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June 13, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

DOCKETED
USNRC

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC	)	Docket Nos. 50-440 OL
ILLUMINATING CO. ET AL.	)	50-441 OL
	)	
(Perry Nuclear Power Plant,	)	
Units 1 and 2)	)	

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OCRE'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A PARTIAL INITIAL DECISION (ISSUE #8, HYDROGEN CONTROL)

Susan L. Hiatt
OCRE Representative

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(ISSUE #8, HYDROGEN CONTROL)

Pursuant to 10 CFR 2.754(a)(2), Intervenor Ohio Citizens for Responsible Energy ("OCRE") hereby submits in the form of a partial initial decision its proposed findings of fact and conclusions of law relating to Issue #8, on hydrogen control. The proposed findings of fact and conclusions of law follow the form prescribed by the Atomic Safety and Licensing Board ("Board") in its April 18, 1985 Memorandum and Order (Proposed Findings and Conclusions).

I. OPINION

A. HISTORY OF THE CASE

This is the third partial initial decision in this contested proceeding on the application for operating licenses for the Perry Nuclear Power Plant (PNPP"). The first partial initial decision, LBP-83-77, 18 NRC 1365 (1983), dealt with quality assurance; the second, LBP-85-__, __NRC__ (1985), concerning emergency planning and diesel generator reliability, included a

thorough discussion of the history of the case which will not be repeated here.

Briefly, the Applicants, The Cleveland Electric Illuminating Co., Duquesne Light Co., Ohio Edison Co., Pennsylvania Power Co. and The Toledo Edison Co., are seeking licenses to operate two boiling water reactors ("BWRs"), Units 1 and 2, at the Perry site in Lake County, Ohio, located approximately 35 miles northeast of Cleveland on Lake Erie. Their application is opposed by the two intervenors in the case, Ohio Citizens for Responsible Energy ("OCRE") and Sunflower Alliance Inc. et al., on a number of grounds. This partial initial decision concerns the last such issue, hydrogen control, to be considered in this proceeding. This issue ("Issue #8") was the subject of an evidentiary hearing held April 30 through May 3, 1985 in Perry, Ohio.

The decisional record of the proceeding on Issue No. 8 consists of the testimony and exhibits filed by the parties, and the other evidence contained in the transcripts of the hearing. Appendix A to this decision identifies the location of written testimony in the transcript. Appendix B lists the exhibits identified, indicates the Board's ruling on any offer of an exhibit into evidence, and identifies the location of admitted exhibits in the transcript.

In preparing our decision, we reviewed and considered the entire record and the proposed findings of fact and conclusions of law submitted by the parties. Those proposed findings and conclusions that are not incorporated directly or by inference

in this partial initial decision are rejected as being unsupported by the record of the case or as being unnecessary to the rendering of this decision.

This Board's jurisdiction is limited to a determination of findings of fact and conclusions of law on matters put into controversy by the parties to the proceeding or found by the Board to involve a serious safety, environmental, or common defense and security question, 10 CFR 2.760a. The Board has made no such additional determinations in this case.

B. BACKGROUND OF ISSUE #8

Issue No. 8 concerns the ability of the PNPP Mark III containment design (and equipment therein) to withstand the effects of the combustion of hydrogen gas, produced from the reaction of fuel rod cladding with steam during a degraded core accident.

Both OCRE and Sunflower originally submitted hydrogen control contentions. The Board denied admission of the contentions because they did not meet the standard for the admission of hydrogen control contentions articulated in Metropolitan Edison Co. (Three Mile Island, Unit 1), CLI-80-16, 11 NRC 674 (1980). See Special Prehearing Conference Memorandum and Order Concerning Party Status, Motions to Dismiss and to Stay, the Admissibility of Contentions, and the Adoption of Special Discovery Procedures, LBP-81-24, 14 NRC 175, 207-08 (1981). CLI-80-16 required that an intervenor wishing to litigate hydrogen control had to first specify a credible

accident scenario entailing hydrogen generation, combustion, containment breach or leakage, and offsite doses exceeding 10 CFR Part 100 guidelines.

