UNITED STATES OF AME?ICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of PORTLAND GENERAL ELECTRIC et al. (Trojan Nuclear Plant)	Docket No. 50-344 (Control Building Proceeding)

LICENSEE'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW CONCERNING INTERIM OPERATION

November 20, 1978

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PRELIMINARY STATEMENT

I. INTRODUCTION AND BACKGROUND

1. The present proceedings are being conducted to determine (1) whether an amendment should be issued to Operating License No. NPF-1 for the Trojan Nuclear Plant (the "Plant") waiving certain requirements of that license during an interim period prior to the completion of modifications to the Plant's Control Building and (2) whether the scope and timeliness of proposed modifications to the Control Building are adequate from a safety standpoint. These issues were set forth in the Commission's Order for Modification of License (the "Order of May 26") issued May 26, 1978 (43 Fed. Reg. 23768), which followed the discovery of a design deficiency that led to a determination that the Control Building walls do not fully meet the seismic design criteria of the Trojan Final Safety Analysis Report ("FSAR").

2. Pursuant to the Board's order of August 25, 1978, the hearings to date and the scope of this initial decision are limited to determination of the first issue enumerated above. A determination on the second issue will be made after further hearings.

3. The Plant is located 42 miles northwest of Portland, Oregon on the Columbia River. Design of the Plant's Control Building was initiated in early 1969, and initial design work was completed in 1970. A construction permit for the Plant was issued on February 9, 1971 and drawings for construction of the Control Building were issued commencing in March of 1971. Construction began immediately thereafter; foundation concrete was placed in 1971, the structural steel framing was essentially completed by the end of September 1971, concrete placements for floor and roof slabs were completed in mid-1972 and construction of the major composite shear walls was completed in late 1972. Operating License No. NPF-1 was issued on November 21, 1975. (Broehl, Licensee Exh. 13, pp. 1-2).[1/]

4. In April of 1978, while the Plant was shut down for re-fueling, an investigation by Bechtel Power Corporation ("Bechtel")

1/ Prefiled testimony in this proceeding was accepted into the record in the form of exhibits. Citations to such testimony in this decision will list the name of the witness (underscored), the exhibit number, and the page of the testimony being cited. Thus, the above citation refers to pages 1-2 of the prefiled testimony of Licensee's witness, Donald J. Broehl, which was accepted as Licensee's Exhibit 13. References to transcript pages will be followed by the name (in parentheses) of the witness being cited, unless the text clearly identifies the witness. All of the exhibits accepted into the record are listed in Appendix A of this decision.

of the feasibility of cutting an opening and installing a security window in a wall of the Control Building disclosed a deficiency in the original design. (Anderson, et al., Licensee Exh. 10, pp. 4-5.)

5. After Bechtel reported to Portland General Electric Company ("PGE" or "Licensee") that it had identified a potential nonconformance of the shear walls in the Control Building with the design criteria stated in the FSAR, PGE advised the NRC of the potential nonconformance and the design review actions being taken. Subsequently, PGE and Bechtel discussed the nonconformance with NRC staff personnel in Bethesda, Maryland. The significance of the nonconformance was documented in the Licensee Event Report 78-13 ("LER"), dated May 5, 1978 (Licensee Exh. 6). In response to requests for information generated in the course of NRC review, the Licensee submitted Supplemental Information to LER 78-13 ("Supplement") on May 24, 1978 (Licensee Exh. 7). (Broehl, Licensee Exh. 13, p. 2).

6. The design deficiency is that there is less continuous reinforcing steel in the Control Building walls than would be necessary to satisfy fully the criteria of the FSAR. Two distinct, generally unrelated, design problems gave rise to the deficiency. The first problem is that some reinforcing steel embedded in the concrete core of the Control Building's shear walls is not continuous. This steel should have been either welded to or run through the steel beams and columns or run outside the steel framing. The second problem resulted from the misapplication of two formulae (a code-design formula used

to determine concrete shear capacity, and a mathematical expression used to determine the quantity of reinforcing steel to be embedded in the concrete). Correct application of formulae would have resulted in more reinforcing steel being placed in the walls. (Anderson, et al., Licensee Exh. 10, pp. 5-7.)

7. After reviewing the LER and Supplement, the NRC Office of Nuclear Reactor Regulation issued the Order of May 26, accompanied by a Safety Evaluation Report, which concluded that, despite the deficiency, there was adequate assurance that the Plant could still safely withstand the Safe Shutdown Earthquake ("SSE") in the interim before completion of Control Building modifications which would restore intended seismic design capacity. This conclusion was subject to two conditions:

a. No modification which may in any way reduce the strength of the existing shear walls shall be made without prior NRC approval; and

b. In the event that an earthquake occurs that exceeds the facility criteria for a 0.11g peak ground acceleration at the plant site, the facility shall be brought to a cold shutdown condition and inspected to determine the effects of the earthquake on the facility.

8. In the Order of May 26, the Office of Nuclear Reactor Regulation provided interested persons the opportunity to request a hearing with respect to two issues related to the nonconformance:

a. Whether interim operation prior to the modifications required by the order should be permitted; and

b. Whether the scope and timeliness of the modification required by the order to bring the facility into substantial compliance with the license are adequate from a safety standpoint.

The Order of May 26 provided that it would not become effective until "expiration of the period during which a hearing may be requested or, in the event that a hearing is requested and held, on the date specified in an order made following the hearing."

9. On June 14, 1978, the Licensee petitioned the Nuclear Regulatory Commissioners for an emergency order allowing temporary operation of the Plant. In a response dated June 23, 1978, the NRC Staff opposed Licensee's petition. The Commissioners considered the petition at a public meeting held on July 5, 1978 and, in an order dated July 7, 1978, denied Licensee's petition. However, the Commissioners did state that "in the event that a hearing takes place, we direct the Licensing Board to proceed expeditiously, consistent with arriving at a sound decision."

10. In response to the Order of May 26, several persons requested that a hearing be held. An Atomic Safety and Licensing Board ("Board") was duly established on June 29, 1978, to rule on petitions and/or requests for leave to intervene with respect

to the Order of May 26.[2] The Board, on July 7, issued a Notice and Order for Special Prehearing Conference to be held on July 24 and 25, 1978 to consider the requests. In addition to those matters, scheduling and discovery were also discussed at this prehearing conference. The Licensee voluntarily committed to make documents available at its Portland offices for discovery by the intervenors beginning on July 27 (Tr. 305-06).

11. Following the prehearing conference, the Board on July 27, 1978, issued an order granting the hearing requests and intervention petitions of Ms. Nina Bell, Mr. Eugene Rosolie on behalf of himself and the Coalition for Safe Power (collectively "CSP"), Columbia Environmental Council ("CEC"), Mr. David McCoy, Ms. C. Gail Parson, and Mr. Stephen Willingham. For all purposes in the proceeding, Ms. Bell, Mr. McCoy, and Ms. Parson were consolidated as one party (the "Consolidated Intervenors"). The Board also granted the State of Oregon's petition to participate as an interested State and deferred action on the intervention petition of the Bonneville Power Administration ("BPA"). (Order Concerning Requests for Hearing and Intervention Petitions.)

12. A second prehearing conference was scheduled for August 14, 1978, to consider all pertinent matters as set forth in 10 CFR §2.751a. Prior to the prehearing conference the Board issued a notice on August 1, scheduling the evidentiary hearing

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2/ This same Board subsequently was authorized to preside over the evidentiary hearings.

in this matter to begin September 6-8, 1978 in Portland, Oregon. The notice also required that testimony be filed at least 15 days prior to the hearing session at which it would be presented. (Notice of Evidentiary Hearing.)

13. At the August 14 prehearing conference the Board ruled that the intervention petition of BPA would be granted on the condition that BPA be consolidated with the Licensee for all purposes (Tr. 6498).[3/] The Board next considered and granted Licensee's motion to bifurcate the hearings, i.e. take evidence, consider proposed findings of fact and conclusions of law, and then reach a partial, initial decision with regard to the issue of interim operation of the Plant prior to addressing the issue of scope and timeliness of the proposed modification (Tr. 6499-6508).

14. After hearing a report from the parties on the progress of discovery, the Board considered and rejected the motion of Intervenor Willingham, supported in writing by Consolidated Intervenor McCoy, to postpone the start of the evidentiary hearing until September 27 (Tr. 6541-66, 6585-87). The Board also ruled that there was no need for stated contentions with respect to the issue of interim operation because the notice of opportunity for hearing set forth with sufficient precision the issue to be determined (Tr. 6584-85).

3/ The transcript pages for the second prehearing conference were mistakenly numbered 6494-6617, rather than 316-439.

15. The above rulings of the Board at the August 14 pre-hearing conference were formalized in the Board's order of August 25.

16. After the Licensee received preliminary information of the results of a new, finite element analysis of the seismic capability of the Control Building, it notified the Board and parties on August 22 that it could not prefile its testimony on that date, but would do so after it had had additional time within which to review the new information (Broehl, Licensee Exh. 13, p. 4). The Board postponed the scheduled hearing accordingly.

17. On August 28, 1978, representatives of the Licensee met with the NRC Staff and the State of Oregon to discuss the new information. (Ibid.) After further review, on September 20, 1978, Licensee served on the Board and all parties the results of its analysis of the new information. Thereafter, on October 2 the Board rescheduled the evidentiary hearing to commence on October 23 in Salem, Oregon. The Board also scheduled limited appearance statements to be heard October 26-27 in Portland, Oregon.

18. The Licensee prefiled the written testimony of its potential witnesses Donald J. Broehl (Broehl), S. R. Christensen (Christensen), Bart D. Withers (Withers), Professors Myle J. Holley, Jr. and Boris Bresler (Holley-Bresler) and Richard C. Anderson, George Katanics, Theodore E. Johnson and Dr. William H. White (Anderson, et al.) on October 3. On October 6, the

State of Oregon prefiled testimony for its witness, Dr. Harold I. Laursen (Laursen). On October 13, the NRC Staff prefiled testimony for its witnesses Kenneth S. Herring (Herring I), Robert T. Dodds (Dodds) and James E. Knight (Knight). Additional testimony for Mr. Herring (Herring II) was prefiled on October 16.

II. EVIDENTIARY HEARING

19. The evidentiary hearing commenced in Salem on October 23 and witnesses presented their testimony as follows:

For Licensee	
Anderson, et al.	October 23-25
Holley-Bresler	October 25, October 30-31
Broehl, Christensen Frewing and Withers	October 31 and November 1
For State of Oregon	
Laursen	November 2
For NRC Staff	
Dodds	October 31
Herring, Knight and Trammell	November 2

Intervenors CEC, CSP, and Consolidated Intervenors (through Ms. Bell) attended the hearing and conducted cross-examination of the other parties' witnesses, but presented no witnesses of their own.

20. On October 26 and 27 the Board heard limited appearance statements in Portland (Tr. 1047-1410). The Board also accepted all written limited appearance statements handed up at any time during the evidentiary hearings (Tr. 528, 632, 1516).

21. The Board determined on the first day of the hearing that questions with respect to floor response spectra and equipment qualification would not be addressed until testimony on those matters had been submitted and all parties had had

reasonable opportunity to review it (Tr. 516-27). The Licensee submitted written supplementary answers to questions of the NRC Staff on the subject on October 27 and November 2 (Licensee Exhs. 19 and 20). Although the Board examined Messrs. Anderson and White, the Licensee's witnesses on this subject, on November 3 (Tr. 2350-83), the other parties indicated that they had not had sufficient time to analyze the testimony (Tr. 2325, 2329, 2384).

