitted to NRC in March 1984 and was approved in August 1984. (Finding 160). The basic issue involved in this contention is the adequacy of the Owners Group plan.

Applicants and Staff presented panels of expert witnesses.[13] The witnesses' testimony described in detail the Owners Group plan and the implementation of it by PNPP. Phase I of the plan was to qualify components which, based on actual operating experience with TDI diesels, exhibited possibly generic problems. Sixteen components were identified as being of concern and were subjected to testing and detailed design review. Pacific Northwest Laboratory (PNL) reviewed the Owners Group effort for the Staff and Southwest Research Institute (SwRI) was retained by the Applicants to independently review the analyses. Both of these efforts were performed independently of TDI. (Findings 161-165).

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13 Applicants' witnesses: Edward C. Christiansen, Perry senior design engineer; John C. Kammeyer, a Program Manager, TDI Owners Group; and Charles D. Wood, III, representing Southwest Research Institute (SwRI). Staff witnesses: Dr. Carl H. Berlinger, NRC Project Group Manager of the TDI Diesel Task Force; Drew Persinko, a member of the NRC TDI project group responsible for review of the PNPP diesels; Dr. David A. Dingee, a representative of Pacific Northwest Laboratory (PNL) which was providing technical assistance to the Staff in its review of TDI engines; and four diesel engine and metallurgy consultants to PNL, Dr. Spencer A. Bush, Howard M. Hardy, Adam Henricksen and B. J. Kirkwood. Ohio Citizens for Responsible Energy (OCRE), lead intervenor for Contention 16,

(Footnote Continued)

Phase II of the plan was aimed at all other components which were important to engine operability. These components were subjected to design review and/or quality revalidation (DR/QR). A "lead engine" concept for component testing was also employed, in which results of tests on engines using the same components could be applied to the same model engine at other sites. (Findings 166-167).

A revalidation phase required that the engines be completely disassembled, inspected and reassembled under the owner's quality control and quality assurance program. The final phase of the Owners Group plan is an ongoing one: it involves the preparation and implementation of comprehensive maintenance and surveillance programs. As the Board understands it, this will be a "living document" which will be revised and refined as operational experience dictates. (Findings 168-169).

Both Applicants' and Staff's witnesses testified that the following had been performed at PNPP:

1. All requirements of the Owners Group Phase I, including disassembly, inspection and replacement/repair/modification as necessary, of the 16 critical components had been completed (Kammeyer, ff. Tr. 2179 at 12, Tr. 2182; Christiansen, ff. Tr. 2179 at 11, Tr. 2499);

(Footnote Continued)
presented no direct evidence, but cross-examined Applicants' and Staff's witnesses.

2. The Phase II design review and quality revalidation had been completed on all applicable components of the Unit 1 engines (Finding 167);

3. The Unit 1 engines had operated for some 20 hours at the time of the hearing (May 1985) with no mechanical problems encountered (Christiansen, ff. Tr. 2179 at 31, 32); and

4. Applicants have developed and is performing maintenance and surveillance programs based upon recommendations by the Owners Group (Finding 169).

OCRE submitted extensive proposed findings, including not only findings in opposition to the Applicants', but also findings where OCRE agrees with Applicants. The Board commends OCRE for this helpful practice. The Board sees no reason to discuss in this opinion items on which the Applicants, Staff and OCRE agree and are concurred with by the Board.

OCRE attacked the credibility of the Owners Group on grounds that it was a "lobbying force" rather than a "disinterested technical organization." OCRE references several Owners Group internal documents and meetings with NRC Staff on various subjects relating to the Owners Group program as evidence of an organization concerned more with its commercial interests than an objective requalification of TDI's diesel engines. (OCRE Proposed Opinion at 13-16; OCRE Proposed Finding at 57; OCRE Ex. 2, 4, 5 and Staff Ex. 1 at 3). The Board finds no violation of 10 CFR 50, Appendix B, Criterion I in the activities of either the Applicants or the Owners Group as alleged by OCRE in this contention.

We find no need to consider the motivation of the members of the Owners Group--the plan they produced is the Board's concern, and to that, the plan speaks for itself.

We have not burdened this decision with a pointless finding-by-finding discussion of irrelevant matters. Those concerns which we consider to be of such significance as to possibly affect our decision are discussed below.

General Design Criteria Nos. 1 and 17

OCRE maintains that the TDI QA/QC programs have been shown to be inadequate, and alleges that the Applicants have not complied with either GDC-1 or 17 of Appendix A in 10 CFR 50. (OCRE Proposed Opinion at 11-13). GDC 1 requires that systems and components important to safety be designed, fabricated and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 17 requires an onsite electric power system with sufficient independence, redundancy and testability to perform its safety function assuming a single failure. The safety function is to assure that design limits are not exceeded during operation and that the reactor's core is cooled while maintaining containment integrity.

OCRE contends that the performance specifications for Perry's diesel engines, SP-562 and its attachment SP-706, should be the standard by which the design criteria are measured. (OCRE Proposed Opinion at 11-12). Although the Board agrees with OCRE's critical assessment of TDI QA/QC, we must view the entire record as the measure for evaluating

the acceptability of Perry's TDI diesel engines. The question here is whether the requalification of Perry's engines through the Owners Group program meets the requirements of GDC 1 and 17. The TDI QA/QC program was audited in 1982 and was found to be adequate for the manufacture of spare parts. Nevertheless, PNPP assigned its own quality assurance representative full-time to TDI to monitor any safety-related engine component ordered. (Christiansen, ff. Tr. 2179 at 16, Tr. 2236-37, 2267).

Engine Maintenance and Surveillance

OCRE charges that PNPP's surveillance and maintenance program appears to be directed at only meeting requirements of 10 CFR Parts 21 and 50.55(e). (OCRE Proposed Opinion at 19). While the requirements of the regulations are in fact met, the PNPP program goes much farther, as it includes all the recommendations in the DR/QR report as well as those proposed by PNL and SwRI. (Finding 169). OCRE also states that there is "no assurance that the maintenance and surveillance program will exist for the life of the plant, or that it will be implemented at all." (OCRE Proposed Opinion at 19). The Board cannot find the basis for this statement in the record, but does find that testimony presented by the witnesses indicates otherwise. See, e.g,. Staff Ex. 4, Tr. 2468. The Board concludes that the PNPP maintenance and surveillance program at PNPP provides assurance of the reliability of the diesels.

OCRE has cited a failure in the Owners Group program in not preventing a crankshaft oil plug defect at Gulf States Utilities River

Bend reactor. The deficiency was reported by TDI on March 18, 1985 in a 10 CFR Part 21 letter to the NRC. (OCRE Ex. 8). It is not clear that the defect referred to, which apparently was a crack and not a failure in the oil plug, is a potential problem at Perry. Nevertheless, it is being subjected to inspections by the Applicant. (Finding 170). The Board concurs with the view that the Owners Group program was never expected to eliminate all possible failures in TDI's diesel engine components. (Kammeyer, Tr. 2230-31; Christiansen, Tr. 2262-63).

Staff Review of Owners Group Program

OCRE criticizes the Staff's review of the adequacy of Perry's diesel engines for permitting plant operation prior to completion of the NRC's final approval and allowing Phase II requirements and diesel tests and inspections to be conducted after the reactor's first refueling outage. This, the Intervenor concludes, is tantamount to "interim licensing" which to OCRE is an illegal act that violates the dictates of GDC 1 and 17. (OCRE Proposed Opinion at 19-23). The basis for OCRE's position is its interpretation of Staff's testimony that outstanding items in SSER-6 would have to be resolved prior to operation above 5% of power and that maintenance and surveillance programs could be deferred until the first fuel outage. (Stefano, Tr. 2473-74; Berlinger, Tr. 2305; OCRE's Proposed Opinion at 20-23).

Interim licensing is a concept developed by the Staff to apply to those plants who would otherwise be eligible for near-term operating licenses prior to their completion of the Owners Group program. (Staff

Ex. 1, ff. Tr. 2284 at 13-19; PNL-5161 at 19). However, the Intervenor concedes that Perry has not applied for an interim license (OCRE Proposed Opinion at 20) and most of the outstanding items in the Staff's evaluation have been considered and handled by the Applicants. (Applicants' Rebuttal Testimony, Tr. 2489-2509). Additionally, PNPP has completed its Phase II program and the Staff will perform a reassessment after the first refueling outage to audit the reviews previously performed on the diesel engines by the Owners Group and the Staff. (Berlinger, Tr. 2305).

The Staff review of the adequacy of Perry's diesel engines to perform reliably has been extensive. It has included (1) a review of the Owners Group program, (2) a review of the adequacy of Phase I components at Perry and the results of the engines' disassembly and inspection, (3) Phase II DR/QR review by PNL of Comanche Peak's diesels and their similarity to Perry's, (4) previous conclusions by Staff on similar engines at Comanche Peak, Grand Gulf and Catawba, (5) preliminary findings by PNL on Phase I generic components, (6) proposed preoperational testing program at PNPP, (7) Applicant's commitment to a maintenance and surveillance program which will be reviewed by Staff, and (8) Applicants commitment to a torsiograph test and several other technical commitments dealing with cylinder heads, push rods and water pump shafts. (Staff, ff. Tr. 2281 at 12; Staff Ex. No. 2 at 1-2).

Phase I Components

Of the sixteen Phase I components, OCRE does not question the adequacy of the design or construction for eleven of them. (OCRE Proposed Opinion at 24-25). However, it expresses reservations about the maintenance and surveillance program as applied to these components. (Id.; OCRE Proposed Findings 33-64). The Board, as a result of the assurances and commitments by the Applicants on the record, finds no reason exists to question the implementation of an adequate maintenance and surveillance program. The record shows that the requirements set by the Owners Group plan, the DR/DQ report, PNL, SwRI and the Staff have either been satisfied or committed to by PNPP. (Findings 171-183). The Board finds that each of these components is suitable for its intended service.

The components contested by OCRE were the engine base and bearing caps, the turbocharger, the crankshaft, the cylinder block and cylinder liners.

Engine Base and Bearing Caps

Although OCRE did not contest design adequacy, it contends that the material properties of the base need to be analyzed to assure that the specifications are met. (OCRE Proposed Opinion at 25-26). Its rationale for this position is the fact that the cylinder block, for which the Owners Group requires such analysis, is made from the same material. The Board is unpersuaded by this argument. The record shows that only one failure of an engine base has been reported for the

hundreds of engines in operation. The Owners Group considered that due to the low loading of the base and its excellent operational history that routine maintenance and inspection was adequate to assure the reliability of the base. (Findings 184-185). The Board agrees and finds Applicants' procedures acceptable.