Several months later, Sunflower moved to resubmit its hydrogen control contention. On March 3, 1982 we granted that motion, largely due to the publication in the Federal Register of a proposed rule wherein additional protection was to be required in Mark III BWRs (such as Perry) "to provide assurance that large amounts of hydrogen can be safely accomodated . . . " 46 Fed. Reg. 62281, December 23, 1981. See LBP-82-15, 15 NRC 555, 560 (1982). Applicants then moved for directed certification of that decision. Their motion, opposed by the NRC Staff and the intervenors, was denied by the Appeal Board. ALAB-675, 15 NRC 1105 (1982).

In LBP-82-15, we reworded the hydrogen control contention submitted by Sunflower to read:

> Applicant has not demonstrated that the manual operation of two recombiners in each of its Perry units is adequate to assure that large amounts of hydrogen can be safely accomodated without a rupture of the containment and a release of substantial quantities of radioactivity into the environment.

(We subsequently designated OCRE as the lead intervenor on this issue. See September 17, 1982 Memorandum and Order (Concerning Procedural Motions).)

Discovery on Issue #8 closed September 30, 1982. It included one set of interrogatories to Applicants by OCRE, one set of interrogatories to the NRC Staff by OCRE (the responses to which we had to compel), and a set of interrogatories and request for production of documents to OCRE from Applicants.

Discovery marked the onset of a two-year dispute between Staff and Applicants, on one hand, and OCRE, on the other, concerning the scope of the issue and the need for the specification of a credible accident scenario, pursuant to CLI-80-16. The tortuous history of Issue #8 is outlined more fully in our March 14, 1985 Memorandum and Order (Motions on Hydrogen Control Contention).

OCRE filed a motion to reopen discovery on Issue #8 and several other issues on November 15, 1983. We denied this motion, but suggested that Applicants update their previous discovery responses. December 20, 1983 Memorandum and Order (OCRE Motion to Reopen Discovery). Applicants did so, and the information supplied therewith prompted OCRE to move for the reopening of discovery on July 30, 1984 and simultaneously to file another set of interrogatories to Applicants. Applicants voluntarily responded to some of the interrogatories and provided some of the documents requested.

The Commission adopted the new hydrogen control rule for BWRs with Mark III containments (and PWRs with ice condenser containments) on January 17, 1985. 50 Fed. Reg. 3498. The rule requires the installation of systems at such facilities capable of safely handling the amount of hydrogen generated from a 75% metal-water reaction. On January 22, 1985 OCRE filed a motion seeking the rewording of Issue #8 so as to conform its language with that of the new rule. On January 28, 1985, the Staff moved for summary disposition of Issue #8, based on its narrow wording; the contention's wording focused on recombiner adequacy, while Applicants are using an igniter system for degraded core

hydrogen control. OCRE's motion was opposed by Staff and

Applicants, while the Staff's motion was supported by Applicants and opposed by OCRE.

In our March 14, 1985 Memorandum and Order we granted OCRE's motion to reword Issue #8; Issue #8 now reads:

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

In our view, this wording basically alleges that Applicants' hydrogen control system does not conform to the new regulatory requirements of 10 CFR 50.44. Our action rendered the Staff's motion moot.

On March 15, 1985 OCRE filed a motion for a continuance of the hearing on Issue #8 from April 9 to June 3, alleging that additional information was forthcoming on the issue, that there was insufficient time to prepare for a hearing on two complex technical issues [this issue and Issue #16, on diesel generators], the unavailability of unnamed witness(es), and the lack of harm to other parties ensuing from the proposed extension. In response to our request, OCRE indicated that it would not proffer witnesses on Issue #8; Applicants and OCRE then agreed upon April 30 as the hearing date for Issue #8. In light of this agreement, and the fact that the Staff's SER on this issue would not be available until mid-April, we continued the hearing until April 30. See March 29, 1985 Memorandum and Order (Motions for Continuance of Hearing and to Compel Appeapance of NRC Witness).

On March 18, 1985 OCRE moved to compel the appearance of Dr.

Marshall Berman of the Sandia National Laboratory. OCRE alleged that there existed a genuine scientific disagreement between Dr. Berman and the Staff on the basis of statements in the Grand Gulf SER on the adequacy of the hydrogen igniter system. Affidavits filed by Dr. Berman and by Allen Notafrancesco, a Staff witness, in response to OCRE's motion indicated that a disagreement did not exist, and that the Staff witnesses were familiar with the Sandia research program on hydrogen control. Thus, the exceptional circumstances necessary to compel the appearance of a particularly-named Staff witness were not present, and we

denied OCRE's motion. (See 10 CFR 2.720(h)(2)(i).) March 29, 1985 Memorandum and Order.