22. The Board indicated that schedule conflicts of its members prevented resumption of the hearings until December 11 (Tr. 1415, 2296-97, 2328-31). The Board suggested that the hearings could be concluded by depositions; however, the use of this process was dependent upon agreement of all parties (Tr. 2327-28). On November 3, the Board closed the record on all matters other than qualification of the safety-related equipment in the Control-Auxiliary-Fuel Buildings based on the STARDYNE floor response spectra. The Board also stated that all proposed findings with respect to matters other than qualification of the equipment on the basis of STARDYNE floor response spectra should be submitted by November 20. (Tr. 2315-20; see also Board Order Regarding Conclusion of Evidentiary Hearings on Interim Operation, dated November 6, 1978).

FINDINGS OF FACT

III. STRUCTURAL EVALUATIONS AND ANALYSES

A. Description of Building Complex

23. The Control Building, the Auxiliary Building, and the Fuel Building (all three collectively referred to as the "Complex") are interconnected by their foundation systems and floor slabs. In the Fuel Building up to an elevation of 48 feet above grade, the lateral resisting members consist of a conventionally reinforced concrete fuel pool and hold-up tank enclosures, connected by reinforced concrete floor slabs. The upper portion of the Fuel Building is structural steel. The Auxiliary Building is between the Fuel Building and Control Building. The Auxiliary Building is supported laterally in part by the Control Building on one end and the Fuel Building on the other, with the reinforced concrete floor slabs acting as diaphragms to transfer lateral loads. (Anderson et al., Licensee Exh. 10, p. 4).

24. The Control Building is composed of a structural steel framing system with steel beams and columns supporting reinforced concrete floor slabs, and with shear walls designed to resist lateral seismic loading. The major shear walls are located around the perimeter of the building and are generally composed of reinforced concrete core placed between two layers of reinforced concrete block walls. The block walls generally sandwich the structural steel frame so that the reinforced concrete core is partially or completely interrupted by the steel frame members. (Ibid., pp. 2-3).

25. The major reason for this composite type of construction in the Control Building was the desirability of first erecting a steel frame building with reinforced concrete slab floors that would allow continued work inside the structure protected from the rain during the winter. Concrete block walls were then erected and reinforced concrete cores were placed between the two layers of reinforced, fully grouted concrete block masonry. The design was uniquely suited to the Trojan site because of its mild, yet wet, winter climate and has not been used elsewhere in nuclear power plant design by Bechtel. (Ibid., pp. 3-4; Tr. 620-21 (Anderson)).

B. Seismic Design Criteria.

26. The criteria which guide the Nuclear Regulatory Commission in its evaluation of a nuclear power plant's seismic design bases are set forth in the Commission's regulations at 10 CFR Part 100, Appendix A. The regulations describe the approach used to assess conservatively a site's potential for seismic activity, the resultant effects of seismic events against which the plant must be designed, and the particular structures, systems and components of the plant whose design must reflect those considerations. The basic parameter for seismic design is effective peak ground acceleration seen by the plant expressed as a fraction of "g" level (gravity acceleration of the earth). There are two "g" levels of interest in the design and operation of a nuclear plant. (Anderson, et al., Licensee Exh. 10, p. 7).

27. In NRC terminology, the "Safe Shutdown Earthquake" ("SSE") defines that earthquake which has commonly been referred to as the "design basis earthquake". Based upon an evaluation of the maximum earthquake potential at any site, the SSE is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components at a nuclear plant are designed to remain functional. (Ibid., p. 8).

28. In addition to the design basis SSE, Appendix A to 10 CFR Part 100 provides for the establishment of an Operating Basis Earthquake ("OBE") -- a lower level earthquake than the SSE. Appendix A to 10 CFR Part 100 provides that if vibratory ground motion exceeding that of the OBE occurs, shutdown of the plant will be required. Prior to resuming operations, a licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public. Although Appendix A to 10 CFR Part 100 contemplates that the OBE will be established at one-half of the SSE, that provision (like the entirety of Part 100) is only guidance to the NRC in the evaluation of the suitability of plant sites and design bases and does not constitute an inflexible requirement. Thus it is recognized that an OBE can be established at more or less than one-half of the SSE, and its selection can take into account operational considerations such as the potential for the need to shut down and inspect, which increases as the selected OBE value is lowered. (Ibid., pp. 19-21; Tr. 757 (Anderson); Tr. 1975-76 (Broehl); Tr. 2251 (Trammell)).

29. The SSE and OBE for the Plant were established at the time the construction permit was issued in 1971. As defined in Section 2.5.2 of the FSAR, the SSE for Trojan is 0.25g; the OBE, 0.15g.

30. While the OBE is always by definition a less severe earthquake than the SSE, the design of nuclear structures may be governed by the OBE rather than the SSE. This was the case for the Plant. The situation results from the application of more conservatism in evaluating and designing for the OBE. The principal additional conservatisms applied for the OBE are lower damping values[4/] (2 percent rather than 5 percent), and a greater load factor[5/] (1.4 rather than 1.0). The necessary effect of designing to meet the selected OBE of 0.15g for the Plant would have been a Control Building with a design SSE capability above the requirements of the 0.25g SSE. In other words, designing for the capacity to resist the factored OBE of 0.15g at 2 percent damping would have resulted in a design SSE capability at 5 percent damping of approximately 0.34g -- an SSE capability roughly one-third in excess of the specified design basis SSE and due solely to designing for the OBE. (Anderson, et al., Licensee Exh. 10, pp. 20-22; Tr. 1443 (Holley)).

4/ Damping is discussed in ¶¶ 58-61, infra.

5/ A load factor is a multiplier (e.g., 1.4) used to compensate for uncertainties in determining design loads (Tr. 868 (Johnson); Tr. 1445 (Holley); Tr. 2232-33 (Herring)).

C. Evaluations and Analyses of Building Complex.

1. Original Evaluations and Analyses, 1970-71

31. The original seismic evaluation of the Control Building in 1971 used the criteria defined and described in Sections 3.7 and 3.8 of the FSAR. The structural model used was a fixed-base beam-stick model. The analysis considered the Control, Auxiliary and Fuel Buildings, but the Auxiliary Building was considered to have no lateral resistance except for a few walls. The analysis was based on the following:

(a) The mass considered was based on the design dead weight (i.e., the structure's own weight) and 50 percent of the specified floor live load.

(b) The stiffness was based on uncracked section properties.

(c) The modal analysis spectrum response technique was used for the determination of inertia loads.

(d) The modal responses were combined using the absolute sum technique, although the square root of the sum of the squares ("SRSS") technique was also available and considered an equally acceptable technique at the time.

(Anderson, et al., Licensee Exh. 10, p. 9; Licensee Exh. 6, Attachment 1, pp. 2-3).

2. Initial Re-evaluation Following Discovery of Design Deficiency, April-May, 1978.

32. As a result of the discovery of the design deficiency (described in ¶6, supra), a detailed re-evaluation study of the Control Building in its existing configuration was performed for the Licensee by Bechtel in April, 1978, to assess its capability to withstand the SSE and meet the OBE criteria. (Herring I, NRC Staff Exh. 5, p. 7; Anderson, et al., Licensee Exh. 10, pp. 8-9; Tr. 915 (White)).

33. The re-evaluation study showed that while the structure continues to satisfy design criteria for the 0.25g SSE (as a result of the OBE controlling design), its ability to meet the OBE criteria has been reduced to 0.11g -- less than the intended OBE capacity of 0.15g (Licensee Exh. 6, Reportable Occurrence, pp. 2-3; Anderson, et al., Licensee Exh. 10, p. 20; Broehl, Licensee Exh. 13, p. 3). The effect of the design deficiency was to reduce the capacity of the Control Building from that originally intended by about 15-30 percent (Tr. 978 (Johnson); Tr. 1583-84 (Bresler); Tr. 2292 (Herring)).

34. The technique used for the re-evaluation study was to employ the original beam-stick model analysis and the design done in 1970, adjusted and modified in the following respects:

(a) Rather than the design compressive strengths of 5,000 psi for the concrete and the grout, the 90-day compressive strengths for the concrete and the 28-day compressive strengths for the grout indicate that the actual design compressive strengths of these materials are in excess of 6,000 psi. The higher strength value was utilized for both materials in the re-evaluation study. (Anderson, et al., Licensee Exh. 10, p. 10; Licensee Exh. 6, Appendix C). This is a justifiable approach. (Herring I, NRC Staff Exh. 5, pp. 8-9, 17; Tr. 1767 (Bresler); Tr. 2108 (Laursen)).

(b) Rather than the 40,000 psi yield strength assumed for the steel reinforcement in the original design, a value of 45,000 psi was utilized in the re-evaluation study. At the time of the original design, the assumed yield strength was based on the specified minimum strength of reinforcement steel called for in the design. The yield strength used in the re-evaluation study was based on the minimum value taken from mill certificates for the actual batches of reinforcement which were placed in the walls. (Anderson, et al., Licensee Exh. 10, p. 10; Licensee Exh. 6, Appendix C). This, too, is a justifiable approach. (Herring I, NRC Staff Exh. 5, pp. 8, 17; Tr. 1767-68 (Bresler); Tr. 2108 (Laursen)).

(c) In determining the seismic loadings, different masses were used in the original analysis and in the re-evaluation study. At the time of the original analysis, the final weights for the Control Building and its contents were not defined precisely. Consideration of the as-built weight information resulted in loads which were 87 percent[6/] of the original design weight. (Anderson, et al., Licensee Exh. 10, p. 11). The inclusion of the as-built weight criteria in the re-evaluation is a reasonable approach. (Herring I, NRC Staff Exh. 5, pp. 8, 16; Tr. 2108 (Laursen)).

(d) The original design assumed that the Control and Fuel Buildings provided almost the entire lateral support for the Auxiliary Building. Only two Auxiliary Building walls in each direction were considered as shear resisting members in the original seismic analysis. In the re-evaluation study, some other walls in the Auxiliary Building were

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6/ The as-built weight calculation was refined for purposes of the subsequent finite element analysis and the actual weight determined to be about 92 percent of the original design weight. (Licensee Exh. 8, Appendix A, pp. A-8 through A-9; Tr. 1768 (Bresler)).

credited with carrying a portion of the lateral load originally assumed to be carried by the Control Building. (Anderson, et al., Licensee Exh. 10, p. 11; Tr. 914 (White)). Only the reinforcing steel in these additional Auxiliary Building walls was considered. (Licensee, Exh. 9-A, p. 7). It is acceptable to rely upon the capacity of other capable shear walls in the Auxiliary Building to carry part of its lateral loading. (Herring I, NRC Staff Exh. 5, pp. 9-10, 17; Tr. 1768-69 (Bresler); Tr. 2109 (Laursen)).

(e) With respect to the seismic forces on the Control Building, the original seismic loadings were derived by combining modal responses using the absolute sum technique, although the original seismic analysis criteria allowed modal response combination by either this method or the more realistic SRSS technique. Utilization of the SRSS technique for the combination of modal responses in the re-evaluation study resulted in loads which are 80 percent of the loads computed by the absolute sum technique. (Anderson, et al., Licensee Exh. 10, p. 11). This procedure is acceptable since the use of the SRSS combination of modal responses was allowed by the original criteria, is acceptable under current standards, and has proved to provide more realistic values. (Herring I, NRC Staff Exh. 5, pp. 7, 16; Tr. 1612 (Bresler); Tr. 2109 (Laursen)).