Turbochargers

OCRE contends that the TDI diesel's turbocharger is unsuitable for nuclear use because the fabrication technique used in assembly of the nozzle ring vanes does not permit the testing of the vanes for incipient failure. (OCRE Proposed Opinion at 26-27). Staff agrees that due to visual and physical impediments such testing is not possible, but argue that enhanced surveillance and maintenance of the turbocharger will provide reasonable assurance of turbocharger reliability. Staff points out that while it is possible for vane failure to seriously damage the rotor, operational experience has shown that no turbocharger in nuclear service has shown severe damage from vane failure. (Staff, ff. Tr. 2281 at 39-40). The Owners Group analysis of this problem also recommends that the turbine exhaust temperatures be monitored, as high temperatures indicate a condition which might lead to a more likely failure situation. (Kirkwood, Dingee, Berlinger, Tr. 2354-56). The Board agrees that it is unfortunate that the nozzle ring vanes cannot be tested for incipient failure by conventional non-destructive testing methods. However, on the basis of the record before us, which includes evidence of testing, commitments to the Owners Group recommendations

on surveillance and maintenance, and the operational experience of these turbochargers, we find there is reasonable assurance that the turbochargers can be relied upon to perform adequately. (Findings 186-187).

OCRE also questioned whether the turbochargers have been properly aligned to their mounting brackets. (OCRE Proposed Opinion at 27). Staff testified that the mounting is adequate to prevent problems, and that the vibration testing to be performed will confirm it. (Finding 188). The Board agrees that vibration testing is adequate to determine any misalignment problem.

Crankshaft

OCRE's concerns with the PNPP crankshaft are several: the results of tests on the San Onofre diesels, the use of the Diesel Engine Manufacturers Association guidelines rather than the European Ship Classification Societies rules, the lack of Staff review of the PNPP torsiograph tests, and the Staff's recommendations for limiting the PNPP diesel's speed range and an evaluation of the effects of cylinder imbalance. (OCRE Proposed Opinion at 27-29; OCRE Proposed Finding 83).

The results of the San Onofre inspections and tests are of dubious value in any direct comparison with the PNPP diesels. The San Onofre engine is a DSRV-20 model, as contrasted to the PNPP DSRV-16-4 models, and has a significantly different crankshaft. The Board notes, however, that Applicants have not ignored the San Onofre problems. The major problem was cracks in the crankshaft around the journal oil holes and

PNPP inspected its crankshaft journal oil holes and subjected them to eddy current testing to show that they were free from such defects. (Findings 180-190). OCRE states that Owners Group analytical and testing procedures failed to predict the San Onofre problem. (OCRE Proposed Opinion at 28). However, the cracks in the San Onofre crankshaft were discovered during onsite inspection (Hardy, Tr. 2326) and were confirmed during onsite torsiograph testing. (Berlinger, Hardy, Tr. 2327-31).

The European Ship Classification Societies rules, which are applied to maritime diesels and which OCRE thinks should be applied to PNPP diesels, are quite conservative. They are meant to apply to continuous-use, variable-speed engines which are of various sizes. (Finding 191). This contrasts with the Diesel Engine Manufacturers Association (DEMA) requirements, that must be met by the PNPP diesels which are land-based, intermittent-use, constant-speed machines. The PNPP diesels comply with DEMA recommendations, and, in fact, go beyond them, as noted below. The Board finds that the DEMA recommendations, particularly with the additional tests done by PNPP, are suitably conservative.

Torsiograph tests on the PNPP diesels have been completed which show that DEMA guidelines have been met. (Finding 192). At the time of hearing the Staff had not reviewed the tests, so this necessary review was carried as an "open item" in the Staff SER, Supp. 4. The Board has no reason to believe that this item will not be resolved before plant operation and notes that the Staff and its consultants have other bases

for concluding that the PNPP crankshafts will be adequate. (Finding 193).

The Staff has recommended that the operating speed range of the diesels be limited. PNPP standards were set limiting the speed range to between 90% to 110% of rated speed. Although it was found that the PNPP crankshaft has a fourth order critical speed which is within this ±10% range, it was determined that the stress imposed was well within DEMA guidelines. To minimize steady operation at or near this critical speed the engine's governor has been set to allow a speed range of -1/2% to +6% when manually controlled and the diesels are not attached to the grid. (Finding 194).

The Staff also recommended that tests involving cylinder imbalance be made. Although not a DEMA requirement, the tests were conducted by cutting the fuel supply to one cylinder, and the Staff is evaluating the results. (Finding 195).

The Board concludes that the record shows that the crankshafts are suitable for their intended service.

Cylinder Block

OCRE's concerns with these components are principally with block cracking and the Owners Group inspection interval. (OCRE Proposed Opinion at 29-31). OCRE concludes that because the Owners Group recommends that the blocks be inspected after 572 hours of operation that the engines cannot meet the requirement for "continuous" operation at rated load. The Board views this conclusion as purely a misapplied

semantic exercise. While the length of time that offsite power might be lost cannot be specifically defined, the length of time emergency power is needed for core cooling purposes is no more than a week. (Kammeyer, Tr. 2221-22). The Board finds that the diesels can fulfill their basic purpose even with the 572 hour inspection limit.

There are four types of cracks which were found in the Shoreham engine blocks. (OCRE Proposed Opinion at 29; Wood, ff. Tr. 2179 at 56). All of these cracks connect with the top surface of the block and could be detected by surface inspection. (Wood, ff. Tr. 2179 at 56). Applicant has conducted a 100% inspection of cylinder block tops and liner landings, and no evidence of cracking was discovered. (Finding 196).

OCRE stated that the Owners Group analysis of the block "found that ligament cracks are predicted to occur, and that their presence then increases stresses in the block and increases the likelihood of stud-to-stud cracks." (OCRE Proposed Finding 87; OCRE Proposed Opinion at 29). After review of the appropriate document, Staff Ex. 5, at 4.5, the Board determined the actual statement reads as follows:

> - Initiation of cracks in the ligament between stud hole and liner counterbore was predicted to occur after accumulated operating hours at high load and/or engine starts to high load. These cracks were considered to be benign because the cracked section is fully contained between the liner and the region of the block top outside the stud hole circle. Field experience is consistent with both the prediction of ligament cracking and the lack of immediate consequences.
>
> - The presence of ligament cracks between stud holes and liner counterbore increases the stress and the

> probability of cracking between the stud holes of adjacent cylinders, and stud-to-stud cracks are predicted to initiate after additional operating hours at high load and/or engine starts to high load.

In consideration of the above verbatim statement OCRE's interpretation is erroneous and the Board therefore finds that OCRE's proposed finding has no probative value.

OCRE observes that actual loads to be carried by the diesel generators have not been determined (OCRE Proposed Finding 7) and that additional loads may be added in the future. (OCRE Proposed Finding 8). Applicants plan to determine actual loads by testing in the near future and notes that any additional loads in the future would require an FSAR amendment and would be reviewed by the Staff. (Finding 197).

OCRE cites a disagreement between Dr. Bush, witness for the Staff, and the Owners Group over the acceptable length of stud-to-stud cracks in the block. (OCRE Proposed Finding 92). However, Dr. Bush's position assumes that ligament cracking already exists. (Finding 198). As no cracks have been found during PNPP's inspection of the block, the difference of opinion between Dr. Bush and the Owners Group is irrelevant.

Cylinder Liners

In order to reduce the possibility of cylinder block cracking, the cylinder liner proudness has been reduced. OCRE argues that the effectiveness of reduced proudness has not been determined. (OCRE Proposed Finding 95). The Board notes that while no quantitative

determination of the effectiveness of this procedure has been made (Kammeyer, Tr. 2508), witness testimony was submitted that reduced proudness, though still greater than zero, reduces pressure stress in the block and should reduce the probability of cracking. (Finding 199).

In consideration of the record before us, the Board finds that the cylinder block and liners are adequate for the intended service.

Phase II Components

Dresser Couplings

OCRE believes that the Dresser couplings which are used in the water jacket and lube oil systems should be replaced prior to plant operation. (OCRE Proposed Opinion at 31). Staff agrees that it would be better to replace them prior to plant operation but that in its judgment replacement could be delayed until the first refueling outage. (Staff, ff. Tr. 2281 at 52). Applicants have committed to monitoring the couplings for leakage and replacing them as necessary. (Christiansen, Tr. 2495). The Board agrees with the Staff that it would be better to replace the couplings before operation, but finds that, with careful monitoring by PNPP, replacement could be delayed until the first refueling outage.

Foundation

OCRE believes that there is insufficient contact between the engine base and its foundation. (OCRE Proposed Opinion at 32). The PNPP requirement was for 85% contact between the engine base and the chock

plates. The chock plates were inspected and an engineering evaluation was made for plates with less than 85% contact. Hot and cold crankshaft deflection measurements were then conducted which showed that the engine was well supported. (Finding 200). The Board has no evidence from the record before us to suggest that this is not an adequate procedure, and therefore finds that the foundation is acceptable.

Conclusions

The Board finds that the Owners Group plan for the PNPP diesel generators, if followed, provides the requisite assurance of the reliability of the emergency power system. It is a straightforward program designed to eliminate generic design and maintenance problems. One purpose of the program is to make sure that any failure of a unit would be limited to some random or unpredictable event. It does not--and, indeed, cannot--claim that failures will never occur. PNPP has committed to following the Owners Group plan, as well as recommendations by the Staff as set forth by PNL. The Board finds that the Owners Group plan provides a well-thought-out program which, if implemented properly, provides reasonable assurance that TDI diesels will reliably carry out their intended function. We further find that Applicants' implementation of the plan meets or exceeds the requirements therein and provides further assurance that emergency power will be available when and if needed. We therefore find that Perry's diesel engines have met regulatory quality standards and that emergency power will be available when needed. We conclude that the Applicants have complied with GDC 1 and 17 and Contention 16 is accordingly dismissed.

II. FINDINGS OF FACT

A. EMERGENCY PLANNING
(Issue 1)

1. NUREG-0654 includes NRC/FEMA regulatory criteria which call for a review of evacuation time estimates (ETE) by principal State and local organizations and the inclusion of such comments in the ETE submittal. Shapiro, ff. Tr. 3111 at 3.

2. The HMM Associates, Inc., prepared the ETE study for the Perry Nuclear Power Plant plume exposure pathway, Emergency Planning Zone (EPZ), and has performed similar ETEs at a number of other nuclear power facilities. McCandless, ff. Tr. 2791 at 1; Tr. 2793-94.

3. In preparation of the Perry ETE, officials of HMM met with officials from Ashtabula, Lake and Geauga County Disaster Services Agencies (DSA) and the Sheriffs' Departments. Agreement was reached with the governmental representatives on methodology, input data, assumptions used, and the plans and procedures for the study. McCandless, ff. Tr. 2791 at 2-3.

4. A March 1984 draft ETE was sent to DSA Directors and Sheriffs of the three Counties and to the Ohio State DSA for review and comment. Id. at 3; Shapiro, ff. Tr. 3111 at 3.