Applicants' witnesses at the hearing were Eileen M. Buzzelli, John D. Richardson, Kevin W. Holtzclaw, Roger W. Alley, Dr. Bernard Lewis, Bela Karlovitz, and Dr. G. Martin Fuls. Ms. Buzzelli is a senior licensing engineer with CEI, and is responsible for licensing issues relating to the PNPP hydrogen control system and for the preparation of the preliminary evaluation, submitted pursuant to the new provisions of 10 CFR 50.44.

Mr. Richardson is employed by Enercon Services and serves as a consultant to CEI on the hydrogen control issue. He also was employed by Mississippi Power and Light Co., the Grand Gulf licensee, and handled all licensing and technical issues relating to the Grand Gulf hydrogen igniter system.

Mr. Holtzclaw is a licensing engineer with General Electric Co. He has managed the GE Severe Accident Program, which has

included the the BWR/6 Standard Plant Probabilistic Risk Assessment and associated evaluations, including hydrogen event risks.

Mr. Alley is employed by Gilbert/Commonwealth, Inc, the PNPP Architect-Engineer, where he is the manager of the Structural Nuclear Engineering Section. Mr. Alley was involved in developing the original ASME containment design specification for PNPP and has been responsible for finite element and stress analyses of the PNPP containment and drywell.

Dr. Lewis and Mr. Karlovitz are founders of Combustion and Explosives Research, Inc. ("Combex"), a consulting firm providing services to government, research institutes, and industry on the fundamentals of combustion, flames, ignition, and explosions of gases, liquids, and solids. Dr. Lewis and Mr. Karlovitz have extensive experience in the field of combustion and have authored many papers on this subject.

Dr. Fuls has a PhD in mechanical engineering and developed the CLASIX-3 computer program, used to evaluate the consequences of hydrogen burning in the PNPP containment.

The NRC Staff's witnesses were Dr. William Trevor Pratt, Allen Notafrancesco, Li Yang, and Hukam Garg. Dr. Pratt is the Principal Investigator of the Accident Analysis Group of the Brookhaven National Laboratory and is familiar with the MARCH computer code. He has conducted a number of studies on core meltdown accidents and reviews of probabilistic risk assessments, including the GESSAR-II PRA. Dr. Pratt holds a PhD in mechanical engineering and has authored or co-authored many

publications.

Mr. Notafrancesco is a Containment Systems Engineer in the NRC's Containment Systems Branch. He is the lead staff reviewer of the PNPP hydrogen igniter system.

Mr. Yang is a Structural Engineer in the Structural and Geotechnical Engineering Branch of the NRC. He is responsible for review of seismic and structural subjects in the Perry Final Safety Analysis Report and has contributed to the Perry SER.

Mr. Garg is employed as an electrical engineer in the Equipment Qualification Branch of the NRC's Division of Engineering. His responsibilities include review of licensee programs for environmental qualification of safety-related equipment and of equipment survivability during and after a hydrogen burn event.

OCRE presented no direct testimony, but cross-examined both Staff's and Applicants' witnesses. In response to this cross-examination, Applicants presented a rebuttal witness, James H. Wilcox, a CEI welding engineer.

C. APPLICABLE STANDARDS

The regulation governing our disposition of Issue #8 is the Commission's new hydrogen control rule, 10 CFR 50.44(c)(3)(iv)-(vii). This final rule was published in the Federal Register on January 25, 1985 (50 Fed. Reg. 3498).

This new rule requires that BWRs with Mark III containments and PWRs with ice condenser containments implement hydrogen control systems capable of handling large amounts of hydrogen, as would result from a Three Mile Island-type accident.

At the outset it must be stressed that compliance with the rule cannot be avoided by arguing that a degraded core accident

resulting in the generation of substantial quantities of hydrogen is unlikely to occur, as Applicants have done (Applicants' Testimony at 10-16). Such an argument is in effect an impermissible challenge to the rule. The commission considered (and rejected) comments alleging that degraded core accidents are unlikely during the rulemaking process. 50 FR 3499.

10 CFR 50.44(c)(3)(iv)(A) requires Mark III plants to have a hydrogen control system capable of handling the amount of hydrogen generated from a 75% metal water reaction without loss of containment integrity. This system must be supported by a suitable program of experiment and analysis.