35. In the re-evaluation study, ACI shear formulae were correctly applied. Where in the original analysis a concrete shear strength of $3.5\sqrt{f_c'}$ had been used, in the re-evaluation study the ACI 318 shear formulae were applied with a conservative concrete shear strength of $2\sqrt{f_c'}$. (Anderson, et al., Licensee Exh. 10, p. 6, n. 3). Also, only the continuous and adequately embedded reinforcing steel was utilized for the capacity determination. (Licensee Exh. 6, Attachment 3, p. 2). The grout, with strength comparable to the concrete, infills approximately one-half the block area. Thus, it is reasonable to use an equivalent thickness (core thickness plus one-half block thickness) in the ACI 318 formulae to calculate the composite wall shear capacity. The discontinuous rebar in the core is neglected, even though it will contribute to the restraint of the concrete. Also, a concrete shear strength of $2\sqrt{f_c'}$ is a conservative approximation of the strength of deep concrete sections such as the shear walls. (Herring I, NRC Staff Exh. 5, pp. 9, 18; Tr. 568 (Anderson); Tr. 581-83, 585-88 (Johnson); Tr. 584-85 (Katanics); Tr. 1584 (Bresler); Tr. 2099 (Laursen)).

36. Further confidence in the capability of the Control Building was provided by evaluating the "dowel action" capacity of the reinforcing steel and columns. (Herring I, NRC Staff Exh. 5, p. 27). For this evaluation, the walls are assumed to be cracked all the way through the most critical horizontal plane so that the entire horizontal shear force is resisted only by the adequately embedded reinforcement bars and fully embedded steel columns acting as "dowels" across the crack. (Anderson, et al., Licensee Exh. 10, pp. 12-13; Tr. 751-52 (Johnson)). Not all walls had dowel capacity greater than the lower of their shear or moment-controlled capacity; however, these walls are compensated for by other walls having dowel capacities greater than the lower of their shear or moment-controlled capacity. The shear friction contribution, which was conservatively neglected, would also compensate for the percentage by which load exceeded dowel capacity in these walls. This dowel capacity evaluation demonstrated that the structure has a minimum dowel capacity approximately 1.4 times that required to resist SSE across a given elevation. In addition, since inelastic structural response would limit the seismic forces to lower levels than would be calculated from a linear-elastic dynamic seismic analysis, it was concluded that even with unrealistically conservative assumptions, the structure has substantial excess ultimate capacity to resist SSE loads. (Herring I, NRC Staff Exh. 5, pp. 14-16, 27-28; Anderson, et al., Licensee Exh. 10, p. 13; Licensee Exh. 7, pp. 4-1 through 5-2).

37. Based on these considerations and evaluations, the Licensee determined that the as-built Control Building can safely withstand the 0.25g SSE with 5 percent damping and that it can meet criteria for a 1.4 factored OBE of 0.11g with 2 percent damping (Licensee Exh. 6, Reportable Occurrence, pp. 2-3). The NRC Staff concluded upon review of these evaluations that there is reasonable assurance that the facility as constructed will safely withstand the 0.25g SSE, but that it should be shut down and inspected in the event that during the period of interim operation an earthquake occurs with an effective peak ground acceleration of 0.11g (Herring I, NRC Staff Exh. 5, pp. 11, 31-32, 34). Licensee's independent consultants similarly testified at the hearings that the evaluations were appropriate and adequate to support these determinations (Tr. 1037-41, 1778 (Holley and Bresler)).

38. Still further confirmation of the structural capability of the as-built Control Building and, as well, of the conservatism of the re-evaluation study was obtained by comparisons of the results of the re-evaluation study with the results of an additional analysis performed by an independent consultant to Bechtel in June, 1978, using the TABS (Three-dimensional Analysis of Building Systems) program. The TABS model idealizes the building system as an assembly of a system of independent plane frame and shear wall elements interconnected by floor diaphrams which are rigid in their own plane. The outputs of the TABS program (moments and shear forces on the walls) indicated internal

loads lower than the re-evaluation study. Although for this particular application, TABS is of questionable accuracy (since the program is unable to represent adequately the flange effect of cross walls or the effect of box-type shear wall systems such as the Fuel Building Hold-Up Tank Enclosure and Spent Fuel Pool), when allowances are made for the inadequacies in the TABS approach, the results obtained do tend to confirm the results of both the re-evaluation study and the subsequent STARDYNE analysis (discussed in Section III.C.3, infra). (Anderson, et al., Licensee Exh. 10, p. 12; Licensee Exh. 8, Appendix A, pp. A-4 through A-5; Tr. 916-17 (White); Tr. 645, 920-21 (Johnson); Tr. 1040 (Holley)).

3. Finite Element Analyses, August-September 1978.

39. For further confirmation of the validity of the re-evaluation study and at the suggestion of Licensee's independent consultants, Bechtel performed finite element computer analyses. These analyses were initiated in August, 1978 to determine the seismic response of the Complex. The STARDYNE finite element computer program was used. (Anderson, et al. Licensee Exh. 10, pp. 13-14; Tr. 1036, 1041 (Holley)).

40. The finite element model for the Complex was originally developed over a period of three months for the analysis of the as-built structure combined with a proposed structural extension at the north end of the Control Building (a proposed modification to strengthen the existing structure and substantially restore the originally intended margins). A finite element

analytical approach with mathematical sophistication beyond that of the beam-stick analysis more accurately evaluates interaction between the existing structure and the proposed extension. In the beam-stick model employed in the original analysis and in the re-evaluation study, all the mass associated with each of the floors in the Control Building, Auxiliary Building, Hold-up Tank enclosure structure and Fuel Pool were lumped into one concentrated mass with these concentrated masses interconnected by vertical sticks and horizontal beams representing the stiffness characteristics of the walls and floors respectively. In the finite element model developed for the STARDYNE program, the Complex is represented by approximately 460 nodal points tied together by 685 plate elements representing walls and floors and 56 beam elements. The program solves approximately simultaneously 600 equations of motion which describe the dynamic responses of the Complex. This finite element model provides an excellent representation of the actual mass and stiffness distribution within the Complex. (Anderson, et al., Licensee Exh. 10, pp. 13-15; Herring II, NRC Staff Exh. 6, pp. 2-3, 7; Holley-Bresler, Licensee Exh. 12, p. 3; Tr. 753-54 (White)).

41. The seismic loads resulting from STARDYNE were somewhat higher than those predicted by the re-evaluation study but were less than those used in the original design. A comparison of the total seismic loads from the STARDYNE analysis and the re-evaluation study for the entire Complex (19500 kips and

18480 kips, respectively) indicates that the base she ɩesults of the models do not differ substantially. (Herring II, NRC Staff Exh. 6, pp. 2-3, 7; Anderson, et al., Licensee Exh. 1', p. 15; Licensee Exh. 8, Appendix A, pp. A-7 through A-8)

42. The STARDYNE analysis predicted greater torsional contributions to the loading of the Control Building than did the beam-stick model. Using STARDYNE, a twisting mode was predicted, with the Complex pivoting about the more rigi. Fuel Building and with greater modal deflection at the Control Building end in the North-South direction. The increase in the total base shear for the Control Building in the North-South direction predicted by the STARDYNE analysis over that predicted by the re-evaluation study was about 20 percent, while the predicted Fuel Building base shear decreased. (Herring II, NRC Staff Exh. 6, p.8; Anderson, et al., Licensee Exh. 10, p. 15; Licensee Exh. 8, App. A, p. A-6).

43. With a sophisticated finite element analysis for determining loads, the use of code allowable capacities would be inappropriate (Tr. 576 (Johnson); Tr. 2084 (Laursen)). The allowable capacities in applicable codes are usually set anticipating a certain level of sophistication when determining the applied loads. The shear provisions of the ACI and UBC codes have not changed significantly for several years. It appears that these codes have not considered that users would be applying techniques as sophisticated as an extensive finite element analysis which can consider the flexibility of an entire

complex of walls and floor slabs and mathematically distribute the loads throughout the elastic system. Moreover, the code provisions for determining shear capacity in walls are based on walls which have sufficient height when compared to their base dimension that 45°-type diagonal tension cracks can develop, essentially running to both outer edges of the wall. In the case of the Control Building, many of the walls are quite short in height compared to their length and such cracks cannot develop. Therefore, particularly as to these walls, the code provisions ae unnecessarily conservative. (Anderson, et. al., Licensee Exh. 20, p. 16; Licensee Exhibit 8, p. 4-1; Laursen, State Exh. 1, p. 10; Tr. 1480-81 (Bresler)).

44. Licensee developed more realistic criteria for assessing the shear capacities of the walls for comparison with the finite element model predicted loads. Empirical relationships derived from tests performed by Professor R. R. Schneider at California Polytechnic College and recent concrete block (masonry) shear wall test results obtained by the University of California at Berkeley were used as the bases for these new shear capacity criteria. These criteria were compared with the Schneider and Berkeley masonry test results, Portland Cement Association concrete shear wall test results and current Uniform Building Code allowable stresses and showed good correlation and conservative capacity values. (Licensee Exh. 8, Appendix B; Tr. 598, 837, 975-78 (Johnson); Tr. 2086 (Laursen); Tr. 2165 (Herring)).

45. The computed wall capacities were compared to the loads derived from the STARDYNE linear elastic analysis, which allowed members to reach their total elastic stiffness-derived loading, irrespective of capacity. The lower ratios of capacity to load for the members parallel to the North-South direction due to an earthquake in that direction indicate potentially greater nonlinear behavior than for the East-West direction. Thus, the North-South direction was considered in detail. (Herring II, NRC Staff Exh. 6, pp. 9-10; Licensee Exh. 8, pp. 5-1 through 5-2).

46. The redistribution ability of the Complex was studied in a variety of cases. The analyses demonstrated that the Complex has an excellent ability to redistribute loads, in the event that some walls yield. (Licensee Exh. 8, pp. 7-1 through 7-7, Tables 7-1 through 7-13; Tr. 929-30, 934-36 (Johnson and White); Tr. 2192 (Herring)).

47. The most realistic analysis considers an earthquake in the North-South direction with redistribution of forces among the walls when the capacities of all walls are limited. This case is the most realistic since the members governed by shear behavior were not loaded much beyond their calculated capacity in the analysis. This analysis shows the lowest capacity to load ratio for any major wall to be 1.153, which is associated with the west wall of the Control Building between elevations 45

feet and 61 feet.[7/] The redistribution of forces, which would take place when some of the walls reach their capacities, results in somewhat increased shear forces in those walls which have substantial reserve capacities. The sum of the computed capacities of these shear walls is well above the sum of the predicted maximum shear forces. Consequently, the group of walls resisting a given story shear can withstand forces imposed during a SSE event. (Licensee Exh. 8, Section 7; Herring II, NRC Staff Exh. 6, pp. 10-11; Holley-Bresler, Licensee Exh. 12, p. 6; Tr. 1744-45 (Holley)).