5. The Agencies' comments have been reflected in the February 1985 revision of the ETE. McCandless, ff. Tr. 2791, at 3.

6. Comments on the ETE from local and State officials were submitted to the NRC with Revision 4 of the emergency plan on February 20, 1985. Id. at 3; Perrotti, ff. Tr. 3111 at 2.

7. NUREG-0654 provides no indication that specific individuals must be contacted for review and comments on a nuclear facility's ETE and the intent is to have knowledgeable officials carry out this assignment. Perrotti, ff. Tr. 3111 at 3.

8. Both HMM Associates and FEMA rely on local emergency response officials to indicate who should be involved in commenting on ETEs. Perrotti, Tr. 3122; McCandless, Tr. 2812-14.

9. Neither the State nor any of the three Counties within the EPZ have indicated that they were not provided an opportunity to comment on the ETE. Shapiro, ff. Tr. 3111 at 3; see also McCandless, Tr. 2830.

10. HMM Associates met with the three County DSA directors and the County engineers in March 1985 and concurrences were received for the February 1985 revisions of the ETE. McCandless, Tr. 2795-99.

11. There are four classes of emergency action levels (EALs)--Unusual Event, Alert, Site Area Emergency, and General Emergency--which, based on the existence of specific plant conditions, can be declared. Hulbert, ff. Tr. 2965 at 2.

12. Revision 3 of the Perry emergency plan sets forth in Table 4-1 over 200 individual EAL indicators. Thirteen (13) indicators were incomplete due to the fact that values which had to be included were unavailable at that time. A comparable value, however, was specified. Id. at 2.

13. Revision 4 of the Perry plant, issued in February 1985, includes either the required missing indicators or alternate indications that were selected. Id. at 2.

14. The Applicants' emergency classification and action level scheme is now considered complete and adequate. Id. at 3; Perrotti, ff. Tr. 3111 at 3-4.

15. Nomograms are recommended in EPA's Manual of Protective Action Guides but other methods can be used as well. Sternglass, Tr. 2728.

16. Information in the Applicants' emergency plan, Table 1.2, on radiation exposure of fetuses refers to the ingestion pathway area and not the inhalation pathway area with which EALs are concerned. Sternglass, Tr. 2650-54.

17. The conformity of Perry's EALs to NUREG-0654, Appendix I, which provides example initiating condition guidance, is under NRC review. Any discrepencies will have to be corrected prior to licensing. Perrotti, ff. 3111 at 4.

18. NUREG-0654, Criterion H.7, states "Each organization (Licensee, State and Local) where appropriate, shall provide for offsite radiological monitoring equipment in the vicinity of the nuclear facility." Bowers, ff. Tr. 2914 at 2.

19. NUREG-0654, Criterion I.8, which calls for each organization--Licensee, State, and local--to provide for rapid assessments of radiological hazards, suggests the use of mobile monitoring teams to perform this assessment. Id. at 4.

20. The planning standards recommend in NUREG-0654, Criteria I.9 and I.11, that the Licensee and State have the capability to detect and measure radioiodine concentrations in the EPZ and also that arrangements

be made with federal and/or State resources to locate and track an airborne radioactive plume. Id. at 2.

21. A task force from FEMA considered the concept of a system of fixed monitoring locations as a method of estimating the dispersal of the plume and for projecting exposure patterns. However, the system was rejected because of the large number of sophisticated detectors and necessary telemetry required and a flexible system with portable instrumentation was considered more cost effective. Id. at 2-4.

22. There is no regulatory requirement or guidance that each jurisdiction within the EPZ have an independent or a fixed radiation monitoring system. Id. at 2; Shapiro, Tr. 3136.

23. For the Perry facility, a fixed monitoring system would require 103 locations at an installation cost in excess of $2,635,000. The maintenance of such a system would require three full time people and the annual cost of calibrating each unit would be approximately $400 per unit. Bowers, ff. Tr. 2914 at 3, 2916-19.

24. Mobile survey teams that can move to where the plume is located to make measurements are the most effective method for evaluating accidental radiation releases. Bowers, ff. Tr. 2114 at 3.

25. To make meaningful projections for monitoring data, it is necessary to identify the center line of the plume. Mobile monitoring teams have this capability but fixed monitors do not. Dose measurements at fixed monitors could give instantaneous readout but their measurements cannot be extrapolated to determine doses at other locations. Bowers, Tr. 2927-32.

26. The State of Ohio maintains three trained and equipped radiological monitoring teams to respond to any radiation emergency. Cole, ff. Tr. 2835 at 1, Attachments 1, 2 and 3.

27. State teams which are dispatched at the alert stage can reach the Perry site area in 3-3½ hours and will transmit samples and data to the State's emergency operation center for assessment. Cole, ff. Tr. 2835 at 2-3.

28. A State response team supervisor is dispatched by air at the alert stage to monitor general plume direction, its center line and also to direct the monitoring teams' location and activities. A DSA employee who works and lives in Perry could, if necessary, take gross gamma readings prior to arrival of the State teams. The State has more than 25 helicopters available for emergency purposes and their response time to Perry is 1 hour and 15 minutes. Id. at 3-4; Cole., Tr. 2859, 2878.

29. State monitoring teams have demonstrated their capabilities during eight emergency planning exercises and have received no Category A deficiency reports. Cole, ff. Tr. 2835 at 5.

30. Emergency plans of the three counties in the EPZ provide that they will rely on the State's field monitoring capabilities. Id. at 5-6.

31. If necessary, the State, prior to the arrival of the monitoring teams, can make protective action recommendations based on data from the Applicants' monitors. Cole, Tr. 2888-91.

32. Both the Applicants and Lake County have monitoring teams for dispatch in a radiological emergency with an Applicant capability to

deploy teams within 30-45 minutes of an alert declaration. The U.S. Department of Energy has a capability to provide offshore monitoring by helicopter on Lake Erie. The State, NRC and the Applicants also have a total of 77 thermoluminescent dosimeters (TLDs) throughout the EPZ. Bowers, ff. Tr. 2914 at 4-6; Cole, Tr. 2901.

33. In addition to the State and Lake County, the U.S. Department of Energy, U.S. Environmental Protection Agency and the NRC all have a capability for radiological monitoring. A radiological monitoring and assessment center will be set up by the DOE to relay coordinated information and data. Cole, ff. Tr. 2835 at 5.

34. The State has the primary responsibility for independent offsite monitoring. Shapiro, ff. Tr. 3111 at 4.

35. Planning standard (L) of NUREG-0654 calls for "arrangements" to be made for medical services for contaminated injured individuals and three criteria--L1, L3, L4--apply to local and/or State governments. Shapiro, ff. Tr. 3111 at 6-7.

36. Four county hospitals referred to in emergency plans near the Perry facility (Lake County Memorial East, Lake County Memorial West, Geauga Community Hospital, and the Ashtabula County Medical Center) have facilities and trained personnel to handle contaminated injured individuals. Letters of agreement to carry out emergency plan duties are being obtained. Linneman, ff. Tr. 2980 at 4-6; Shapiro, ff. Tr. 3111 at 6-7 and Attachment 6 at 5, 12, 15.

37. Additionally, there are twenty-six (26) hospitals in counties around the EPZ which are accredited for providing emergency handling of

contaminated individuals and are capable of dealing with radiation victims. All hospitals have written disaster plans which provide for emergency patient overflow from the hospital to be handled by other hospitals. The hospitals with capabilities to handle radiation victims are listed in State and local plans. Linneman, Tr. 2998-99, 3037, 3041; Shapiro, ff. Tr. 3111 at 6-7.

38. Approximately eighty-five personnel in the four country hospitals have received radiological training for treating injured persons. Linneman, ff. Tr. 2980 at 5; Linneman, Tr. 3010-13, 3023-25.

39. All four county hospitals have equipment for detecting high and low radiation levels. Linneman, Tr. 3025.

40. The Ashtabula General Hospital has a radiation emergency area designated with radiation measurement and contamination equipment available. The hospital is accredited not only under the State's Joint Commission on Accreditation of Hospitals but it also has an NRC nuclear license which requires the facility to demonstrate a capability to handle contaminated injured persons. Linneman, Tr. 2982-83, 3034-36.

41. Contaminated persons with injuries seldom require hospitalization and contaminated persons without injuries would be decontaminated outside of hospitals. Linneman, Tr. 2995, 2998, 3033.

42. Medical facilities within a ten to twenty-five mile radius of the Perry facility are adequate to handle even the worst-case assumptions of consequences of a nuclear accident at Perry. Linneman, Tr. 3003.

43. The best equipment or procedure to diagnose radiation overexposure is for the obtaining of complete blood counts and platelet counts. This is available at the four county hospitals. Linneman, Tr. 3029-31.

44. Local plans are responsible for transportation of radiological victims to medical facilities. The State's emergency plan provides for sixty-five ambulances to be furnished by the Ohio National Guard, if necessary, for additional transportation needs. Shapiro, ff. Tr. 3111, at 7.

45. A siting study by the Sandia National Laboratory (NUREG/CR-2239) was inappropriate to cite for a reactor accident at a facility like Perry. The study is based on extremely conservative assumptions. Hankins, ff. Tr. 3158 at 2-7, Tr. 3172-76.

46. Severe accident evaluations of the BWR/6-Mark III, comparable to the PNPP specific design, show releases orders of magnitude below those assumed in the Sandia study. Those design specific releases cause no early fatalities and do not require emergency care. Hankins, ff. Tr. 3158 at 4-7. A GE study, utilizing Perry's specific site characteristics, shows that a core-melt accident produces low doses (less than 25 rem) as close as one mile to the plant. Id. at 9.

47. Letters of agreement from school districts for the use of school buses during emergency evacuation activities have not, as yet, been obtained. Baer, Tr. 3049.

48. Since school buses are owned and controlled by local school districts, it is necessary to obtain school officials' cooperation for the use of buses during emergencies. Shapiro, ff. Tr. 3111 at 9.

49. All three counties are in the process of obtaining the required letters of agreement and FEMA will review them for compliance with NUREG-0654. Id. at 9; Baer, Tr. 3049.

50. There are twenty-four (24) school districts from which letters of agreement are being solicited. Baer, Tr. 3050.

51. It is anticipated that letters of agreement concerning school bus use will be obtained before fuel load. Baer, ff. Tr. 2047 at 2.

52. The letters of agreement are to follow the installation of school bus radios which are being provided by the Applicants. Most school districts appear receptive to issuing the letters. Baer, Tr. 3050-52.

53. The emergency plans of Lake, Ashtabula and Geauga Counties provide for monitoring, decontamination and evaluation of vehicles and property at reception centers. Standard operation procedures are maintained as supporting documents by the Disaster Services Agencies of the three counties and provide guidance to fire departments. The fire departments are responsible for monitoring and decontamination of property at reception centers. Baer, ff. Tr. 3055 at 1-2.