10 CFR 50.44(c)(3)(vi) requires that the analysis (B)(1) evaluate the consequences of the generation of large amounts of hydrogen (that equivalent to a 75% metal-water reaction), including consideration of the hydrogen control measures; (2) include the period of recovery from degraded core conditions; (3) use accident scenarios describing the behavior of the reactor system during and following a degraded core accident; (4) support the design of the selected hydrogen control system; and show that (5)(i) containment structural integrity is maintained and (5)(ii) necessary equipment will survive the environment created by the burning of hydrogen.

The latter two points are explained more fully in Sections (iv)(B) and (v) of the rule. Section (iv)(B) requires that containment structural integrity be demonstrated by an analytical technique which describes the containment response to the structural loads involved. Two examples of acceptable

analytical techniques are given:
one is an undescribed method including the use of actual material properties with suitable margins to account for uncertainties in modeling, material properties, construction tolerances, etc. The other is a showing that steel containments meet the requirements of the ASME Code, Service Level C Limits, considering pressure and dead load alone.

Section (v) requires that systems and components needed to establish and maintain safe shutdown and to maintain containment integrity must be capable of performing their functions during and after exposure to the environmental conditions created by the burning of the amount of hydrogen resulting from a 75% metal-water reaction, including local detonations, unless such detonations can be shown unlikely to occur.

While the "Supplementary Information" section of the Federal Register notice is not technically part of the regulation, it is part of its "legislative history" which is useful in interpreting this regulation. From that discussion we find that the Commission's policy for preventing excessive radiation dose to the public is best assured by maintaining a leak tight containment. This is to be demonstrated by compliance with Section (iv)(B), which requires a showing of "structural integrity with margin." 50 FR 3500.
The methods specifically outlined in the rule are meant as examples only; other methods may be acceptable, provided that "convincing evidence" is presented regarding their suitability. 50 FR 3501.

Similarly, the discussion at 50 FR 3501 illuminates the

requirements of Section (v) on equipment survivability. Equipment survivability is essentially equipment qualification, except that the margins considered adequate are less that those required for design-basis equipment qualification. There must be adequate evidence, generated by the qualification methods of 10 CFR 50.49(f), that components needed for mitigating the consequences of an accident, for maintaining integrity of the containment pressure boundary, for maintaining the core in a safe condition, and for monitoring the course of an accident are capable of functioning during and following exposure to the environmental conditions created by the postulated accident, including the burning of hydrogen. The discussion describes an acceptable method of analysis; first an acceptable containment thermal analysis must be performed. Then, the thermal response of equipment analyzed thereby should be compared to the thermal response of equipment during qualification testing. The qualification response should envelope the hydrogen burn thermal response. Selected tests may also be necessary to assure survivability.

The operation of the hydrogen control system must not further aggravate the course of the accident or endanger the plant during normal operations. 50 FR 3500.

Further guidance from the "Supplementary Information" section will be used in our determination of compliance with the rule, infra.

Section (vii) of the rule contains an unusual implementation schedule. Each applicant is required by Section (vii)(A) to

submit by June 25, 1985 a schedule for meeting the requirements of the rule. Section (vii)(B) requires OL applicants to comply with Section (iv)(A) prior to exceeding 5% power; however, complete final analyses are not required if an acceptable preliminary analysis has been submitted. The rule gives no criteria for determining what shall be addressed in the preliminary analysis and what may be deferred to the final analysis, although it is the Commission's objective that the rule be complied with without undue delay. Section (vii)(D).

We reserved our judgement regarding this matter until the evidence was received. Our conclusions on the adequacy of the scope of the preliminary analysis submitted by Applicants are presented below.

A final word is needed. Section (vii)(B) and other parts of the rule ((iv)(B) and (vi)(B)(3)) appear, if taken literally, to give the Staff the sole power to determine whether compliance with the rule has been achieved. The testimony presented by Staff and Applicants, not surprisingly, endorses this interpretation, I.e., they would have us summarily affirm the Staff's findings. We cannot accept this interpretation; were we to do so, we would be abdicating our responsibility as judges. The NRC Staff is not the trier of fact in this proceeding; we are. The Staff is just another party, the views of which cannot be given any more weight than those of the other parties. Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-138, 6 AEC 520, 532 (1973); Consolidated Edison (Indian Point Nuclear Generating Station, Units 2 & 3), ALAB-

304, 3 NRC 1, 6 (1976); Southern California Edison (San Onofre Nuclear Generating Station, Units 2 & 3), ALAB-268, 1 NRC 383, 389 (1975).