48. In order to assure the shear transfer capacity at the wall-slab and sidewall-endwall interfaces, these interfaces were investigated and were found to be adequate. The shear capacity of the slabs was evaluated in accordance with the shear friction

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7/ In view of the fact that this particular wall shows a ratio of capacity to load of 1.153, CEC attempted to argue that the Complex has a reduced margin of about 15% (Tr. 2191 (Kafoury)). Such argument misunderstands that the fact that a wall shows a ratio of 1.153 is not particularly significant. It is the displacement associated with a particular load that is significant (Tr. 1581-82 (Holley)). The record amply demonstrates that, even using conservative upper-bound calculations of maximum displacements, the ability of the Complex to safely withstand the SSE will not be adversely affected (See Section III. C. 4, infra). Additionally, it has been shown that the ratio for any particular wall is no problem because the structure has a good ability to redistribute forces and that the ultimate strength of other walls will not be exceeded such that there will be severe degradation of their load carrying capacity (Tr. 657-64 (Johnson); Tr. 2191-96 (Herring)). If the capacity to load ratio is of interest to quantify a "safety factor", comparison of load to group capacity is more appropriate (Tr. 1745 (Holley)). For this "more realistic case" (Case 4) at elevation 45 feet to 61 feet, it can be calculated that the group capacity to load ratio is 16720/11620 = 1.44 (Licensee Exh. 8, Table 7-11).

provisions of ACI 318-71. The wall-slab interfaces were found to have adequate dowel capacities with the exception of some horizontal interfaces within the wall in the lower elevations of the west wall. At these locations, however, a conservative contribution from shear friction due to dead load provides the necessary resistance in addition to the dowels. Shear transfer at the sidewall-endwall interfaces was found to be adequate considering the dowel capacity of the rebar and the beam-to-column connection capability, even while neglecting any shear transfer by the concrete. (Licensee Exh. 8, p. 6-1 and Table 6-2; Holley-Bresler, Licensee Exh. 12, pp. 6-7; Herring II, NRC Staff Exh. 6, p. 3).

49. In view of the fact that the steel framing of the structure is designed to carry the vertical loads and that the results of the above analyses considered forces resulting from linear-elastic seismic analysis and neglected the increased energy dissipation (and, therefore, lower seismic loading) which would result from any nonlinear behavior of the structure, all witnesses on structural integrity agreed that there is assurance that the structure will withstand the specified SSE of 0.25g with substantial margin. (Herring II, NRC Staff Exh. 6, pp. 11, 16; Anderson, et al., Licensee Exh. 10, pp. 26, 28. See also Sections III.C.6 and III.C.7., infra).

50. There is a conflict, however, in the testimony presented as to the level of earthquake at which it would be appropriate to require that the Plant be shut down and inspected, should

such an earthquake occur during interim operation. Licensee presented testimony demonstrating the ability of the structure to meet an OBE criterion of 0.11g; Licensee's position was based both on the beam-stick model evaluations and on the STARDYNE analyses (see ¶¶ 33, 37, supra; Licensee Exh. 8, p. 2-2). The NRC Staff initially agreed with this value (see ¶ 37, supra), but following the STARDYNE analyses, now recommends that the facility be shut down and inspected in the event that an earthquake which exceeds a 0.08g effective peak ground acceleration occurs during interim operation. This recommendation is based on the NRC Staff's conclusion that nonlinear behavior of a single wall, i.e., the west wall of the Control Building, could begin at an earthquake level (considering 2 percent structural damping) of about 0.087g. The NRC Staff rounded this number down for conservatism. The 0.08g level would establish an inspection point which takes into account NRC Staff views concerning possible cyclic degradation in the walls. (Herring II, NRC Staff Exh. 6, pp. 11-12, 16). Although Licensee believes this value is unduly conservative (Tr. 858-64 (Johnson)), it has committed to shut down the Plant and inspect it following any 0.08g earthquake occurring during the period of interim operation (Tr. 1807-08 (Broehl)). As the NRC Staff acknowledged, the level of 0.08g is sufficiently conservative that it would be an acceptable OBE for the as-built Plant for its lifetime (Tr. 2255-57 (Herring)).

51. Based on its review of the results of the STARDYNE analysis, the NRC Staff's basic conclusion as to questions of structural integrity continues to be that the originally intended margins of safety have been reduced and that the applicable codes are not fully satisfied, but that interim operation for a period necessary to effect structural repairs and improvements continues to be appropriate. It also concludes that the original structural safety margins should be restored to the extent practicable in order to ensure adequate protection of the health and safety of the public during the long-term operation of the facility. (Herring II, NRC Staff Exh. 6, p. 16).

4. Structural Displacement Considerations.

52. Excessive building displacements may cause problems by buildings striking each other during a seismic event ("inter-structure displacement"), and excessive relative displacements between the floors of a building ("interstory displacement") may adversely affect the function of equipment supported by or attached to more than one floor. Interstructure displacements may also affect equipment (piping, cabletrays) running from one building to another.[8/] (Holley-Bresler, Licensee Exh. 12, pp. 10-13).

53. The maximum amounts of interstructure displacements between the Control Building and Containment and between the Control Building and Turbine Building were conservatively

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8/ Effects on equipment are discussed in Section IV, infra.

estimated considering nonlinear behavior of the Control Building. Interstory displacements within the Control Building were also calculated based on the same conservative assumption. (Herring II, NRC Staff Exh. 6, pp. 13-14).

54. The Control Building displacements were conservatively calculated using the loads from the STARDYNE elastic analysis for the west wall (the major and most highly loaded wall relative to its capacity) and a shear stress vs. shear strain curve derived from test data on both reinforced concrete and masonry walls (Herring II, NRC Staff Exh. 6, p. 13). These displacement evaluations indicated a maximum displacement at the top of the Control Building of 0.9 inch in the North-South direction and 0.09 inch in the East-West direction. The calculated maximum interstory displacement is 0.53 inch in the North-South direction between elevations 45 feet and 61 feet. (Licensee Exh. 8, Appendix D). These values constitute upper limit building displacement and interstory displacement (Herring II, NRC Staff Exh. 6, p. 13).

55. The Turbine Building displacements were refined from those in the original analysis considering the as-built condition of the building (Tr. 832 (Katanics); Tr. 1755 (Holley)).

56. In calculating interstructure displacements, relative displacements were added absolutely, which provides additional conservatism (Herring II, NRC Staff Exh. 6, p. 14). The maximum interstructure displacements determined from these analyses, about 2.4 and 2.49 inches between the Control and Turbine Buildings in the North-South and East-West directions,

respectively, at the top of the Control Building, and about 0.76 inch between the Control Building and Containment at Elevation 77 feet, were found to be acceptable (Ibid., p. 5; Tr. 1755 (Holley)). Thus there is adequate assurance that increased displacements resulting from nonlinear behavior of the Control Building will not have any adverse effects on public safety. (Herring II, NRC Staff Exh. 6, pp. 5, 14; Licensee Exh. 9-D, pp. 1-1 to 1-2; Licensee Exh. 9-E, Clarification No. 29; Holley-Bresler, Licensee Exh. 12, pp. 7-9, 12-13, 15).

5. Conservatisms in Analyses and Evaluations.

57. Confidence in the structural integrity of the Control Building and its ability to safely withstand the SSE is buttressed by consideration of a number of additional factors of convervatism inherent in the evaluations and analyses of the as-built Control Building. These factors provide assurance that the Complex has a considerable margin of safety and will withstand an earthquake at least 50 percent greater than the designated SSE of 0.25g (Anderson, et al., Licensee Exh. 10, pp. 27-28; Tr. 1472-76 (Holley and Bresler); Tr. 2097 (Laursen)). Among such factors of conservatism which the Board finds significant are the following:

58. a. Damping - The damping characteristics used to evaluate a structure constitute one of the major factors of conservatism in nuclear plant design. Damping is the ability of a structural system to dissipate vibratory energy (for example, when a spring is pulled and released, the amplitude of the

vibration becomes smaller and smaller from one cycle to another, and eventually the vibratory motion stops). This ability to dissipate vibratory energy is an inherent property of any real structural system. The degree of damping depends not only on the material, but also on the severity of vibration to which the structure is subjected. (Anderson, et al., Licensee Exh. 10, pp. 23-24).

59. The damping value of a structure is usually expressed in terms of the percentage of critical damping, for example, 2 percent or 5 percent (critical or 100 percent damping implies that the energy dissipation is so high that a vibratory motion will never be induced in the system). A higher damping value implies a higher energy dissipation ability of a structure. For building structures, the damping values are normally in the 2 percent to 15 percent range, depending on the structural materials and types, construction, and the level (amplitude) of vibratory motion. (Anderson, et al., Licensee Exh. 10, p. 24).

60. The selection of 2 percent damping for the design of the Control Building for the OBE reflects substantial conservatism. If 4 percent damping were used instead of 2 percent, the OBE which the Control Building presently has the capacity to resist, would increase from 0.11g to 0.15g. (Anderson, et al., Licensee Exh. 10, p. 25).

61. Licensee and its consultants assumed 5 percent damping for calculation of responses to the SSE event in accordance with FSAR requirements. This is reasonable for the reinforced

concrete floor slabs for which it is reported that in-plane capacities have been found to be well in excess of the imposed forces. However, for the masonry shear walls, subjected to large fractions of their computed capacities, substantially higher damping percentages would be appropriate. Such higher damping would lead to smaller predicted forces and displacements. The use of more realistic, higher damping percentages is not permitted by the NRC, but they represent an unaccounted-for conservatism in all the analyses. (Holley-Bresler, Licensee Exh. 12, p. 9; Anderson, et al., Licensee Exh. 10, p. 25).

62. b. Inelastic deformation - In addition to damping, design conservatism also results from the ability of building structures to deform inelastically (beyond the limit at which some permanent deformation occurs) before reaching their ultimate capacities. In the event of a severe earthquake, this ability to deform inelastically will increase the ability of the structure to dissipate vibratory energy. (Anderson, et al., Licensee Exh. 10, p. 26; Herring I, NRC Staff Exh. 5, p. 28; Herring II, NRC Staff Exh. 6, p. 11; Tr. 646 (Johnson)).

63. c. Dynamic vs. Static Strengths - It is also recognized that under a dynamic loading environment such as earthquake loadings, the dynamic strengths of building material are higher than their static strengths. Each of the evaluations of the Control Building was conservatively based upon the static strength of materials. Also, the actual ultimate strengths are larger than the ultimate strengths of materials used in the re-evaluation study and supplementary evaluations. (Anderson, et al., Licensee Exh. 10, pp. 26-27; Tr. 911-13).

64. d. Foundation damping - The various seismic analyses of the Plant assumed that it is founded on rock with infinite rigidity. This is a conservative assumption because the infinitely rigid foundation does not allow any structural vibratory energy to be absorbed by or dissipated into the foundation. Incorporating the actual elastic properties of the Trojan foundation material and the energy dissipation (radiation damping) characteristics of the foundation system in the seismic analyses would lead to lower but more realistic seismic responses. (Anderson, et al., Licensee Exh. 10, p. 27; Tr. 740-41, 782, 921-22 (Johnson); Herring II, NRC Staff Exh. 6, p. 7).

65. e. Steel frame - The steel frame of the Control Building is designed to carry the weight of the building (dead load) while the shear walls provide structural resistance to shear loads (Tr. 738 (White)). Nonetheless, the steel frame can be expected to contribute significantly to the structural integrity of the Control Building by preventing building collapse (Herring II, NRC Staff Exh. 6, p. 11), even in a postulated earthquake substantially in excess of the designated 0.25g SSE (Holley-Bresler, Licensee Exh. 12, p. 10; Tr. 1469 (Bresler)).