54. More than twice the number of fire department personnel needed for monitoring and decontamination activities at reception centers are in the process of being trained for this purpose and the training course

has specific instructions for handling contaminated property and vehicles. Id. at 2-3.

55. Emergency kits containing equipment and supplies for monitoring, decontamination and handling of property and vehicles are being assembled and will be in place at each reception center prior to fuel load. Id. at 3; Baer, Tr. 3056.

56. The Ohio Environmental Protection Agency will have responsibility for disposal of contaminated property even though County plans mistakenly indicate the Applicants will provide this service. Baer, Tr. 3057; Shapiro, Tr. 3130.

57. The Ohio Disaster Services Agency Radiological Training Manual does not require fire and police department or individually owned vehicles to be decontaminated at places other than reception centers. Wills, Tr. 3202-04.

58. Decontamination procedures for vehicles call for use of open fields adjacent to reception centers. If the fields are subsequently found contaminated, there are decontamination methods available, including spraying with water, to dilute the concentrations accumulated. Baer, ff. Tr. 3055 at 3; Tr. 3065-66.

59. Commission regulations at 10 CFR § 50.47(b)(11) require that means be available for controlling radiation exposures to emergency workers. NUREG-0654, Criteria K.3., provides the guidance of recommending dosimeters for emergency workers. There is no regulatory requirement for bus drivers to be provided with protective gear such as

respirators and goggles. Baer, ff. Tr. 3069 at 2; Shapiro, ff. Tr. 3111 at 11.

60. County emergency plans and school district procedures provide for the distribution and use of dosimeters for bus drivers. These dosimeters will be worn at all times and the drivers will be trained in their use. Standard operating procedures require bus drivers to report to decontamination stations if radiation exposure is indicated. Baer, ff. Tr. 3069 at 2-3; Shapiro, ff. Tr. 3111 at 11-12.

61. Bus drivers are unlikely to be exposed to radioactivity since their duties in an emergency call for their responsibilities to be carried out and completed prior to any significant releases; all buses will have radios for receipt of information from radiological monitoring teams. Baer, ff. Tr. 3069 at 3-4; Baer, Tr. 3073-74, 3080-82.

62. The Ohio Department of Health requires that respiratory equipment be provided to all emergency workers under certain airborne release deposit conditions. However, that requirement is in the process of being revised to eliminate any necessity for such equipment for bus drivers. Shapiro, ff. Tr. 3111 at 11; Wills, Tr. 3207.

63. More than half of the deficiencies reported in the FEMA Interim Report (March 1, 1984) were noted as corrected in the report itself and the remaining have since been corrected or are in the process of being corrected. Baer, ff. Tr. 3088 at 2.

64. The remaining deficiencies in the plans concern an emergency information handbook and this handbook has been reviewed by NRC and found adequate. Id. at 3 and Tr. 3097-98; Perrotti, Tr. 3145-46.

65. All local plans have been revised to reflect corrective actions of the deficiencies noted in the Interim Report. Baer, ff. Tr. 3088 at 2 and Attachment A.

66. A full exercise in late 1984 was held and no Category A deficiencies were noted for any State or local response organization. Shapiro, ff. Tr. 3111 at 13.

67. The thirty-five deficiency items referenced by the Staff in SSER-4 have been resolved in subsequent revisions of the PNPP emergency plan. Hulbert, ff. Tr. 3091 at 2.

68. The Staff is reviewing the Applicants' commitments in the review it is undertaking of the PNPP Emergency Plan, Revision 4, dated February 1985. Perrotti, ff. Tr. 3111 at 5.

B. HYDROGEN CONTROL (Issue 8)

69. Applicants installed a distributed igniter system in PNPP to control large amounts of hydrogen that could be released in a degraded core accident. Applicants' Ex. 8-1.

70. Applicants prepared a preliminary analysis of the hydrogen control system at Perry and an analysis of its containment strength which have been submitted to the Staff. Applicants' Ex. 8-1, 8-4.

71. Applicants have proposed a scope of the preliminary evaluation and analysis at Perry which was accepted by the Staff. Applicants' Ex. 8-1, 8-2, 8-3.

72. Applicants' preliminary analysis references the hydrogen control system at the Grand Gulf plant which was analyzed and licensed

by the Staff. Grand Gulf is similar in design to Perry. Applicants' Ex. 8-1.

73. Applicants' final analysis was expected to be complete by mid-1986. Applicants, ff. Tr. 3241 at 22. Applicants now expect the final analysis to be complete by the end of 1986 (Applicants' Letter to the Board, June 28, 1985).

74. Hydrogen control at the Perry plant will be achieved using two different systems. The first, a combustible gas control system, is designed to meet the original provisions of 10 CFR § 50.44 for design basis accidents. Applicants, ff. Tr. 3241 at 36-37. The second, a hydrogen ignition system, is designed to control large amounts of hydrogen beyond those covered by design basis accidents. Id. at 35. The combustible gas control system is undisputed in this proceeding. OCRE Proposed Opinion at 14.

75. The Applicants' distributed igniter system is designed to burn an amount of hydrogen equivalent to that generated from a metal-water reaction involving up to 75% of the fuel cladding in the active fuel region. Hydrogen will burn at low concentrations below levels which could potentially threaten containment integrity. Applicants, ff. Tr. 3241 at 29; Karlovitz, Tr. 3258; Notafrancesco I ff. Tr. 3676 at 3.[14]

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[14] Mr. Notafrancesco of the NRC Staff submitted testimony in two parts which we designate I and II.

76. The distributed igniter system consists of 102 thermal glow plug igniters of a type used in diesel engines which are placed throughout the drywell, wetwell and upper containment at Perry. Applicants, ff. Tr. 3241 at 32-34. The igniters are powered from 120 volt AC class 1E power distribution panels and are also capable of being powered by the emergency diesel generators. Id.; Applicants' Ex. 8-1 at 9. The glow plug igniters have been tested and have demonstrated reliable ignition of hydrogen. Applicants, ff. Tr. 3241 at 30-31; Fuls, Karlovitz, Tr. 3639-40.

77. The igniters will achieve a service temperature of 1700 degrees which will cause ignition of hydrogen in a controlled manner at or near its lower combustion limit. Applicants, ff. Tr. 3241 at 32; Notafrancesco I, ff. Tr. 3676 at 3.

78. The 102 igniters are divided into six groups of approximately equal number and 2 power divisions: three groups are in Division 1 and three in Division 2. Each group is powered from a separate distribution power panel and each division from a separate power supply. Applicants, ff. Tr. 3241 at 32; Notafrancesco I, ff. Tr. 3676 at 4.

79. Spaced approximately 30 feet apart with alternating divisional power supplies, a distance of approximately 60 feet separates igniters powered from the same emergency power division. Two igniters, one from each power division, are located in enclosed containment areas that could accumulate hydrogen and the number and arrangement of igniter assemblies is similar to those at the Grand Gulf Nuclear Power Station. Applicants, ff. Tr. 3241 at 33-34; Buzzelli, Richardson, Tr. 3642-44.

80. Locations of the igniters have been changed as planning has progressed. OCRE's Ex. 16 showed a draft description of the PNPP igniter locations which was superseded by location descriptions in the preliminary evaluation report (Applicants' Ex. 8-1). The changes were made to meet spacing criteria and to insure availability of support structures in the containment. Buzzelli, Tr. 3503-05, 3607-08, 3640-42.

81. Applicants' combustion consultants have reviewed the PNPP igniter system and concluded that it will be able to safely and effectively burn large amounts of hydrogen. Applicants, ff. Tr. 3241 at 30-31. Neither the spray shield affixed to the igniter assembly nor the placement of assemblies underneath ceilings or near walls will inhibit hydrogen burning. Lewis, Tr. 3513-14. Hydrogen will be ignited at approximately 8% concentration in air by the igniters and burning of hydrogen would be repeated as succeeding flammable mixtures are formed during an accident. This result is substantiated by both experimental and theoretical data. Applicants, ff. Tr. 3241 at 31.

82. The igniter system is designed for manual initiation when the reactor water level drops to the top of the active fuel. Applicants' Ex. 8-1 at 12; Applicants, ff. Tr. 3241 at 34; Buzzelli, Tr. 3424.

83. The distributed ignition system is designed to cover all of the area in the containment. When a particular area exceeds 6 to 8% of hydrogen, ignition will occur and hydrogen thus will not accumulate at higher concentrations. Hydrogen burns will take place at different places in the containment. Garg, Tr. 3749-50; Notafrancesco, Tr. 3749.

84. The igniters used at Perry have been tested under experimental conditions where only one igniter was used. There are 102 igniters in containment to account for variation in hydrogen concentrations at different locations. Karlovitz, Tr. 3639-40. An igniter is placed everywhere in containment that hydrogen can go and specific locations of igniters are not therefore significant to the overall analysis. Richardson, Tr. 3643.

85. PNPP emergency instructions for operation of the igniter system and the generic emergency procedure guidelines on which they will be based are currently under development and will be in place prior to exceeding 5% power. Buzzelli, Tr. 3425-27.

86. The Applicants have prepared an analysis of the PNPP containment ultimate structural capacity and an evaluation of the pressure capability of the PNPP drywell. Applicants, ff. Tr. 3241 at 24-25; Applicants' Ex. 8-1, Sections 3.1, 3.2 and 5.3; Applicants' Ex. 8-4.

87. The Ultimate Capacity Report contains analyses of the internal pressure capacity of the containment using ASME service level C and D stress limits as well as mean and lower bound yield values using actual material strengths. Applicants' Ex. 8-4; Alley, Tr. 3253-54, 3283-85, 3583-85.

88. The ultimate pressure capacity of the Perry containment steel shell was analyzed in accordance with the requirements of the ASME boiler and pressure vessel code Section III Division I subarticle NE-3220 service level C limits. In performing this analysis, a

combination of dead load and internal pressure load of 45 psig was used as required by the NRC. Yang, ff. Tr. 3676 at 2. The pressure capacity of the steel shell knuckle region is 78 psig and the capacity of the limiting shell penetration is 50 psig. The design pressure for the Perry containment is 15 psig and the design meets NRC requirements. The pressure capacity of the limiting shell penetration of 50 psig has a factor of 3 margin with respect to the containment design pressure. This demonstrates a conservative Perry design. Id.

89. The Staff concluded that the analytical results demonstrate the adequacy of the pressure capacity of the PNPP containment steel shell and has accepted Applicants' analysis. Id. at 2; Staff Ex. 8 at 6-5; Applicants, ff. Tr. 3241 at 24; Alley, Tr. 3588-89.

90. The Ultimate Capacity Report established that the PNPP containment vessel and all key components meet the service level C requirements of the ASME code considering pressure and dead load alone. Applicants, ff. Tr. 3241 at 26; Alley, Tr. 3590; Applicants' Ex. 8-4 at 1-2; Yang, ff. Tr. 3676 at 2.