Thus, we must subject the Staff's position to the same scrutiny as that of the other parties, and reject it if necessary.

D. COMPLIANCE WITH THE RULE

1. Operation of the System

We will confine our attention to the hydrogen control system used for degraded core events, the distributed igniter system. Finding 4. We note that another hydrogen control system, the combustible gas control system, is used for design basis events. Finding 3. Because Issue #8 concerns the degree of compliance with the Commission's new rule on degraded core hydrogen control (Finding 2), we will not consider the design basis system further.

The distributed igniter system consists of glow plug igniters distributed throughout the containment, wetwell, and drywell. Finding 4. The system is powered by diesel generator-backed AC Class 1E power systems. Id. The system is designed to burn off the hydrogen resulting from a degraded core accident at low concentrations, so as to prevent accumulation of more dangerous, higher concentrations. Id.

The system is to be manually actuated, in accordance with PNPP emergency instructions, when the reactor water level reaches the top of the active fuel. Finding 6. However, these instructions, and the generic emergency procedure guidelines on which they are based, are still under development. Finding 7.

The vent path for containment venting, to be used for overpressure threats to the containment, has not even been established. Finding 8.

Although Applicants claim that the procedures will be available before PNPP exceeds 5% power (Tr. 3426), we cannot approve procedures which we have not seen and which are yet unfinished. 10 CFR 50.44(c)(3)(vii)(B) requires the hydrogen control system to be installed and operable before exceeding 5% power. (Compare 50 FR 3500.) We interpret this section to mean that the procedures for operating the system must also be finalized. Applicants apparently share this interpretation.

The record indicates that these procedures are of sufficient complexity to warrant our thorough evaluation. For example, an analysis of containment hydrogen concentration is necessary in those situations (such as a station blackout degraded core accident) where hydrogen concentration may be high. Finding 13. Potential radiological releases are evaluated before containment venting is done. Tr. 3443 (Richardson). We note that Applicants' design basis offsite dose calculations are in a state of flux (Tr. 3265-70, 3600, 3652-57 (Buzzelli)), and the flow path and rates for containment venting are undetermined. Finding 8.

We therefore have no way of assuring that the procedures used will be appropriate, that instruments relied upon by the operators are available or reliable under accident conditions (compare OCRE Ex. 21 at 196), or that 10 CFR 100 guidelines will be met for containment venting. We will require that the finalized procedures be provided for our scrutiny (and that of

the parties) and approval before PNPP operates above 5% power.

2. Scenarios

10 CFR 50.44(c)(3)(vi)(B)(3) requires Applicants to use in their analysis accident scenarios that describe the behavior of the reactor system during and following a degraded core accident. Applicants in their preliminary analysis evaluated two scenarios, a small break in the drywell with extended ECCS failure ("DWB"), and a transient with a stuck-open relief valve with extended ECCS failure ("SORV"). Finding 9. Recovery was assumed to occur at the point of a 75% metal-water reaction. Finding 10.

Both scenarios used identical steam and hydrogen release histories calculated by the MARCH code. Finding 11. The steam and hydrogen were assumed to be released to the drywell as well as to the suppression pool in the DWB case. Id. The MARCH analysis did not model recovery, but was modified to yield a 75% metal-water reaction. Id.

Since the Commission considers the use of a single release history, with variation of key parameters, to be acceptable (50 FR 3502), we consider Applicants' approach with respect to the release histories for these two scenarios to be acceptable. A 75% metal-water reaction was assumed, as required by the rule. We are aware of the uncertainties associated with the MARCH code (OCRE Ex. 21 at 11, 18; Notafrancesco Testimony II at 4-5); however, since the analysis did use an amount of hydrogen corresponding to a 75% metal-water reaction, and it is not apparent that better analytical tools exist (compare OCRE Ex. 21 at 11), we find the use of MARCH acceptable.

Our findings on other aspects of these 2 scenarios, such as those used in the containment response analysis, are presented in that section of our opinion, infra.