66. In addition to the above factors, both the NRC Staff and Licensee in their testimony discussed further conservative assumptions and considerations providing still further assurance of substantial margin over the designated SSE. Such assumptions and considerations include: increasing strength of concrete with

age; higher capacities for most walls based on tests of concrete and masonry walls; reduced nonlinear behavior of the structure as a whole (and therefore lesser effects of cyclic degradation) due to these higher wall capacities and to the tensile capacity of concrete, which were neglected in analysis; conservative application of the Schneider criteria in the Supplemental Structural Evaluation (Licensee Exh. 8); and use of criteria based on masonry testing with material strengths approximately one-half of the strengths of the materials in the Control Building walls. (Herring I, NRC Staff Exh. 5, pp. 17-27; Licensee Exh. 8, pp. C-1 through C-3; Tr. 2165 (Herring); Tr. 598, 976-77 (Johnson); Tr. 913 (Katanics)).

6. Independent Consultant Evaluations and Analyses

67. Licensee also presented testimony by Professors Myle J. Holley, Jr. and Boris Bresler confirming the ability of the as-built Control Building to withstand the design basis SSE of 0.25g. Professor Holley is a registered engineer who has had lengthy experience with concrete structures, has taught structural engineering for almost 30 years at MIT and has consulted extensively concerning the structural design of nuclear power plants (Holley-Bresler, Licensee Exh. 12, attached statement of qualifications of Myle J. Holley, Jr.; Tr. 1024-25). Professor Bresler is also a registered engineer; he has engaged in teaching and research in structural engineering at the University of California, Berkeley since 1946. He has extensive consulting experience, has participated in developing systematic

procedures for seismic evaluation of existing buildings and, for approximately 20 years, has been active on technical committees of the American Concrete Institute, including the Committee on Nuclear Reactor Structures (Holley-Bresler, Licensee Exh. 12, attached statement of qualifications of Boris Bresler; Tr. 1027-28).

68. Professors Holley and Bresler submitted their testimony in the form of a study report entitled "Response of Trojan Nuclear Power Plant Control Building to Specified SSE Event" (Licensee Exh. 12). This study was based on their review of the design and principal details of construction of the Complex, including relevant technical drawings, analyses, experimental data and other information used to judge the strength and performance of the walls; their tour of the Complex to observe its physical layout and construction features; and discussions with PGE and Bechtel engineers (Holley-Bresler, Licensee Exh. 12, pp. 1-2; Tr. 1030). They did not attempt to duplicate the extensive Bechtel analyses and calculations, since this was both impractical and unnecessary for their independent evaluation and judgment (Tr. 1032, 1431-34). The Holley-Bresler report outlines the purpose and bases of the authors' study, discusses relevant aspects of the evaluation (including methods of dynamic analysis; assessment of shear wall capacities, slab and slab-to-wall forces, and non-linear shear wall response; damping; the construction of the steel frame; and the displacement capacities of the building and equipment) and concludes

that the Control Building can withstand a 0.25g SSE with no consequences that could interfere with safe shutdown (Holley-Bresler, Licensee Exh. 12, p. 16; Tr. 1035).

69. Professors Holley and Bresler emphasized that, in their view, the significant consideration in evaluating the safety of the Control Building was the limited time-varying displacements to which the safety-related equipment in the building might be subject (Tr. 1032-33). They reviewed the STARDYNE model's prediction of displacements of 0.15 inch in the elastic range, which led Bechtel engineers to estimate (based on shear strains developed in test specimens when they reached their ultimate shear capacities) that maximum displacement at the level of the Control Building roof could not exceed 0.9 inch (Anderson, et al., Licensee Exh. 10, p. 30; Holley-Bresler, Licensee Exh. 12, p.8). The Holley-Bresler study independently validated this estimate of maximum displacement. By assuming reduced stiffness in the Control Building structure from that used in the STARDYNE analysis, Professors Holley and Bresler found that an estimated maximum displacement at the roof level of 1.0 inch was appropriate and that corresponding maximum floor-to-floor relative displacement would be less than this value. (Holley-Bresler, Licensee Exh. 12, pp. 8-9; Tr. 1757). In view of the Bechtel survey showing that safety-related equipment piping, and systems could withstand greater than 1 inch between floors and displacements of 3 to 6 inches between adjacent buildings (Licensee Exh. 11, p. 2), Professors Holley and Bresler were confident that these components

could withstand displacements substantially greater than the conservatively predicted displacements (Holley-Bresler, Licensee Exh. 12, p. 15-16; Tr. 1034, 1475, 1603). Evaluating the various dynamic analyses performed by Bechtel, the Holley-Bresler study concluded that the STARDYNE analysis realistically modeled the Control Building, and predicted useful upper-bound estimates of internal forces for the principal structural components (Ibid., p. 3; Tr. 1033-34, 1528, 1553-54, 1588, 1777).

70. The Holley-Bresler study buttressed its conclusions regarding the ability of safety-related equipment, piping and systems to withstand the maximum displacements expected during an SSE by noting the conservative effects of damping and the contribution of the structural steel. Although the structural damping of 5 percent utilized by Bechtel and by Professors Holley and Bresler for SSE would be higher if the shear walls were subjected to large fractions of their computed capacities, this increased damping was conservatively neglected (Ibid., p. 9; Tr. 1785-86, 1789). The structural steel framework's design to carry the entire gravity loading of the Control Building means that horizontal displacements many times the maximum displacements actually predicted would be required for building collapse, and therefore, such an event does not appear to be a possible consequence of a Trojan SSE (Ibid., p. 10; Tr. 1789). In response to Board inquiries, Professors Holley and Bresler expressed confidence that even an earthquake of 1.0g would not result in collapse of the structure (Tr. 1788-89).

71. Based on additional Bechtel evaluations and analyses, they also concluded that the maximum forces which could be imposed on the holdup tank enclosure and Spent Fuel Pool as a result of Control Building displacements were well below the capacities of these Fuel Building structures. They similarly concluded that clearances between the Control Building and the Turbine Building were sufficient to avoid contact between the two buildings. (Holley-Bresler, Licensee Exh. 12, pp. 11-13; Tr. 1591-92, 1755-56).

72. As a result of their study and investigation, Professors Holley and Bresler concluded that the Control Building, in its as-built condition, can withstand the specified SSE for Trojan with no consequences that could interfere with safe shutdown (Ibid., p. 16; Tr. 1035). On the basis of their extensive relevant background and experience, their demonstrated knowledge of the Complex and their thorough responses to numerous questions asked by both the parties and the Board, we have given significant weight to the positive conclusions reached by these two experts.

7. State of Oregon Evaluations and Analyses

73. Oregon, appearing in this proceeding as an interested State, presented testimony by Dr. Harold I. Laursen further confirming the ability of the as-built Control Building to withstand the Trojan specified SSE of 0.25g. Dr. Laursen is Professor of Structural Engineering at Oregon State University and a registered engineer in Oregon with considerable background

in structural analysis and fields of specialization in structural mechanics, structural engineering and computer applications of large structural analysis programs (Laursen, State Exh. 1, pp. 1-4). As in the case of Professors Holley and Bresler, the Board gives significant weight to Dr. Laursen's expert testimony based on his background and knowledge in the field of structural engineering.

74. Dr. Laursen based his testimony on his review of the extent to which sound engineering principles were applied in analyzing whether the existing shear walls of the Control Building could withstand the 0.25g SSE. Following numerous conversations with employees of PGE, Bechtel and the NRC Staff, Dr. Laursen utilized reports of the design deficiency, Control Building design drawings, and additional information regarding the construction of the Plant for his review. This review included the definition of appropriate design response spectra under both OBE and SSE conditions, the technique for modeling and analyzing the Complex under dynamic behavior, general features of the design and construction sequence potentially affecting the performance of the structure, the as-built characteristics of the structure, and the nature and seriousness of the design deficiencies. (Ibid., pp. 5-6; Tr. 2072-73). Dr. Laursen did not examine in detail or reconstruct the computer programs utilized by Bechtel, but is familiar with the types of programs utilized from his own applications of similar programs (Ibid., pp. 6-7; Tr. 2073).

75. When he first reviewed this matter following the Licensee Event Report of May 5, 1978, Dr. Laursen satisfied himself that the results of the original beam-stick computer model for the Complex were reasonable and that significant modes of vibration were properly considered in the dynamic analysis. He was initially concerned about credit being taken for the strength of the concrete core in light of the discontinuity of some of the core reinforcing steel (Ibid., pp. 7-8), but subsequently satisfied himself that dowel action of the reinforcing and structural steel would provide reserve strength to tie the Control Building walls together and afford desired ductility and damping (Ibid., pp. 8-9; Tr. 2080-82). Based on his review and analysis as of June 1978, Dr. Laursen prepared a report to the Oregon Department of Energy in which he concluded that there was reasonable assurance that the Control Building shear walls could withstand a 0.25g SSE (Ibid., p. 9).

76. Dr. Laursen's subsequent review of information developed in the STARDYNE analysis confirmed and provided further confidence in his original conclusion (Ibid., p. 12; Tr. 2086-91). Of particular importance to him was the fact that the criteria used to compare capacity to loads calculated by the STARDYNE analysis were based on tests (using conservative assumptions limiting allowable stresses) which provide assurance that the integrity of the concrete core will be maintained during an earthquake (Ibid., p. 11; Tr. 2086, 2098, 2100). Dr. Laursen concluded that the Control Building shear walls will experience

only limited cracking at SSE levels (Tr. 2098, 2106-07, 2112), and that the structure has a margin of safety of 50 percent above the SSE, thus providing additional confidence for his conclusion that it can withstand the SSE of 0.25g. It is his view that the structure would likely withstand a very large earthquake, such as 0.35g to 0.40g. (Tr. 2097-99, 2105, 2109-10).

D. Board's Determination

77. The bulk of the testimony in this proceeding related to demonstrating that the existing Complex can safely withstand the design basis SSE. Most of the cross-examination by the intervenors and most of the questioning by the Board were devoted to testing the basis for the testimony of the witnesses, as well as to exploring the margins by which the Complex can be said to meet the SSE. The Board is satisfied both from qualitative expert judgments of the various witnesses and from quantified factors of conservatism in the analyses and evaluations that, despite the admittedly reduced strength of the Control Building due to the deficiency, that structure and the interconnected Complex will safely withstand the 0.25g SSE, even if we are to assume that prior to that event the Plant might have previously experienced several earthquakes below the 0.08g inspection level. Not only are we convinced that the facility will withstand the SSE, but we find very persuasive the uncontroverted testimony of the Bechtel structural experts, with unqualified endorsement by acknowledged expert independent consultants, that the Complex will safely withstand the forces of earthquakes at

least 50 percent greater. (Anderson, et al., Licensee Exh. 10, pp. 27-28, Tr. 1472-76 (Bresler and Holley); Tr. 2110 (Laursen)). This is consistent with the testimony of the NRC Staff witness that, notwithstanding the deficiency, the as-built structure could safely withstand an SSE of 0.35g (Tr. 2291 (Herring)).