91. The shell capacity in the cylindrical shell region is 79 psi, which is well above the 50 psi minimum for the controlling penetration, and is only marginally affected by slight increases in stress caused by the as-built condition. The pressure capacity of the dome is about 78 psi and cylindrical region about 79 psi. Alley, Tr. 3596-97.

92. Use of service level C limits to define the PNPP pressure capability represents a conservative approach to assuring that containment integrity will be maintained. The ASME code permits higher

service level D limits to be used where the primary intent is to assure that violation of the pressure retaining boundary will not occur. The code states that service level D limits are appropriate for extremely low probability postulated events. A degraded core accident addressed in the hydrogen rule would fall into this category. Applicants, ff. Tr. 3241 at 27; Staff Ex. 8 at 6-5. (Staff Ex. 8, SSER-6, is not bound in the record.)

93. The service limits of the ASME code are directly applicable to stress ranges that are within the elastic limits of the containment materials. The programs used by CEI used stresses within the elastic limits and do not consider inelastic nonlinear behavior of the structure. Codes for analyzing nonlinear inelastic behavior are limited in applicability, but these have not been used to predict ultimate capacity to failure of the PNPP containment structure. Alley, Tr. 3393.

94. The Applicants rely on ASME service level C limits to establish the pressure retaining capacity of containment. The controlling lower bound pressure capacity for the PNPP containment is 50 psig for penetration 414. Applicants, ff. Tr. 3241 at 24, 28; Alley, Tr. 3585-86; Applicants' Ex. 8-4 at 16-17, 20-23, Table 10.

95. The ASME code provides for approximately a 10% reduction in stress allowables due to increased temperature in the range expected from hydrogen burns; this reduction also applies to the limiting containment penetration, P414. In the analysis performed of penetration P414, minimum specified material strength was used rather than actual material strength. The capacity would be 30% higher using actual

material strength. Accordingly, the 10% reduction due to temperature effects is not significant and the analysis remains conservative. Alley, Tr. 3286, 3586-87.

96. The actual pressure capacity of the containment is over three times greater than the design level of 15 psig. Applicants, ff. Tr. 3241 at 26; Yang, ff. Tr. 3676 at 2; Staff Ex. 8 at 6-5.

97. The PNPP containment design is adequate to handle negative pressures following the combustion of hydrogen since PNPP has redundant vacuum breakers which would alleviate such pressures. Applicants, ff. Tr. 3241 at 25; Applicants' Ex. 8-1 at 13-15.

98. The ultimate capacity of the PNPP containment structure for the limiting penetration is significantly above the pressures predicted for a hydrogen event at PNPP. Id. at 24. Applicants, ff. Tr. 3241 at 28.

99. The PNPP containment vessels contain welds inaccessible for repair that deviated from ASME code requirements. OCRE Ex. 13 at 1-1 to 1-3; Staff Ex. 6 at 3-1, 3-2. The location of the inaccessible weld flaws is in the lower weld courses of the containment vessel for both Units 1 and 2. The Applicants' request for Staff acceptance of the flawed welds without repair or reradiography was supported by a technical report commissioned by the Applicants from APTECH Engineering Services, Inc. OCRE Ex. 13.

100. The APTECH report presents a fatigue and fracture mechanics analysis to predict the possibility of fracture based on weld flaw sizes, material properties, and operating conditions of the welds. The assessment depends in part on weld radiographs, some of which were of

poor quality. Computer enhancement of the radiograph was used to aid interpretation. However, APTECH has demonstrated through calibration measurements that their method for flaw depth measurement, while not always accurate, is conservative (i.e., projects deeper flaws than actually exist). OCRE Ex. 13, Appendix B at 9 and Table 2. APTECH concludes that sufficient data from radiographs exist to characterize the maximum extent of a defect that could remain in the structure and has conservatively analyzed the maximum potential defects. OCRE Ex. 13 at ii.

101. On the basis of its review of the APTECH Report, the Staff found that the analysis and techniques performed to assess the effects of the flawed welds demonstrate that general design criteria (GDC) 51, Fracture Prevention of Containment Pressure Boundary, would be met without repairing the flaws in the inaccessible welds of the containment shell. The analyses show that the flaws will have virtually no growth under the operating loads for which the shell was designed and that the steel materials used in the containment pressure boundary have adequate toughness so that a large through-thickness flaw would not cause a rapidly propagating fracture. Staff Ex. 6 at 3-2, 3-3.

102. The Staff accepted Applicants' proposal to leave the weld flaws in the containment shell without repair. The containment shell would not be strengthened significantly by repairing the welds which are a small percentage of the wall thickness, and that there could be added risks in making weld repairs because of the distortion induced and high restraint of the joint figurations. Id. at 3-3.

103. The APTECH report did not specifically consider fatigue crack growth in the containment shell due to multiple hydrogen deflagrations. Alley, Tr. 3306, 3325. Applicants have confirmed through additional analysis that stresses in the shell due to hydrogen burns would be less severe than those used in the APTECH analysis. Alley, Tr. 3313. Pressures from hydrogen burns are expected to be of such short duration that they would not be of significant concern to fatigue crack growth. Alley, Tr. 3325, 3592-93; Buzzelli, Tr. 3326; Wilcox, Tr. 3755. The APTECH conclusions are not affected by consideration of elevated temperatures expected from hydrogen burning. Alley, Tr. 3590-91.

104. The APTECH Report used the criterion of fracture initiation in containment as the definition of failure in its fracture analysis because the materials used in containment vessels have lower resistance to a propagating fracture than they do to fracture initiation. This criterion is therefore conservative. OCRE Ex. 13 at 2-3.

105. When the materials of the Perry containment were welded, a heat affected zone, which is not reduced in strength due to welding, occurred in the base metal. Wilcox, Tr. 3753-54. Repeated repair of a weld on the materials used in the containment vessel does not cause embrittlement of the base metal and will not lessen the strength of the weld or the areas around the weld. Wilcox, Tr. 3754-55.

106. Brittle fracture is not a concern for the carbon steel used in the Perry containment because it is a ductile material and has excellent toughness properties even at room temperature. Wilcox, Tr. 3757.

107. Applicants have analyzed possible leakage in containment through the upper personnel air lock and the equipment hatch as well as through other containment penetrations. The personnel air lock, equipment hatch and penetrations meet the prescribed service level C limits of the ASME code. The analyses showed that deflections at the cover flange and barrel flange of the equipment hatch would occur at 45 psig internal pressures. The magnitude of the deflections were calculated and were found to be less than the O-ring compression on those flanges. Therefore, springback of the O-rings is sufficient to prevent leakage. The maximum permissible internal containment pressure to meet level C stress limits is 50.2 psig for the personnel air lock and 52.6 psig for the equipment hatch. Applicants' Ex. 8-4 at 14-15.

108. The integrity of O-ring seals is not a safety problem at temperatures up to 300°. Compression set is not likely for the temperature range and durations of temperatures that are likely to occur in a hydrogen burn event considering the specific materials from which the O-ring seals are fabricated. Alley, Tr. 3581-82. Maintenance of seals will be performed. Alley, Tr. 3278.

109. The smoothness on mating surfaces of equipment hatch flanges is 80 micro-inches, which is more than adequate to facilitate leak tightness of the seals. Alley, Tr. 3583.

110. One penetration, numbered 205, does not satisfy the service level C limits of the ASME code when minimum specified material strengths are used to perform the analysis. However, when the analysis

is based on actual material certification data, the stress is smaller than the code allowable stress. Applicants' Ex. 8-4 at 16.

111. All lower containment penetration analyses showed that actual stress intensities were less than allowables without consideration of the annulus concrete that fills the space between the steel containment and the outer concrete shield building. This is a conservative analysis since the stiffness of the concrete would prevent the steel containment vessel and penetration area from being stressed to as great a value as used in the analysis. Applicants' Ex. 8-4 at 18.

112. The PNPP personnel air locks use inflatable seals to prevent leakage. Alley, Tr. 3362. Evaluation of inflatable air lock seals at Perry found them to be qualified for use at Perry during accident conditions at anticipated drywell temperatures. OCRE Ex. 14. An equipment qualification program which includes qualification of the drywell personnel air lock and the containment personnel air lock seals is in progress. Buzzelli, Tr. 3375.

113. Seals used in the Perry drywell equipment hatch and lower personnel air lock hatches would be able to survive expected temperatures from diffusion flame burning. Richardson, Tr. 3623-24.

114. The Applicants demonstrated the pressure capacity of the containment drywell structure by referencing a similar structure at the Grand Gulf Nuclear Station. The Grand Gulf drywell structure had a positive pressure capacity of 67 psig and a negative pressure capacity of 89 psig. The Staff accepted the Applicants referencing of the Grand Gulf results as appropriate for demonstrating that substantial margin

and capability above required capacities are expected for the Perry drywell structure. Yang, ff. Tr. 3676 at 3; Staff Ex. 8 at 6-5.

115. Voids found in the concrete wall of the Perry drywell consisting of small gaps behind a quarter inch liner have no effect on the drywell structural capacity. A few larger voids were found and subjected to a detailed evaluation. All voids have been repaired. An inspection program was instituted to make sure there were no other voids. Alley, Tr. 3415-17.

116. The purpose of containment response analysis is to predict the thermal and pressure environments in containment as a result of burning large amounts of hydrogen. The analysis depends on postulated accident scenarios and computer models to predict hydrogen releases, combustion, temperatures, pressures and effects on equipment survivability. Applicants, ff. Tr. 3241 at 37.

117. Two postulated accident scenarios for containment response analysis were selected on the basis that they represent probable events initiated by plant transients and they are scenarios which dominate plant risk. Id. at 37-39.

118. The Staff considers the scenarios used by the Applicants for their preliminary analysis to be adequate. Notafrancesco II ff. Tr. 3676 at 2-3.

119. A computer program termed MARCH used in the analysis was developed by Battelle Columbus for the NRC and models the release of hydrogen with steam from whatever openings in the primary system may be appropriate to the scenario. Applicants, ff. Tr. 3241 at 40.

120. In their preliminary analysis, the Applicants adopted the hydrogen and the steam release rates calculated for the Grand Gulf plant. Because Perry has fewer fuel bundles than Grand Gulf, the releases would be bounded by the rates calculated for Grand Gulf. The Staff considers the MARCH 1.1 code, which was used to calculate the hydrogen release rates, appropriate for the preliminary analysis. Notafrancesco II, ff. Tr. 3676 at 3-4.

121. The Staff is aware of MARCH code shortcomings and newer versions of MARCH have incorporated corrections which show substantially less hydrogen production. Id. at 4-5.

122. Containment response to hydrogen combustion was analyzed using the computer program termed CLASIX-3. The code takes into account Mark III containment features including the suppression pool, refueling pool, vacuum breakers, and drywell purge system. It computes temperatures, pressures and the distribution of atmospheric components in containment which are oxygen, nitrogen, hydrogen, and steam. Applicants, ff. Tr. 3241 at 41, 48.