We are concerned that the scenarios chosen do not represent the most severe challenge to containment from hydrogen combustion, as is required to be considered. 50 FR 3502. The station blackout degraded core accident is such a scenario. The igniters would not be available during a station blackout accident. Finding 12. The hydrogen would accumulate; a 75% metal-water reaction would result in a concentration of 28% in the containment. Id. A deflagration of this amount of hydrogen (as would result from igniter system actuation when power is restored) would produce high overpressures. Id.

Although Applicants' emergency procedures will have provisions for such situations (Finding 13), we cannot conclude that these provisions (measuring containment hydrogen concentration and containment venting) are acceptable. As discussed above, we cannot accept these unfinished procedures on faith. Even if such procedures could be accepted, we are not convinced that they will prevent deflagrations caused by random ignition sources. We therefore find Applicants' spectrum of scenarios to be deficient; station blackout must be satisfactorily addressed before exceeding 5% power.

At the hearing Applicants objected to any discussion of station blackout, claiming that it is an issue more appropriate for the final analysis and implying that the new rule prohibits its consideration. Tr. 3428-30. We can easily reject their latter objection. The Staff is requiring that station blackout

be considered in Applicants' final analysis, unless there is justification for its exclusion. Notafrancesco Testimony II at 6. We don't see how this fact can be squared with an interpretation that the rule precludes a consideration of station blackout. Moreover, as Applicants admitted (Tr. 3429), the sentence at 50 FR 3502 on which they rely pertains to the Staff's findings at

ice condenser plants. The Commission goes on to state that applicants at different reactors may have to address other scenarios. Furthermore, we do not interpret the Commission's statement that the rule does not require backup power for all types of hydrogen control systems at all affected plants as a prohibition of a finding that backup power is necessary in a particular proceeding.

As for Applicants' first objection, we would first note that a thorough discussion of the issue of preliminary vs. final analyses is given below. We think that this issue is of such importance that it must be addressed now. Indeed, we see no valid reason to defer it to the final analysis. It may even be to Applicants' benefit to consider the issue now, as any design changes we may require might be made more easily now than after power operation.

Nor do we consider Applicants' discussion (Tr. 3609-10) of the low probability of an extended station blackout leading to a degraded core accident (based on an analysis supposedly showing that RCIC would keep the core cooled for 9 hours) to be dispositive. First, Applicants' witness Holtzclaw was unsure whether analysis assumed that the RCIC system took suction first

from the suppression pool or the condensate storage tank. Tr. 3660. On redirect he said that he believed that RCIC took suction first from the suppression pool and then from the condensate storage tank. Tr. 3664.

This method of operation is contrary to that normally occurring. Normally RCIC takes suction from the condensate storage tank, and automatically switches to the suppression pool on low CST water level. NUREG-0887, PNPP SER, May 1982, at 7-25 to 7-27. It thus appears that the method of operation claimed by Applicants would involve override of automatic functions. This fact, along with the uncertainty expressed by Applicants' witness, and the unavailability of the analysis for our scrutiny (and that of other parties) leads us to reject Applicants' claim. We believe this is consistent with the Commission's finding that the new hydrogen rule is necessary to cope with unexpected events,

regardless of the likelihood of hydrogen generation therefrom. 50 FR 3499.

We therefore conclude that station blackout must be analyzed for degraded core hydrogen control, with our approval, before PNPP exceeds 5% power.

3. Containment Integrity

(a) Containment Vessel Capacity

(i) Positive Internal Pressure

Applicants have performed analyses of the PNPP Mark III containment vessel to determine its ultimate internal pressure capacity. Finding 14. Their analyses included determination of static shell capacity, buckling analyses, and evaluations of

penetrations according to Service Limits C and D of the ASME code. Findings 15, 16, 19; Applicants' Testimony at 24-28.

The Staff requested that Applicants perform an analysis of the PNPP containment demonstrating that the requirements of the ASME Code, Service Level C, of Subarticle NE-3220 are met for an internal pressure of 45 psig (and considering dead load). Finding 17. This is essentially the same as the "acceptable method" of 10 CFR 50.44(c)(3)(iv)(B).

Applicants performed this analysis; they believe that the most limiting penetration (P414) has a pressure capacity of 50 psig. Applicants' Testimony at 28. However, we note that dead load, which the Staff and the rule require to be considered, was neglected for penetrations other than the personnel airlock and equipment hatch. Finding 20. Dead load contributes a maximum of 4.7% of the stress intensity contributed by the internal pressure. Id.