78. With respect to the OBE capacity of the as-built structure, the Board finds persuasive the Licensee's testimony demonstrating that the facility will withstand an OBE of 0.11g and the NRC Staff's concurrence in that value in its testimony with respect to the pre-STARDYNE analyses (see ¶¶ 33, 37, 50, supra). After the STARDYNE analyses, however, the NRC Staff testified that a level of 0.08g would be appropriate for purposes of shutdown and inspection during interim operation based on conservative calculations related to when first cracking might occur and to obviate any concerns regarding possible cyclic degradation (see ¶ 50, supra). Since the Licensee has agreed, for purposes of interim operation, to shut down and inspect the facility at the lower level of 0.08g, we find that such level is acceptable and that we need not decide whether it is unduly conservative. We do note that, in light of the NRC Staff's acknowledgement that this shutdown and inspection level would be acceptable for the as-built Plant for its lifetime (Tr. 2255-57 (Herring)), the specific period of interim operation need not be of concern.

IV. EVALUATIONS OF SAFETY-RELATED EQUIPMENT IN BUILDING COMPLEX

A. Evaluations and Analyses Considering Structural Displacement

79. The NRC Staff and the Licensee also submitted testimony regarding the ability of safety-related equipment, piping and electrical systems in the Complex to withstand displacements within or between buildings which might conceivably occur during an SSE at Trojan.

80. On September 27 and 28, 1978, Mr. James E. Knight of the NRC Staff visited the Plant and personally examined relevant safety-related cable runs and piping to determine whether seismic displacement might cause their failure (Knight, NRC Staff Exh. 7, p. 2). Mr. Knight's investigation revealed no case where cable runs might fail as a result of maximum postulated displacements, and only one instance where piping might be affected (service water lines to switchgear room coolers). In this latter case, even a complete piping break would not significantly impact the relevant switchgear equipment in the Turbine Building, since drains would carry the water away and there would be several hours to provide alternate methods (such as fans) for cooling equipment. There is also redundancy for the affected equipment. (Knight, NRC Staff Exh. 7, pp. 3-4; Licensee Exh. 9-E, Clarification No. 28; Tr. 2206-10). Mr. Knight concluded from his investigation that Plant systems required for safe shutdown, maintenance of the integrity of the reactor coolant system pressure boundary, and prevention

or mitigation of the consequences of accidents would not be affected by displacements which might occur during an SSE or by substantially higher displacements which might be postulated (Knight, NRC Staff Exh. 7, pp. 4-5).

81. In addition to the NRC Staff's investigation, Licensee has conducted its own analyses and surveys of safety-related equipment, piping and electrical systems. On the basis of a thorough on-site survey by qualified engineers, Bechtel concluded that the safety-related components in the Complex would withstand displacements greater than one inch between floors and three to six inches between adjacent buildings without loss of function. (Licensee Exh. 11, p. 2; Anderson, et al., Licensee Exh. 10, App. A, pp. 30-31; Tr. 801-05, 813-19, 999-1000, 1007 (Anderson, et al). This survey confirmed that safety-related systems and components would be able to withstand displacements well in excess of the maximum credible calculated displacements for the Complex. The survey also concluded that any potential effects from pieces of falling concrete (spalling) dislodged from a wall during a seismic event could not result in loss of function of safety-related equipment and components. (Licensee Exh. 11, p. 2; Anderson, et al., Licensee Exh. 10, App. A, pp. 31-32; Tr. 554-58, 564-66, 834-36 (Anderson and Johnson)).

82. Based on the results of the foregoing investigations and analyses by the NRC Staff, Bechtel and PGE, as well as the confirmatory evaluations by Professors Holley and Bresler, the

Board concludes that safety-related equipment and components required for safe shutdown of the Plant, maintenance of the integrity of reactor coolant systems and prevention or mitigation of the consequences of accidents will be able to withstand and remain functional despite displacements experienced during an SSE at the Plant.

B. Evaluations and Analyses Considering Response Spectra of Complex

[To be inserted later.]

V. BOARD QUESTIONS

83. At the special prehearing conference of August 14, we requested the Licensee and the NRC Staff to be prepared to respond at the hearing to a number of questions relevant to the decision we had to reach concerning interim operation (Tr. 6589-93).

84. A number of the questions concerned such matters as the relationship between the OBE and SSE and the reduction in safety margins in the existing as-built structure. These matters have been addressed in previous portions of this Initial Decision. We have discussed both the differences in the assumptions used in calculating the SSE and OBE and the relationships in the methods of calculations in the process of reaching our conclusion that the Complex still satisfies an SSE of 0.25g, even though the Plant now meets a reduced OBE criterion. (See Section III.B; ¶¶ 33, 37, 49-50, *supra*). We have also discussed the reductions in safety margins and pointed out the unanimous view of all witnesses that the structure can still withstand earthquakes well in excess of 0.25g, with a safety margin of at least 50 percent in the opinion of several experts (see ¶ 77, *supra*).

85. There remain to be discussed several Board questions which were not encompassed in previous portions of this decision. First, we inquired as to the effects upon safety-related equipment of a hypothetical structural failure due to the reduced shear capacity of the building (Tr. 6589-90).

86. The record demonstrates that an actual failure of the Control Building due to an SSE is not a credible event, in light of, _inter alia_, the steel framework, the strength of the structure and its ability to redistribute loads (_Anderson, et al._, Licensee Exh. 10, pp. 16-17; _Holley-Bresler_, Licensee Exh. 12, p. 10; Tr. 678-79 (Anderson); Tr. 687-88 (Johnson); Tr. 1032-33, 1471, 1527-29, 1565, 1572-75, 1581-82, 1600-01, 1751-52, 1756 (Holley and Bresler)). In fact, the witnesses had difficulty hypothesizing any event which would cause such failure (Tr. 676-79 (Anderson); Tr. 1787-89 (Holley and Bresler)). Thus, potential concerns center not on failure, but on the possiblity that displacements would exceed predicted levels. As discussed at length above, the record demonstrates that the ability to achieve safe shutdown, assure integrity of the reactor coolant system and prevent or mitigate consequences of serious accidents would not be affected by estimated displacements during an SSE and that, in fact, substantially higher displacements could be accommodated (¶¶ 69-70, 79-82, _supra_). Moreover, even though for spalling to occur displacement of the structure would have to be on the order of several feet (as opposed to the calculated conservative upper bound displacement of one inch) (Tr. 554-58, 564-66, 834-36, 875 (Anderson and Johnson)), the Licensee has examined the possible impact on safety-related equipment of a 10-lb fragment of concrete dislodged from a wall and has shown that no adverse effects would result (¶ 81, _supra_). The Licensee answered Board questions

concerning the ability to shut down the reactor safely postulating loss (regardless of the likelihood) of the Control Room. One witness expressed the view that the reactor could be shut down without use of any equipment in the Control Room (Tr. 670-74 (Anderson)), but it is apparent that this would be very difficult (Tr. 675 (Anderson); Tr. 1968 (Withers); Tr. 2175 (Knight)). However, there is no regulatory requirement that the Plant be designed with this capability; instead, it is only necessary that the Plant be able to be shut down from outside the Control Room if the Control Room must be evacuated for any reason. The testimony demonstrates that this can be accomplished and that Plant operators are knowledgeable as to necessary steps to be taken. (Tr. 1968-72 (Withers)). On the basis of the answers to our inquiries, we are satisfied that the reduced capacity of the Control Building walls will not adversely affect safety-related equipment within the Complex if a seismic event up to and including an SSE were to occur.

87. We next requested the Licensee to provide us with information concerning the Plant's seismic monitoring equipment and operational procedures in the event of an earthquake (Tr. 6591).

88. Licensee provided detailed testimony by its Manager of Generation Engineering, concerning the location and function of the three independent seismic instrumentation systems in operation at the Plant:

(1) A triaxial multi-element response spectrum recorder with peak shock annunciator,

(2) Five triaxial time-history recording accelerographs, and

(3) Seven triaxial peak recording accelerographs.

(Christensen, Licensee Exh. 15, pp. 1-3)

89. As several witnesses pointed out, the orientation of the instruments on the three principal axes (north-south, east-west and vertical) assures detection and accurate recording of seismic events regardless of orientation of the earthquake (Tr. 902-04, 944-48 (White); Tr. 1771-73 (Holley and Bresler); Tr. 1869-70 (Christensen)). The first system provides a permanent record of peak response accelerations of measurable ground motions in the three directions. The peak shock annunciator provides instantaneous visual indication in the Control Room of any earthquake in excess of OBE acceleration levels. Licensee has reset the peak shock annunciator to correspond to response acceleration for an OBE level of 0.11g for purposes of Plant shutdown and inspection during interim operation as a result of the NRC Order of May 26 (Christensen, Licensee Exh. 15, p. 2), and has committed to accept the NRC Staff's proposed level of 0.08g (Tr. 1807-08, 2043-44 (Broehl)). System 2 instrumentation provides data on frequency, amplitude, and mode shapes for the seismic responses of Trojan Category I structures, and is activated at a ground acceleration of 0.01g. System 3 provides additional data for the evaluation of the effect of an earthquake on structures and equipment. (Christensen, Licensee Exh. 15, pp. 2-3).

90. In the event of an earthquake exceeding the OBE level, the peak shock annunciator of System 1 would provide an immediate visual indication to the Plant operators. Such an earthquake would also be felt by Plant operators, providing confirmation that a seismic event had taken place (Withers, Licensee Exh. 16, p. 2; Tr. 1646 (Dodds); Tr. 2051 (Withers)). The time-history recorders' read-out in the Control Room (System 2) would also provide an audible indication of the earthquake. As the Plant Superintendent testified, under these circumstances the Plant's emergency procedures direct that the Plant be brought to a cold shutdown and inspected to determine the effects of the earthquake. (Ibid., pp. 1-2; Tr. 1959, 2050-54 (Withers)). In the process of achieving cold shutdown, the operators monitor Plant instrumentation to assess Plant status and the presence of any abnormal condition. They conduct an overall Plant inspection to identify any potential problems. This is followed by a more detailed inspection focusing primarily on any identified problems. During this period, Trojan technical personnel are called to the Plant to process the information recorded by the seismic instrumentation, which is then used for a detailed evaluation of the earthquake (Ibid., p. 2).

91. Once the Plant has been brought to cold shutdown, Licensee's management would proceed with a program to evaluate the effects of the earthquake. This would include engineering investigations in order to inspect Plant structures and to inspect and test mechanical and electrical systems. In addition,

information provided by seismic instrumentation would be analyzed to determine any variations in performance from seismic design bases and analytical predictions of structural and equipment behavior, as well as to identify any nonconformity with design criteria and Technical Specifications. All such investigations and analyses would be performed by qualified personnel, including consultants as required (Withers, Licensee Exh. 16, p. 3; Christensen, Licensee Exh. 15, p. 4; Tr. 1918 (Christensen)).

92. All seismic events would be reported to the NRC within 10 days, and any seismic event which exceeds an OBE and thus requires Plant shutdown would be reported to the NRC immediately. The Plant Superintendent further confirmed that reporting would take place immediately after assuring safe conditions at the Plant and that notification of the NRC by the Plant Superintendent or the Assistant Superintendent would be a priority action. (Tr. 1873-76 (Withers)). If OBE levels had been exceeded, Plant operation would not resume without the prior approval of the NRC (Withers, Licensee Exh. 16, p. 3; Christensen, Licensee Exh. 15, p. 4).

93. The Licensee's testimony demonstrates that the Plant's seismic instrumentation will provide immediate indication of an earthquake of OBE level and permanent data needed for evaluation of the event, that the Licensee has adopted appropriate procedures for actions to be taken during and after an earthquake and that the NRC will be notified in a timely manner.