123. The CLASIX-3 model used for PNPP analysis is identical to that used for the preliminary Grand Gulf analysis, but input parameters were modified as necessary to account for differences in design features or design values. Hydrogen and steam release calculated by the MARCH code are provided as input to the CLASIX code. Id. at 42-44.

124. Additional input parameters needed to model ignition and combustion of hydrogen in CLASIX-3 are based on experimental data and

plant design values and also come from engineering judgment or handbook values. Id. at 44-45.

125. Theoretical and experimental data show that hydrogen, given an ignition source, will burn in a propagating manner at concentrations in air above 8%. The pressures calculated from hydrogen burn are conservative because experimental results show that theoretical pressures are not realized for burns of hydrogen below about 12% concentration. Id. at 45.

126. The hydrogen concentration range over which burning can occur is bounded by the limits of flammability. The lower deflagration limit of hydrogen in air is 4% hydrogen by volume for upward propagating flame, about 6-1/2% hydrogen for horizontal propagation and about 8% for downward propagation. The CLASIX-3 analysis conservatively assumes ignition and propagation of hydrogen at 8% concentration. Id. at 46.

127. Detonation of hydrogen in the Perry containment by acceleration from deflagration cannot occur with thermal igniters. Lewis, Tr. 3617.

128. The CLASIX-3 analysis shows that frequent periodic deflagrations occur in the wetwell for both scenarios analyzed. Applicants, ff. Tr. 3241 at 47.

129. Most temperature excursions peak at around 800°F and pressure excursions peak at around 6 pounds per square inch (psig). Id. at 47-48.

130. Results of CLASIX-3 analysis show that a peak temperature of 1700°F is attained during a hydrogen release event in the wetwell under the scenario of the stuck-open relief valve. Peak pressure of slightly

over 21 psig in both the wetwell and the containment also occur under the scenario. Id. at 48.

131. Hydrogen flame will propagate from an igniting source which is the glow plug. If the concentration is in the right range, the flame will move downwards, sideways, and upward and closeness to a wall, ceiling or shield are not impediments. The criterion for downward propagation is that hydrogen concentration be on the order of 8-1/2% to 10%. Lewis, Tr. 3514.

132. A flame speed of 6 feet per second in hydrogen-air mixtures is conservative for use in the CLASIX analysis. Lewis, Tr. 3520-21. Lower flame speeds have been measured, and, at concentrations of 8% hydrogen, speeds of 20 meters per second have occurred due to acceleration of flames by obstacles in experimental devices. Lewis, Tr. 3523.

133. The CLASIX-3 model for containment response is conservative because hydrogen combustion will take place at varying locations in the containment; the postulated hydrogen release rates are overestimated, and the natural upward propagation behavior of hydrogen will result in burning at lower than postulated concentrations. The result is that a piece of equipment in the wetwell would be exposed to fewer burns and the burns would be spaced further apart than those shown in the CLASIX results. The concomitant pressure pulses would likely be less than the containment design pressure as distinguished from the containment pressure capacity. Notafrancesco I, ff. Tr. 3676 at 8.

134. The detonatable concentration range for hydrogen ranges from a lower limit of about 14% to an upper limit of about 60%. Lewis, Tr. 3523-24.

135. Detonations of hydrogen-air mixtures can be promoted by the presence of certain reactive chemical radicals. However, no evidence exists that such chemical radicals would be present in the Perry containment or that ionizing radiation could produce specific reactive chemical radicals that could accelerate deflagrations into detonations. Lewis, Tr. 3526-28.

136. Applicants' preliminary analysis of the survivability of the essential equipment exposed to the thermal environment postulated in containment during a hydrogen burn demonstrated a similarity between equipment in the Grand Gulf and Perry plants. The analysis determined the thermal response of a selected piece of essential equipment exposed to a hydrogen burn in containment. The maximum temperature reached by equipment in Perry was found to be lower than the corresponding temperature calculated for Grand Gulf and the analysis demonstrates on a preliminary basis that essential equipment in Perry will survive a hydrogen burn. Garg, ff. Tr. 3676 at 2-4; Applicants, ff. Tr. 3241 at 49-51. A preliminary identification and evaluation has been performed at Perry on equipment required to survive a hydrogen burn based on its function during and after a postulated degraded core accident. _Id._ at 50.

137. The thermal response analysis used the HEATING computer code in comparing Perry equipment with that in Grand Gulf. Garg, ff. Tr. 3676 at 5.

138. The Applicants' conclusion, with which Staff agrees, is that the thermal response of all safety-related equipment in Perry will be less severe because of lower temperatures at Perry than at Grand Gulf. Id. at 5. However, two items, containment locks and hatch seals and a transformer, have not been qualified. Id. at 6.

139. The Staff concluded there is reasonable assurance that essential equipment will survive the pressures generated during a hydrogen burn. Staff testimony did not explicitly demonstrate that the exceptions listed in Applicants' preliminary analysis had been considered. Id. at 6; Applicants' Ex. 8-1 at 21D.

140. The Applicants' analysis of equipment pressure survivability concluded that qualification or design pressures bound the calculated peak pressures from hydrogen combustion in all cases except the containment vacuum breaker, the hydrogen mixing compressors and the discharge check valves. Applicants' Ex. 8-1 at 21D.

141. Active components of the vacuum breakers and check valves were accepted in the preliminary analysis because they will not be exposed to peak external pressures and are expected to function during hydrogen burning. Hydrogen mixing compressors of identical design at Grand Gulf were shown to survive pressures of 24 psig which bounds the peak calculated containment pressure of 21.2 psig at Perry. Id.; Staff Ex. 8 at 6-11.

142. The Board finds that the Staff and Applicants have given inadequate explanation for their acceptance of the exceptions of equipment pressure survivability noted by Applicant. Id.; Buzzelli, Tr. 3570-71.

143. The Staff accepted the Applicants' analysis of equipment operability during a hydrogen event for four reasons. (1) The list of equipment identified in the Applicants' submittals are similar to the list of equipment found acceptable by the NRC during the Grand Gulf licensing review; (2) the comparative analysis of thermal response of similar equipment in the Perry and Grand Gulf plants shows that equipment temperatures in Perry will be lower; (3) the qualification temperatures for Perry equipment are higher than the corresponding qualification temperatures for Grand Gulf equipment; and (4) the pressures developed during hydrogen burns are smaller than the design or qualification pressures. The Staff concludes that it has been satisfactorily demonstrated from the preliminary analysis that essential equipment can reasonably be expected to survive such pressures. Garg, ff. Tr. 3676 at 7.

144. In a station blackout that had progressed to a degraded core accident which was generating hydrogen, there would be no power to the igniters. Buzzelli, Tr. 3428, 3432, 3438.

145. In a station blackout accompanied by 75% metal-water reaction, hydrogen could accumulate to a concentration of around 28% in the containment. Buzzelli, Tr. 3438. Under those conditions, hydrogen could ignite if electric power is restored to the plant but it would not

detonate. Lewis, Tr. 3440. High pressures are produced by deflagrations in the station blackout scenario and 100 to 110 psi could be attained. Id. In the event of station blackout, containment venting could be utilized to dissipate hydrogen rather than permitting large deflagrations. Buzzelli, Tr. 3441-42. The vent path that might be used in such an accident at Perry has not been established. However, it is under review and evaluation by plant engineering staff and consultants. Richardson, Buzzelli, Tr. 3443.

146. The probability of a need for venting the containment during a hydrogen generation event in a station blackout situation would be low since in a station blackout, the reactor core isolation cooling system would still be operable and would maintain coolant makeup. The system has the capability to maintain core makeup in a station blackout for at least 9 hours. As long as coolant flow is occurring or is maintained, there would be no hydrogen generation. Richardson, Tr. 3609.

147. No burning of cable insulation has been observed for hydrogen concentrations less than 10% in tests at the Nevada test site. Garg, Tr. 3748. Burning has been observed at hydrogen concentrations of 10% and above. At 13% the burning was for a long time while at 10% it was for a short time. The burning of the cable did not affect its function in tests. Id.

148. Applicants have considered the potential for combustible material in the containment or drywell to be ignited by hydrogen burning. It was concluded in the Grand Gulf analysis and the Perry analysis that there is no potential for secondary fires. Richardson,

Tr. 3580-81. However, the thermal environment from diffusion flames has not yet been defined. The capability of equipment to survive will be evaluated after the thermal environment of diffusion flames is defined. Id.

149. The Hydrogen Control Owners Group program intends to conduct large scale tests on the thermal environment from diffusion flames. Richardson, Tr. 3552. Results of past 1/20th scale tests are inconclusive with regard to their applicability to a full scale containment like Perry because the scaling relationships break down yielding conservative temperatures. Richardson, Tr. 3557.

150. Equipment and hatch seals will be evaluated in future quarter-scale testing for thermal environments which could exist from diffusion flames. Richardson, Tr. 3558-59.

151. Applicants' experts on hydrogen combustion agree that the testing of diffusion flames above the suppression pool should be conducted. The test does not have to be full scale, however, since one quarter scale tests would be accurate. Karlovitz, Tr. 3560-65; OCRE Ex. 17.

152. Testing of diffusion flames will utilize hydrogen release rates which are consistent with the release rate from a degraded core which could reach 75% metal water reaction. The hydrogen release history will not attempt to be equivalent to a 75% metal-water reaction because the metal oxidation rate would be so high that it would lead to a very rapid and high degree of core melt: this is beyond the scope of degraded core accidents and beyond the scope of the tests. Richardson, Tr. 3568.

153. Applicants have considered in its design basis analysis the possibility of violent overthrow of the suppression pool into the drywell. Differential pressures between wetwell and drywell generated during hydrogen combustion are less than those considered for the design basis. Applicants reference the Grand Gulf analysis to show that such an event would not have adverse effects on essential equipment. Richardson, Tr. 3485-96.

154. The loss of low-pressure coolant injection has been postulated for the purpose of creating accident scenarios and specific means of how this would happen have not been established. The loss of containment spray cannot be inferred from a scenario which postulates that core cooling would not be available. Richardson, Tr. 3445.

155. Heat removal from containment during a hydrogen release event can be accomplished through the residual heat removal system (RHR). This redundant system is designed to meet long term decay heat removal requirements and all active components are located outside of containment. Richardson, Tr. 3453-54, 3611-13.

156. Mr. John M. Humphrey, a former General Electric containment systems engineer, identified a number of concerns about Mark III containments. Richardson, Tr. 3478-79. Utilities having Mark III containments have considered the so-called Humphrey concerns and have concluded, along with NRC and ACRS, that all of these concerns are second or third order effects and not significant from a safety standpoint. Richardson, Tr. 3481-84, 3613-14.