Applicants also neglected the effect of elevated temperatures due to hydrogen combustion on material properties. Finding 24. Applicants admit that this could decrease material strength by 16%. Id.

Applicants also neglected the effects on stresses resulting from the as-built, out-of-tolerance conditions of the PNPP containment vessels. Finding 25. Analyses at the design pressure, 15 psig, showed stress increases from 4 to 20 percent. Id.

Applicants have defended their neglect of these items by claiming that they are insignificant. See, e.g., Applicants' Ex. 8-4 at 16 (dead load); Alley, Tr. 3586-87 (elevated

temperature, claimed to be insignificant because there is sufficient margin between the minimum properties used and the actual material properties of the limiting penetration, P414); Alley, Tr. 3596 (out-of-tolerance condition, said to be insignificant as the vessel shell capacity is greater than that of the limiting penetration).

We disagree. We note that containment capacity is an important factor in determining the adequacy of the hydrogen control system, so much so that an increase or decrease of 5-10 psi can be significant. OCRE Ex. 21 at 9-10. If we make the simplistic assumption that the above-noted effects are additive, the containment capacity could decrease by 18.7 to 34.7 percent, I.e., instead of 50 psig, the capacity would be 32.6 to 40.6 psig.

We also disagree with Applicants' rationalizations of these effects. The hydrogen rule requires dead load to be considered, regardless of its magnitude. Applicants' assertion that the out-of-tolerance condition is insignificant is predicated on the assumption that only the vessel shell, and not the penetrations, will be affected thereby. This assumption is unsupported by evidence, as no calculations were performed to consider higher internal pressures. Finding 25.

Applicants' argument that there is sufficient margin in the material properties used to account for the effects of elevated temperature also fails. An examination of Applicants' Ex. 8-4 shows that P414 is not the limiting penetration; P205 is. For this penetration, the stresses exceeded ASME Code allowables, assuming minimum specified yield strength. Applicants' Ex. 8-4

at Table 10. Applicants therefore took credit for the actual material properties, which allowed a pressure limit of 54.8 psig. Id.; Finding 20. Had P205 been treated the same as all other penetrations, the limiting pressure would be 57000/63500 x 45 or 40.4 psig, using the stresses and factoring method of Table 10 of Applicants' Ex. 8-4.

A decrease of 10% in strength due to hydrogen burn temperatures for P205 would result in a pressure capacity of 49.3 psig, using the actual material strength. While this is only slightly less than the pressure for P414, 49.7 psig (Finding 20), it illustrates that the use of actual material properties for P205 results in no margin to account for uncertainties in material properties, modeling techniques, or construction tolerances, as required by 10 CFR 50.44(c)(3)(iv)(B) (contrary to Applicants' assertion at Tr. 3589-90).

Applicants' analysis did not consider the effect of defective containment welds. The PNPP containment vessels contain welds deviating from ASME code requirements. Finding 26. A fracture and fatigue analysis was performed, for design basis loadings, by Aptech Engineering Services which indicated that the defects would not propagate for the loads considered. Finding 27, 33.

We believe that hydrogen combustion loads (as determined from an appropriate containment response analysis) should be evaluated for their effect on these defects. We realize that the Aptech analysis contains conservatisms, such as the temperature assumed for determining fracture toughness. Finding 33; Tr. 3590-91. However, we note that the minimum ratio of fracture toughness to applied stress intensity factor is 1.34.

Finding 33. I.e. a 34% increase in stress would result in fracture initiation. (According to the principles of linear elastic fracture mechanics, fracture results when the applied stress intensity factor equals or exceeds the fracture t' ghness. OCRE Ex. 13 at 2-4.)

We also believe that conservatism is necessary, as the rule requires that there be structural integrity with margin. We also are concerned that the technique used by Aptech to determine flaw size may not be sufficiently accurate. We note the importance of flaw size in the fracture mechanics evaluation. Finding 30. The Staff considers the digital radiographic enhancement technique used by Aptech to be acceptable only when the radiographs used are of good quality, and that it cannot be considered a standard technique without further demonstrations of accuracy. Finding 29. The PNPP weld radiographs in question are of poor quality. Finding 26. The welds are now inaccessible for repair or re-radiography. OCRE Ex. 18 at 1. We would not characterize the "results of calibration" of the digital enhancement technique given in Table 2, p. 11 of Appendix B of OCRE Ex. 13, showing that the technique can be from 2 to 90% off, as a demonstration of accuracy. Given these uncertainties, and the severe consequences of fracture initiation (Findings 31, 32), we believe that an evaluation of containment capacity must include a thorough assessment of these defects for hydrogen combustion loads.