94. We also asked the NRC Staff about the inspection and evaluation it would conduct following an earthquake in excess of an OBE (Tr. 6591-92).

95. The Staff's witness on this subject was Mr. Robert Dodds of NRC's Office of Inspection and Enforcement, who had conducted the NRC inspection in a previous instance when an earthquake affected a licensed nuclear plant (Dodds, NRC Staff Exh. 1, p. 1). He described the NRC's incident response procedures for handling abnormal occurrences in general, which would be applicable to seismic events. These would require activation of Headquarters and Regional response teams based upon the severity of the event and any potential threat to the public. NRC inspection would encompass determinations that the Licensee has taken appropriate actions to assure safe conditions and has obtained any necessary support, as well as review of effects upon the facility, including any breach of protective barriers, damage to equipment or property and any risk of escalation of consequences of the accident. If any significant anomalies are observed, visual inspections would be augmented by additional steps, such as observation of non-destructive testing of piping or structures or functional testing of systems. Additional testing and more detailed structural analysis would be expected as the severity of the event increased. (Ibid., pp. 11-15; Tr. 1669-71, 1674-75, 1708-12).

96. With specific reference to the Plant, the NRC Staff witness confirmed that the Licensee had appropriate emergency

procedures in effect that would be applicable in the event of an earthquake. He also noted that the NRC has had a resident inspector on-site since August 14, 1978, who, in the event of an earthquake exceeding an OBE, would be available to verify the status of the Plant and make other necessary determinations. With the assistance of an additional inspector or inspectors dispatched by NRC, extensive inspections would be undertaken to assure that safe conditions existed and, ultimately, that the Licensee was taking appropriate actions to demonstrate whether the Plant could resume operation. The witness described the types of examinations that would be undertaken of the Complex, including specifically an inspection to ascertain the existence of any deformation or breakage of components or piping, spalling or cracking of concrete, damage to insulation or wiring, and inoperability of any equipment. He explained that, in view of the design deficiencies of the structure, the inspection would be conducted on more equipment, components and portions of the structure than is standard NRC practice. Based on these on-site inspections and examination of the Licensee's evaluations, the Regional Office would make recommendations to Headquarters as to whether resumption of operation should be allowed or additional measures, such as tests, repairs and evaluations, should be completed. (Dodds, NRC Staff Exh. 1, pp. 12-15; Tr. 1683-84, 1715-23, 1726-27).

97. We are satisfied that the NRC Staff procedures for inspection of the Plant following an earthquake and for evaluation

of the actions being taken by the Licensee will assure that safe conditions are maintained at the Plant, that appropriate tests, analyses and repairs are undertaken and that resumption of operation will not be authorized by the NRC until it has undertaken a thorough investigation and analysis of any effects of the earthquake. We also note that the presence at the Plant of an on-site resident NRC inspector, although not essential, will facilitate prompt action, as necessary, by the NRC following a seismic event.

98. Finally, since there is a possibility that preliminary activities relating to the Plant modifications would be undertaken during the period of interim operation, we asked the Licensee to describe its procedures that will assure that such activities would not adversely affect operation of the Plant (Tr. 6593).

99. The testimony of Mr. Broehl, Licensee's Assistant Vice President, Generation Engineering-Construction, first described the Licensee's standard procedures, which require that any requested design change be reviewed to assess its potential impact on the Plant, particularly from the standpoint of Plant safety. These standard procedures provide assurance that no design change or modification is approved unless a determination has been made that it does not require revision of Technical Specifications nor constitute an unreviewed safety question. They provide for Licensee review, at several management levels, of the detailed design drawings, specifications and construction

procedures in order to provide this assurance. In addition to review at these levels, documentation of the design change procedure for each project is prepared and retained in files at the Plant. (Broehl, Licensee Exh. 13, Attachment 1, pp. 1-2).

100. In addition, Licensee has adopted special design review procedures in connection with any proposed modifications related to the Control Building. Any such activity will be reviewed and evaluated to assure that it will not reduce the strength of existing shear walls. This review and evaluation will be conducted by Licensee's Engineering Department and must then be approved by the Plant Review Board and the Nuclear Operations Board. If it is determined in the process of this review that a proposed modification would reduce the strength of the shear walls, that modification will not be made without prior NRC approval. (Broehl, Licensee Exh. 13, Attachment 1, p. 2; Tr. 2045-46).

101. To assure that everyone is fully informed of all relevant matters, and in response to an NRC request, Licensee has agreed to provide advance notification to the Board, the NRC Staff and all parties of any preliminary construction activities relating to the proposed modification of the Control Building. Licensee has also undertaken to provide such advance notification concerning any other construction or maintenance activity relating to the shear walls of the Complex. (Broehl, Licensee Exh. 13, Attachment 1, p. 2). In accordance with such

undertaking, Licensee has periodically provided in the course of this proceeding advance notification of design changes that were being undertaken (Licensee Exhs. 2, 3 and 4).

102. We are satisfied that the Licensee has appropriate procedures in effect to assure that it will properly review any activities that might affect the Plant during interim operation and that NRC approval will be obtained prior to any modifications that would reduce the strength of existing shear walls. We are also satisfied with the procedures adopted and implemented by the Licensee to assure that the Board, the NRC Staff and all parties receive advance notification of any related activities.

VI. POSITION OF INTERVENORS

103. As we have previously discussed, in view of the limited issue involved in determining whether interim operation of the Plant should be permitted, this phase of the hearing has proceeded without a statement of contentions by the intervenors. The intervenors[9/] did not present any witnesses and thus did not directly raise factual issues.[10/] To the extent that they sought to raise factual issues through cross-examination, we have carefully reviewed and evaluated the portions of the record resulting from such cross-examination in reaching our decision. Moreover, in order to assure that our decision reflects that we have fully taken into account the matters raised by all parties, we review and discuss in this section the statements of position expressed by the intervenors and the record developed on these positions.

104. The CEC was the only intervenor represented by counsel. In his opening statement (Tr. 461-68) and in other remarks throughout the hearing, he appeared to make three basic arguments:

9/ We deal separately with the participation of the State of Oregon in Section III.C.7, *supra*.

10/ The Consolidated Intervenors did request that the Board take official notice of a portion of a report prepared by a review group appointed by the NRC, which, according to the Consolidated Intervenors, shows that there are differing views concerning probability of earthquakes than those reflected in the testimony of the NRC Staff (Tr. 2259). Since neither the conclusions of the NRC Staff nor those of the Board concerning interim operation are based upon any findings concerning probability of earthquakes, views within the scientific community on this subject are not relevant to our decision.

(1) When the Plant was originally licensed for operation, its design reflected a certain margin of safety. Since there are admitted deficiencies in the original design, the margins of safety have been reduced. The public is entitled to the originally expected margins and should not be exposed to the risk of reduced margins.

(2) The original designer was Bechtel, which has also conducted the principal reevaluations, calculations and analyses that provide the basic information relied upon by the parties to this proceeding. In view of Bechtel's potential liability for damages arising from the design deficiencies, the information it has provided is not reliable.

(3) In view of the errors in review and supervision that took place at the time of the original design and the lengthy period required to discover such errors, when coupled with other alleged problems at Trojan, the Plant should remain shut down and a complete "safety audit" should be conducted of the Plant "from top to bottom."

105. These arguments both misunderstand the scope of the proceeding and misstate the pertinent facts.

106. There is no doubt that there were deficiencies in the original design which led to reduced margins in the structure's capability to withstand the SSE and the OBE. But this fact is obviously not dispositive. Instead, it results in the need to determine whether the finding made when the operating license was issued (i.e., that there is reasonable assurance that the

activities authorized by the operating license can be conducted without endangering the health and safety of the public) is negated by the design deficiencies. This is precisely the question that has been explored at length throughout this proceeding. The record demonstrates that since, inter alia, the Plant will still withstand the design basis SSE with significant safety margins and appropriate conditions can be applied with respect to a reduced OBE, there continues to be reasonable assurance that the activities authorized by the operating license can be conducted without endangering the health and safety of the public during a period of interim operation.

107. To the extent that CEC's second argument is intended to attack the credibility of Bechtel's witnesses because of any potential Bechtel liability for the deficiencies, we find such argument completely without merit. We observe at the outset that it was Bechtel who discovered and reported the deficiency. This hardly supports a claim of irresponsibility or lack of reliability. To the contrary, it is indicative of a responsible attitude. (Tr. 602-03). Since the contract between Licensee and Bechtel was referred to at length in the cross-examination of the Bechtel witnesses, we admitted it into the record as a Board exhibit (Board Exh. 1). The witnesses had never seen the contract, however, and detailed questioning has convinced us that matters relating to potential Bechtel liability had no influence of any kind upon their professional judgment or their

testimony (Tr. 844-49). Based upon their background and experience, as well as their demeanor and responses to questions, we find their testimony wholly credible and unbiased.

108. To the extent that CEC's second argument is intended additionally to allege that the other parties to the proceeding, including the NRC Staff, have improperly relied upon Bechtel's information rather than performing independent analyses, we again find such argument without merit. The principal NRC Staff witness pointed out that his 34 pages of prepared written testimony concerning pre-STARDYNE analyses (NRC Staff Exh. 5) contained extensive calculations and evaluations he had personally performed, that his supplemental post-STARDYNE testimony (NRC Staff Exh. 6) was also essentially confirmed by these independent calculations and analyses, that the NRC Staff performs as much independent checking and recalculation as is warranted in specific circumstances, and that extensive additional resources within the NRC Staff would have been available for further work if deemed necessary (Tr. 2161-62, 2203 (Herring)). The independence of the NRC Staff's review is further reflected by the differing approach it adopted in determining an OBE level resulting in a reduced OBE of 0.08g. We find that the NRC Staff's analysis was both independent and thorough. We have also noted that the experts retained by Licensee and by Oregon performed independent analyses and evaluations and satisfactorily explained why they were able to reach valid professional judgments without unnecessarily duplicating Bechtel's calculations (¶¶ 68, 74, <u>supra</u>).

109. CEC's third argument, _i.e._, the demand that we require a "safety audit" of the entire Plant clearly constitutes a request beyond the jurisdiction of this Board. As we explained to the intervenors (_e.g._, Tr. 2033-34), the Licensee is the holder of an operating license for the Plant and we are not authorized to reexamine every matter that was explored when the license was issued or to expand the scope of this proceeding beyond matters relating to the design deficencies that gave rise to the notice of hearing. This hearing relates to a proposed amendment to an operating license and, as the Appeal Board has held, the proper scope of such a hearing is limited to matters that have an appropriate nexus to the proposed amendment and arise from such proposal. _Vermont Yankee Nuclear Power Corporation_ (Vermont Yankee Nuclear Power Station), ALAB-245, 8 AEC 873, 875 (1974).