157. The Staff concludes that the Applicants' preliminary analysis as required by the hydrogen rule is acceptable and that CEI is in compliance with the hydrogen rule. Notafrancesco II, ff. Tr. 3676 at 6; Staff Ex. 8 at 6-1 to 6-14.

158. The Staff will require additional work for the final analysis. The Applicant must consider ATWS and station blackout as initiating events, or provide suitable justification for their exclusion. Code deficiencies must be addressed in an acceptable manner and user input parameters to the code must be treated in a sensitivity analysis. Notafrancesco II, ff. Tr. 3676 at 6. The Hydrogen Control Owners Group has been made aware of the Staff concerns. Id.

C. DIESEL GENERATORS
(Issue 16)

159. Perry Nuclear Power Plant has installed four Transamerica Delaval (TDI) diesel generators to provide emergency power. The generators are DSRV-16-4 models, with 16 cylinders arranged in two banks in a V-type block. Kammeyer, ff. Tr. 2179 at 2. Each engine-generator set is rated for continuous operation at 7000 kw with a short-term overload rating of 7700 kw. Staff Ex. 5 at 2.1.

160. To resolve major problems in TDI diesel generators, an Owners Group was formed by 13 TDI-using utilities in December 1983. A program plan was submitted to the NRC Staff in March 1984, and was approved by the Staff. The plan incorporated design review, quality revalidation, engine tests and component inspection to resolve concerns about the reliability of the TDIs and was made a condition for licensing for all

TDI owners. Kammeyer, ff. Tr. 2179 at 7-10; Staff, ff. Tr. 2281 at 14; Staff Ex. 1 at 1-2, 6-7.

161. The Owners Group plan consists of four parts: a resolution of known generic problems (Phase I), a systematic design review and/or quality revalidation (DR/QR) of all components important to engine reliability and operability (Phase II), engine inspection and testing in accordance with Phase I and Phase II results, and appropriate maintenance and surveillance programs. Kammeyer, ff. Tr. 2179 at 10-11, Tr. 2181-83.

162. The Owners Group plan did not rely on TDI's quality assurance program, but completely revalidated the diesels onsite. Kammeyer, Tr. 2238, 2240. TDI evaluations and recommendations were considered, but were not relied on. Staff, ff. Tr. 2281 at 16.

163. In Phase I, an "Emergency Diesel Generator Component Tracking System" was developed which collected data on failures experienced in TDI diesels, both in nuclear and in other service. Sixteen components showed potential generic problems and were subjected to detailed design review. Technical problems were identified and resolved through analysis, testing and documentation reviews. Kammeyer, ff. Tr. 2179 at 11-14; Staff, ff. Tr. 2281 at 13; Dingee, Berlinger, Tr. 2470.

164. Southwest Research Institute (SwRI) reviewed, evaluated and independently verified the methodology, results and conclusions of the Owners Group studies for PNPP. Christiansen, ff. Tr. 2179 at 7-8. SwRI concluded that the 16 Phase I components in the PNPP engines will perform their intended function. Wood, ff. Tr. 2179 at 86-87.

165. Pacific Northwest Laboratories has reviewed for the Staff the Owners Group report on Phase I components applicable to PNPP. Berlinger, Tr. 2300-01, 2329. PNL considers all the Phase I components are suitable for full-load operation, but will review the torsiograph tests to independently evaluate results received from PNPP. Staff Ex. 5 at 3.1; Hardy, Tr. 2416.

166. Phase II of the program examined the design and quality of those components important to the operability of the engines which were not examined in Phase I. The components were reviewed in a DR/QR program. The function and role of a specific component determined whether a DR/QR or both were performed. For the PNPP engine 171 components were selected for review. Kammeyer, ff. Tr. 2179 at 16-24.

167. The DR/QR program is based on a "lead engine" concept. A DSRV-16-4 engine installed in the Comanche Peak Steam Electric Station was chosen for the lead engine. This engine was subjected to extensive evaluation and review. Each of the 171 Phase II components at PNPP were evaluated to see if the Comanche Peak review was applicable, and the PNPP DR/QR review was applied accordingly. 153 components were found to be the same. Id. at 23; Kammeyer, Tr. 2224, 2492-95.

168. The third phase of the Owners Group program required a complete disassembly of the PNPP engine, field inspection and reassembly. Christiansen, ff. Tr. 2179 at 7, 11, Tr. 2499.

169. The final requirement of the Owners Group plan consists of an ongoing comprehensive set of surveillance and maintenance procedures. PNPP is implementing all Owners Group and regulatory recommendations, as

well as additional surveillance and maintenance recommendations made by PNL and SwRI. Christiansen, ff. Tr. 2179 at 12-13, 18, Tr. 2498. The Staff will review PNPP's implementation of the various recommendations, and believes that proper implementation will guarantee that the engines will meet GDC 17 throughout the life of the plant. Staff, Tr. 2303, 2468.

170. A deficiency was reported to the NRC from Transamerica Delaval, Inc. on March 18, 1985 as required by 10 CFR Part 21. OCRE Ex. 8. The potential defect was the failure of an oil plug in a DSRV-16 crankshaft which could result in engine non-availability. The problem was apparently caused by the use of a plug made with 22 rather than 16 gauge material. PNPP diesels use plugs made of 16 gauge material, but will inspect them in any event. Kammeyer, Tr. 2230-31; Christiansen, Tr. 2262-63.

171. The failure history of TDI connecting rods principally involved use of 1-7/8" bolts. Wood, ff. Tr. 2179 at 82. The PNPP design uses 1-1/2" bolts, which reduces stress levels in the master rod box. Id. There have been no failures with the use of 1-1/2" bolts at the current torque level. Staff, Tr. 2435-37. Preventive maintenance at PNPP will include torque checks. Christiansen, Tr. 2489-90.

172. Connecting rod bearing shell failure has been caused by a large chamfer (1/4" x 45 degree). PNPP connecting rod bearing shells have a smaller chamfer (1/6" x 45 degree) which relieves the problem. Wood, ff. Tr. 2179 at 21-22. The bearings in place at PNPP have been inspected and found to meet Owners Group criteria. Id. at 26;

Henricksen, Tr. 2440. The Staff agrees that the bearings are adequate. Staff, ff. Tr. 2281 at 36.

173. Cracking in the skirt-to-crown stud attachment bosses in AF model pistons at the Shoreham plant has been observed. However, the PNPP pistons are model AE, which have a lower ferrite content and are structurally stronger. The Owners Group and SwRI conclude that the PNPP pistons are capable of unlimited life under full load conditions. Wood, ff. Tr. 2179 at 49-55. The Staff finds them to be satisfactory. Staff, ff. Tr. 2281 at 28.

174. Some instances of fretting and chrome flaking have been observed in the AE pistons. Fretting is not a serious problem. Henricksen, Tr. 2426-27; Staff, ff. Tr. 2281 at 26. Chrome flaking from piston rings and wrist pins has shown no serious damage. Id.

175. The friction-welded pushrods used at PNPP have experienced no failure, although there have been problems with other designs. The Owners Group conducted intensive analysis and testing of pushrods, including non-destructive tests and metallurgical evaluation. The conclusion was reached and concurred in by SwRI that the friction-welded pushrods were satisfactory. Wood, ff. Tr. 2179 at 38-42.

176. PNL has concluded that the PNPP pushrods are adequate. Staff Ex. 5 at 4.24-4.25. PNPP will confirm that Owners Group requirements for random sample testing have been followed. Christiansen, Tr. 2499.

177. Rocker arm capscrews have experienced random failures due to insufficient preload. After intensive evaluation, the Owners Group and SwRI have concluded that the capscrews are adequate. Maintenance will

assure that proper preload is maintained. Wood, ff. Tr. 2179 at 6-11. PNL also found that the capscrews were satisfactory. Staff Ex. 5 at 4.27.

178. Fuel oil injection tubing has shown some problems involving leakage. The Owners Group conducted an extensive analysis of the tubing, including eddy current testing to assure the absence of flaws. Both the Owners Group and SwRI concluded that the tubing in place at PNPP is adequate for continued use. Wood, ff. Tr. 2179 at 11-14. PNL concurred that the tubing is suitable for its use. Staff Ex. 5 at 4.33.

179. TDI identified two potentially defective engine-mounted cables that did not meet IEEE-383-1974 standards. The Owners Group evaluated the PNPP design and concluded that the PNPP wiring is suitable; SwRI agreed with their conclusion. Wood, ff. Tr. 2179 at 27-29. PNL agrees that the PNPP wiring is acceptable. Staff Ex. 5 at 4.36.

180. Airstart capscrews supplied by TDI to one utility were too long for their bolt holes, thus not allowing the airstart valves to seat properly. The Owners Group conducted an analysis and evaluation, and SwRI performed a number of analyses to determine the functional attributes of the capscrews. Wood, ff. Tr. 2179 at 34-36. PNPP inspected every bolt hole depth to determine that the capscrews would not bottom out. Christiansen, Tr. 2223; Persinko, Tr. 2420-21. The Owners Group and SwRI concluded that the capscrews were adequately designed and satisfactory for nuclear service. PNL concurs. Wood, ff. Tr. 2179 at 36; Staff Ex. 5 at 4.35.

181. Isolated failures of cylinder head studs have occurred due to insufficient preload. The Owners Group performed stress analyses of two types of studs: straight-shank and neck-shank (which are used at PNPP). SwRI reviewed the Owners Group work and agreed with the results. They both concluded, and PNL concurs, that either stud design is satisfactory. Wood, ff. Tr. 2179 at 29-33; Staff Ex. 5 at 4.23. The necked design was recommended by SwRI in part because it is less likely to lose preload. Maintenance procedures will assure that preload is maintained. Wood, ff. Tr. 2179 at 33.

182. Although the jacket water pump used on DSRV-16-4 engines does not have a history of failures, it was included in Phase I components because of failures on the DSR-48 engines used at Shoreham. Both the Owners Group and SwRI analyzed the component and concluded that the pump design at PNPP was adequate. Id. at 69-73. PNL concurred in their analysis. Staff Ex. 5 at 4.31.

183. Cracking in TDI cylinder heads has been observed in a number of locations. The Owners Group evaluated the cylinder heads, and a number of recommendations have been implemented by PNPP. The more important recommendations are:

a) The PNPP cyclinder heads have been stress-relieved and welded. This treatment assures that valve seat cracking will not be a problem. Staff, ff. Tr. 2281 at 6; Kammeyer, Tr. 2232-35.

b) A recommendation by PNL was that none of the PNPP heads have through-wall weld repairs on one side only to avoid

stress concentration. Staff Ex. 5 at 4.20. None of the cylinder heads at PNPP have such weld repairs. Berlinger, Tr. 2428-29.

c) All of the cylinders meet minimum fire deck thickness requirements. Kirkwood, Tr. 2431.

d) PNPP engines will be air-rolled prior to any planned startup and after each operation to check for water leakage from the head to the cylinders. Christiansen, Tr. 2489, 2501; Staff, ff. Tr. 2281 at 6.

e) Inspection of the heads will be made regularly. Berlinger, Tr. 2427-28.