110. We would not want our denial of CEC's request on indisputable jurisdictional grounds, however, to be read as implying that any matters arose during the hearing that would have warranted review if jurisdictional limitations could be overlooked. We gave the intervenors broad latitude in the course of cross-examination to explore many areas which had at best marginal relationships to the matters before us. These included, for example, such disparate matters as additional construction costs which would have been incurred had there been no design deficiencies (Tr. 589); the jurisdiction of the United States Geological Survey (Tr. 2153); Licensee's conduct of radiation

emergency response drills (Tr. 1986-92); problems (unrelated to deficiencies in the walls) incurred in the functioning of the DBA sequencers (Tr. 2002-23); and the Licensee's on-site supervision of construction of the Control Building (Tr. 1848-57). Suffice it to say that none of these cross-examinations have resulted in our becoming aware of any matters which require further exploration by the NRC Staff or in other forums.[11/]

111. CSP (Tr. 479-82, 700) and the Consolidated Intervenors (Tr. 482-91) essentially expressed positions very similar to those of the CEC. Thus, we have addressed above their arguments that the Plant should remain shut down until an independent safety investigation is undertaken, that Bechtel is not a reliable source of information, that the NRC Staff did not do sufficient independent analysis and that conservatisms are being improperly reduced. The Consolidated Intervenors also requested that the Commission provide funding for their participation in this proceeding (e.g., Tr. 492). The Board denied such request as beyong its jurisdiction and contrary to Commission decisions (Tr. 492).

11/ The NRC Staff indicated that it is undertaking a review of whether the changes in floor response spectra arising from use of a more sophisticated finite element analysis, such as STARDYNE, would have generic impact on other plants (Tr. 1794 (Gray)). Insofar as the Plant is concerned, the record indicates that the different results produced by the original and finite element analyses of the Complex are due to its assymetrical arrangement and short, squatty nature. (Tr. 2140-41 (Herring); 2379 (White); Licensee Exh. 19, Further Response to NRC Staff Technical Question of October 16, 1978, p. 6). There thus appears no basis for exploring additional structures at Trojan which do not have such features, even if such exploration were within our jurisdiction.

112. At the second prehearing conference, one of the Consolidated Intervenors raised the question of whether an environmental impact statement was required in connection with the proposed amendment allowing interim operation (Tr. 6552-56). The NRC Staff has concluded that, since interim operation of the Plant will not result in environmental effects or impacts that differ from those previously evaluated at the operating license stage, authorizing such operation is an action falling under 10 CFR § 51.5(d)(4) and preparation of an environmental impact statement or environmental impact appraisal and negative declaration is not necessary (NRC Staff Memorandum of August 15, 1978, NRC Staff Exh. 8; Tr. 6556-58 (Gray); Tr. 2126 (Trammell)). The intervenors did not present any contrary testimony and did not cross-examine the NRC Staff witness on these conclusions.[12/] The Board concurs in the NRC Staff's determination that interim operation does not involve any environmental impacts not previously evaluated and that no environmental impact statement or environmental impact appraisal is required.

12/ A limited appearance statement submitted by Doreen L. Nepom (Tr. 1516) argued that since the amendment to the operating license authorizing interim operation will modify the technical specifications, it must be accompanied by an environmental impact statement as if it involved initial issuance of an operating license. This argument ignores the fact that an amendment to a license does not automatically constitute a major federal action significantly affecting the quality of the environment. In determining whether an environmental impact statement is needed in connection with an amendment, the Commission considers whether the proposed licensing action would bring about significant environmental effects beyond

(Continued next page)

12/ (Cont'd.)

those previously assessed. See Boston Edison Co. (Pilgrim Nuclear Power Station, Unit No. 1), LBP 74-57, 8 AEC 176, 184 (1974); Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 46 n.4 (1978). As noted in the text above, since no such additional adverse impacts are associated with interim operation, neither NEPA nor the Commission's regulations require preparation of an environmental impact statement.

VII. CONCLUSIONS OF LAW

113. The scope of this proceeding has been limited to the issue of whether interim operation of the Plant should be permitted prior to the modifications required by the Order of May 26. We have reviewed all of the evidence submitted by all parties relating to this issue, including the record of the hearings held on October 23-27, October 30-November 3, and December [ ]. We have also considered all of the proposed findings of facts and conclusions of law submitted by the parties. Those proposed findings not adopted in this decision are hereby rejected.

114. Based upon our detailed evaluation of the entire record, we conclude that:

(1) The Trojan Nuclear Plant Facility Operating License No. NPF-1 should be amended in accordance with our order below to permit interim operation, subject to certain conditions set forth in the order;

(2) There is reasonable assurance that the activities authorized by the Facility Operating License, as amended, will be conducted in compliance with the Commission's regulations and will not endanger the health and safety of the public;

(3) The issuance of the amendment to the Facility Operating License will not be inimical to the common defense and security or to the health and safety of the public; and

(4) The issuance of the amendment to the Facility Operating License is not a major Commission action significantly affecting the quality of the human environment and does not require the preparation of an environmental impact statement or an environmental impact appraisal and negative declaration under the National Environmental Policy Act of 1969, as amended, and Part 51 of the Commission's regulations.

VIII. ORDER

115. Wherefore, it is ORDERED, in accordance with the Atomic Energy Act of 1954, as amended, and the regulations of the Nuclear Regulatory Commission, and based on the findings and conclusions set forth herein, that the Director of Nuclear Reactor Regulation is authorized to make appropriate findings in accordance with the Commission's regulations and to issue an appropriate amendment to Facility Operating License No. NPF-1 authorizing interim operation of the Trojan Plant. Such amendment shall include the following provisions and conditions:

(1) Upon the effective date of this amendment to Facility Operating License No. NPF-1 and until issuance of a superseding amendment pursuant to subsequent order of the Atomic Safety and Licensing Board following completion of proceedings to consider the scope and timeliness of proposed modifications to the Trojan Control Building, Facility Operating License No. NPF-1 is modified by waiver of the following portions of Technical Specification 5.7.1:

(a) the requirement that the Control Building meet an OBE capacity of 0.15g using 2 percent damping as required by FSAR Table 3.7.1;

(b) the requirement that the Control Building meet an OBE capability of 0.15g and an SSE capability of 0.25g using a yield strength for reinforcing steel of 40,000 psi in accordance with ASTM minimum values as required by FSAR Section 3.8.1.3.3; and

(c) the requirement that the masonry portions of the Control Building walls meet Uniform Building Code requirements for reinforced grouted masonry as specified in FSAR Section 3.8.1.4.

(2) During the term of the amendment, the facility shall be operated in accordance with the following conditions:

(a) no modification which may reduce the strength of the existing shear walls shall be made without prior NRC approval; and

(b) in the event that an earthquake occurs that exceeds the facility criteria for a 0.08g peak ground acceleration at the plant site, the facility shall be brought to a cold shutdown condition and be inspected to determine the effects, if any, of the earthquake. Operation cannot resume under these circumstances without prior NRC approval.

116. It is further ORDERED, in accordance with 10 CFR §§ 2.760, 2.762, 2.764, 2.785 and 2.786, that this Initial Decision shall be effective immediately and shall constitute the final action of the Commission forty-five (45) days after the issuance thereof, subject to any review pursuant to the above-cited Rules of Practice. Exceptions to this Initial Decision may be filed within ten (10) days after service of this Initial Decision. A brief in support of the exceptions shall be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff).

Within thirty (30) days of the filing and service of the brief of the Appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exceptions.

IT IS SO ORDERED.

THE ATOMIC SAFETY AND
LICENSING BOARD

APPENDIX A

List of Exhibits Admitted in Evidence

No.		Identified	Admitted Into Evidence
Licensee Exhibits			
1.	Protective Agreement signed by Gregory Kafoury on 8/14/78	956	956
2.	Licensee's Statement dated August 14, 1978 - with itemized Request for Design Changes -	956	990
3.	Licensee's letter to Board (Johnson to Board) dated Oct. 10, 1978 - with itemized Request for Design Changes -	958	990
4.	Licensee's letter to Board (Johnson to Board) dated Oct. 23, 1978 - with itemized Request for Design Changes -	958	990
5.	Reserved		
6.	Licensee's letter to NRC (Withers to Engelken) dated May 5, 1978 - with LER 78-13 and attachments -	531	532
7.	Licensee's letter to NRC (Broehl to Schwencer) dated May 24, 1978 - with "Supplemental Information to LER 78-13" -	531	532
8.	"Trojan Control Building Supplemental Structural Evaluation" dated Sept. 19, 1978	531	532
9.	Responses to NRC Staff Questions	531	532
	A. Licensee's letter to NRC (Broehl to Schwencer) dated Aug. 19, 1978 - with attached responses to NRC Staff Questions -		
	B. Licensee's letter to NRC (Goodwin to Schwencer) dated Aug. 21, 1978 - with attached responses to NRC Staff Questions -		
	C. Licensee's letter to NRC (Goodwin to Schwencer) dated Sept. 12, 1978 - with attached correction to information in Exh. 9-B -		

No.		Identified	Admitted Into Evidence
Licensee Exhibits (Cont'd.)			
	D. "Response to Questions From the Nuclear Regulatory Commission Dated August 30, 1978" dated Sept. 20, 1978		
	E. Licensee's letter to NRC (Broehl to Schwencer) dated Oct. 10, 1978 - with attached responses to NRC Staff Questions -		
	F. Licensee's letter to NRC (Goodwin to Schwencer) dated Oct. 13, 1978 - with attached responses to NRC Staff Questions -		
	G. Licensee's letter to NRC (Broehl to Schwencer) dated Oct. 17, 1978 - with attached response to NRC Staff Question of Oct. 16, 1978 -		
10.	Licensee's Testimony of Anderson, et al. - with statements of qualifications -	549	549
11.	"Flexibility Survey Response" dated Sept. 18, 1978	949	953
12.	"Response of Trojan Nuclear Power Plant Control Building to Specified SSE Event" dated Sept. 20, 1978 (Testimony of Professors Holley & Bresler) - with statements of qualifications -	1029	1030
13.	Testimony of Donald J. Broehl - with statement of qualifications -	1808	1808
14.	Statement of qualifications for John L. Frewing	1810	1810
15.	Testimony of S. R. Christensen - with statement of qualifications -	1814	1814
16.	Testimony of Bart D. Withers - with statement of qualifications -	1816	1816
17.	Trojan Final Safety Analysis Report (FSAR) Figures 3.7-1 and 3.7-2	1840	2060

No.	Identified	Admitted Into Evidence
Licensee Exhibits (Cont'd.)		
18. Licensee's letter to Oregon DOE (Williams to Miller) dated April 26, 1978	2114	2115
19. Licensee's letter to NRC (Broehl to Schwencer) dated Oct. 27, 1978 - with further response to NRC Staff Question of Oct. 16, 1978 -	2334	2336
20. Licensee's letter to NRC (Broehl to Schwencer) dated Nov. 2, 1978 - with responses to NRC Staff Questions -	2335	2336
NRC Staff Exhibits		
1. Testimony of Robert T. Dodds - with statement of qualifications -	1624	1624
2. Statement of qualifications for Kenneth S. Herring	2117	2128
3. Statement of qualifications for James E. Knight	2118	2128
4. Statement of qualifications for Charles M. Trammell, III	2119	2128
5. Testimony of Kenneth S. Herring (Herring I)	2122	2128
6. Testimony of Kenneth S. Herring (Herring II)	2124	2128
7. Testimony of James E. Knight	2125	2128
8. NRC Memorandum (Grotenhuis to Schwencer) dated Aug. 15, 1978	2127	2128
Board Exhibit		
1. Contracts between PGE and Bechtel	961	961
State of Oregon Exhibit		
1. Testimony of Harold I. Laursen	2068	2069

No.		Identified	Admitted Into Evidence
Consolidated Intervenors Exhibits			
1.	Not admitted	2004	[Withdrawn at 2021]
2.	State of Oregon, Department of Energy, letter to Licensee (Miller to Williams) dated April 13, 1978	2017	2018