The Owners Group, SwRI, the Staff and PNL consider that the design of the cylinder heads at PNPP is adequate. Wood, ff. Tr. 2179 at 48; Berlinger, Henricksen, Tr. 2427-28.

184. Various failures in the engine bases and bearing caps have occurred which include cracking in a DSR-4 model, a nut pocket failure in a DSRV-16-4 model and through-bolt failures on a DSR-46 model. The Owners Group conducted a thorough analysis, reviewed by SwRI, of the saddle and caps, the through-bolts and the bearing cap and fastener system. The nut pocket failure was found to result from impurities in the casting material. The results of the analyses show that there would be no lateral movement of the cap due to the horizontal force of the crankshaft. Wood, ff. Tr. 2179 at 16-20.

185. Only one failure in an engine base due to material abnormalities has been reported for hundreds of diesel engines in

various uses. The Owners Group did not perform metallurgical or material composition tests on the base due to the low loading of the base and its operational history. Kammeyer, Tr. 2216-17, 2504-05. The NRC Staff and PNL agree with this conclusion. Staff, ff. Tr. 2281 at 37-38. Inspection of the most highly-loaded bearing cap and the most highly stressed saddle, with routine maintenance and visual inspection at each outage, is sufficient. PNPP routinely performs maintenance which includes visual inspection of each bearing saddle area. Kammeyer, Tr. 2216, 2239; Christiansen, Tr. 2260.

186. The turbochargers on TDI diesels have experienced failures in the thrust bearing, nozzle vane, nozzle ring capscrew and washer, and nozzle ring. PNPP replaced or refurbished bearings and thrust collars damaged by shop testing, and has implemented prelubrication systems recommended by the Owners Group. Staff, ff. Tr. 2281 at 38-39.

187. Vane cracking initiates below the surface of the vane hub in the vane root and cannot be detected by visual or liquid penetrant testing. Berlinger, Tr. 2353-54. Due to this impediment, vane failures can be expected to occur in the future, and could possibly severely damage the rotor; however, no turbocharger in nuclear service has ever been seriously damaged by vane failure. Staff, ff. Tr. 2281 at 39-40; Henricksen, Tr. 2443-45. The turbocharger will be subjected to stringent maintenance and surveillance to identify problems at an early stage. Berlinger, Dingee, Tr. 2470-72; Christiansen, Tr. 2490.

188. Proper alignment of the turbocharger on its mounting brackets is important as vibration or distortions can be deleterious to proper

operation. However, when misalignment is discovered, it can be properly resolved. Staff, ff. Tr. 2281 at 6, 41-42.

189. While there have been no failures of the 13" x 13" crankshafts used in the DSRV-16-4 engines, the catastrophic failure of a 11" x 13" crankshaft in one of the Shoreham engines, and cracks found in the other two Shoreham crankshafts prompted an extensive Owners Group detailed design review. Wood, ff. Tr. 2179 at 74-81.

190. At San Onofre, cracking was observed in the main journal oil holes of one of the TDI engines. As the San Onofre machine was a DSRV-20 model with a significantly different crankshaft, the problems experienced at San Onofre are irrelevant to the PNPP crankshafts. Berlinger, Tr. 2329, 31; Hardy, Tr. 2409. However, the PNPP oil holes were inspected and eddy current tested to a depth of three inches, as recommended by the Owners Group. Kammeyer, Tr. 2210. The eddy current tests indicated that the journal oil holes were free from defects. Christiansen, ff. Tr. 2179 at 12.

191. The PNPP diesels are not required to meet the European Ship Classification Societies Rules, and the NRC Staff does not recommend that they be applied to land-based units. These requirements are inapplicable to land-based units used in nuclear service. Staff, ff. Tr. 2281 at 25. PNPP crankshafts are required to meet the Diesel Engine Manufacturers Association (DEMA) recommendations. Wood, ff. Tr. 2179 at 75.

192. The Owners Group analysis showed that crankshaft stresses satisfied the DEMA requirements. Torsiograph testing, as required by

the Group, was conducted on both PNPP engines to confirm the Owners Group analysis. Id. at 78-81; Kammeyer, Tr. 2245-46. Tests showed that the PNPP crankshaft has a fourth-order critical speed which is close to the engines' operating speed, but the resulting stresses were well within the DEMA allowable stresses. Kammeyer, Tr. 2245-46. The engines therefore meet PNPP's specifications. Christiansen, Tr. 2196.

193. The Staff's and PNL's interim basis for evaluating the PNPP crankshafts is the torsiograph testing previously conducted by TDI, the fact that the Owners Group calcuations confirmed the previous test results, and results of torsiograph testing of other crankshafts at other plants. Hardy, Tr. 2324-26.

194. PNPP's specification on speed ranges require that there be no deleterious critical speeds within ±10% of operating speed. Christiansen, Tr. 2187. Although the fourth-order critical speed is well within this range, it was found to be within allowable DEMA stress requirements. Kammeyer, Tr. 2245-46. However, PNL recommended, and the Staff concurred, that steady operation of the engines below 450 rpm, the operating speed, be minimized. Staff Ex. 5 at 4.10; Staff, ff. Tr. 2281 at 9. PNPP has set the governor on the engines to limit the speed range to between -1/2% to 6% of 450 rpm during manually controlled operation when the diesels are not attached to the grid. Christiansen, Tr. 2498.

195. The Staff recommended that cylinder imbalance tests be made. Radical imbalance was simulated by cutting off all fuel to one cylinder on a PNPP engine. Christiansen, Tr. 2265; Kammeyer, Tr. 2511. Cylinder imbalance can be determined by monitoring cylinder firing pressures and

exhaust temperatures. Staff, ff. Tr. 2281 at 9. This will be performed at PNPP. Christiansen, Tr. 2497.

196. PNPP conducted a 100% inspection of cylinder block tops and liner landings and no evidence of any cracking was found. Christiansen, Kammeyer, Tr. 2222.

197. Actual electrical loads to be carried by the diesel-generators and their time sequence are set forth in Applicants' Ex. 16-1 and will be verified by testing. Christiansen, Tr. 2215. Additional loads in the future would require an amendment to the FSAR and would be reviewed by the Staff. Christiansen, Tr. 2258.

198. After operation at over 50% of nameplate rating the Owners Group recommends that blocks with ligament cracks should be inspected for stud-to-stud cracks. Any stud-to-stud cracks that extend less than 1.5" from the block top are acceptable. Bush, Tr. 2370, 2372; Wood, ff. Tr. 2179 at 62. Dr. Bush argues that he would limit the cracks to 0.4" to 0.5" in the presence of ligament cracks. Bush, Tr. 2372-74.

199. Cylinder liner proudness has been reduced on the PNPP engines to reduce pressure on the liner and thereby reduce the possibility of block cracking. Henricksen, Persinko, Tr. 2447-48; Christiansen, Tr. 2508. Cylinder lining proudness has been reduced to two mils, which will reduce stress but still maintain liner crush. Persinko, Tr. 2448; Christiansen, Tr. 2508.

200. PNPP's architect-engineer established an inspection requirement of 85% surface contact between the engine base and chocks. If contact was less than 85%, an engineering evaluation was required.

The PNPP engines were inspected, and the requisite engineering evaluation was performed for chock plates with less than 85% contact. In all cases, the contact exceeded TDI's minimum requirement. Hot and cold crankshaft deflection measurements were performed which confirmed that TDI's criteria have been met and that the engine is well-supported. Christiansen, Tr. 2496-97.

III. CONCLUSIONS OF LAW

In reaching a decision herein, the Board has considered all of the evidence submitted by the parties on emergency planning, hydrogen control and diesel generators. Based upon the foregoing Findings of Fact, which are supported by reliable, probative and substantial evidence in the record of this proceeding, the Board, with respect to the issues and controversy before us, reaches the following conclusions pursuant to 10 CFR § 2.760(a):

The Applicants, Cleveland Electric Illuminating Company, et al., have met, subject to the conditions below, their burden of proof on each of the contentions decided in this partial initial decision, and, as to these issues, there is reasonable assurance that the Perry Nuclear Power Plant, Units 1 and 2, can be operated without endangering the health and safety of the public.

In accordance with the Atomic Energy Act of 1954, as amended, and the Commission's regulations, and based on the Findings of Fact and Conclusions of Law, set forth in a Partial Initial Decision on quality assurance previously rendered and in this decision, the Director of Nuclear Reactor Regulation, upon requisite findings with respect to matters not resolved in the Board's Partial Initial Decision, is authorized to issue to the Applicants, licenses for the operation of the Perry Nuclear Power Plant, Units 1 and 2, upon the conditions set forth below.

> Prior to the issuance of the aforementioned licenses, the Applicants shall demonstrate to the Director of Nuclear Reactor Regulation satisfactory completion of the following

1. The Applicants' EAL's conform to the initiating guidance of NUREG-0654, Appendix I.

2. Letters of agreement have been obtained from all school districts for the supply of buses for evacuation purposes.

3. Training of fire personnel in monitoring and decontamination procedures is complete and all reception centers are provided with necessary decontamination equipment.

4. Applicants have committed in writing to comply with the Commission's response to the remand in GUARD v. NRC 753 F2d 1144 (D.C. Cir. 1985). (Board has received notice of August 13, 1985 concerning a commitment on this issue.)

5. The items described in SSER-6 at 9-7, concerning the TDI diesels have been completed.

6. Written procedures for operation of the hydrogen igniter system are available before operation in excess of 5% power.

7. Applicants have made further confirmatory analysis of equipment in the containment that has not been qualified for pressure survivability, or have narrow margins of pressure survivability: this includes containment vacuum breaker, hydrogen mixing compressor and discharge check valves.

IV. ORDER

Pursuant to 10 CFR § 2.760 of the Commission's Rules of Practice, this partial initial decision shall constitute the final decision of the Commission forty-five (45) days from the date of issuance subject to any review pursuant to 10 CFR §§ 2.760, 2.762, 2.764, 2.785 and 2.786, or as the Commission directs otherwise.

A notice of appeal from this partial initial decision may be filed within ten (10) days after its service in accordance with 10 CFR §

2.762. A brief in support of the appeal shall be filed within thirty (30) days thereafter and forty (40) days if 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), any 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 CFR § 2.762.

IT IS SO ORDERED

THE ATOMIC SAFETY AND
LICENSING BOARD

James P. Gleason, Chairman
ADMINISTRATIVE JUDGE

Glenn O. Bright
ADMINISTRATIVE JUDGE

Jerry R. Kline
ADMINISTRATIVE JUDGE

Bethesda, Maryland