At the hearing Applicants claimed that they had evaluated the stresses at these welds for 50 psig and found that the

stresses at the higher internal pressure were less than at 15 psig, as analyzed by Aptech. Alley, Tr. 3306, 3313. We can give this testimony no weight because the calculations are not available for the scrutiny of the Board and parties (Tr. 3315), because the claimed stress reduction seems preposterous, and because we have serious problems with Mr. Alley's credibility. For example, Mr. Alley stated that the stresses in the containment vessel are less than those analyzed by Aptech, due to the addition of the annulus concrete. Tr. 3313-15. Mr. Alley later stated that the Aptech analysis did not credit for the annulus concrete (Tr. 3593), which is contradicted by OCRE Ex. 18, a Staff summary of a meeting on the subject which Mr. Alley attended. Mr. Alley claimed to be unaware of the problem with the surface roughness of the welds (Tr. 3324), again discused at the meeting summarized by OCRE Ex. 18. Rather than admit that the digital enhancement technique could be inaccurate by a factor of 10, Mr. Alley questioned the accuracy of a table in the Aptech report (Tr. 3323), which he later characterized as an excellent report (Tr. 3341). Mr. Alley's contradictory and evasive responses do not meet our standard of reliable, probative evidence.

(Nor do we consider the Staff's testimony on this matter to meet this standard. Although Mr. Yang stated that Applicant's analyses demonstrate the adequacy of the as-built containment pressure capacity (Yang Testimony at 2), his opinion is based on a review of the analyses contained in Applicants' Ex. 8-4, which, as discussed previously, did not consider the as-built condition of the containment. And his statement that the defective welds would not impair the integrity of the

containment (Tr. 3733) was based on an apparently limited review of the Aptech report, which only addressed design basis loads. Finally, we must state that we find the cursory, "rubber-stamp" nature of the

Staff's testimony to fall short of our requirements. The Staff should not support an application. Carolina Power & Light (Shearon Harris Nuclear Power Plant, Units 1-4), LBP-79-19, 10 NRC 107 (1979), modified and affirmed, ALAB-577, 11 NRC 18 (1980).)

As discussed above, the hydrogen rule basically requires that the containment serve as a leak-tight barrier to the release of radiation. Applicants agree that this is the containment's function. Finding 35. We are not convinced that meeting the ASME service level C limits ensures a leak-tight containment.

The PNPP equipment hatch is unseated by increasing internal pressure; the only force opposing this action is the preload on the closure bolts, which is overcome at 21.6 psig. Finding 37. Applicants take credit for the spring-back of the O-ring seals to prevent leakage. Finding 38. However, Applicants neglect the phenomenon of compression set, which increases as a result of thermal and radiation aging. Finding 39.

We cannot accept Applicants' rationalizations of this problem at the hearing. Mr. Alley claimed that maintenance procedures, which he could not identify, would prevent compression set from becoming a problem (Tr. 3278). Mr. Alley, who previously did not know if the hatch O-rings were made of

ethylene propylene, later testified that the O-rings (now naming the specific compound) would not be subject to compression set for the temperatures and pressures expected in a hydrogen burn event (Tr. 3582). Mr. Alley mentioned the Arrhenius equation as a method by which this was determined. But, later he didn't know what synergistic effects are (Tr. 3650). We take official notice of the fact
that the Arrhenius methodology is a method of addressing accelerated aging for the purposes of equipment qualification. See NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" at 16. Page 15 of that document addresses synergistic effects, which should be considered in accelerated aging programs. (See also 10 CFR 50.49(e)(7).) We do not believe that a witness who does not know what synergistic effects are can be considered competent to testify on the effects of elevated temperature on O-ring behavior and aging.
(We likewise reject Mr. Alley's testimony (Tr. 3583) on the smoothness of hatch seating surfaces as lacking in credibility.)

We therefore conclude that equipment hatch leakage (including the effects of compression set) due to elevated temperatures and pressures must be formally considered. We must also have reliable evidence concerning the effects of seal wear and seating surface roughness on seal leakage during degraded core accident conditions.

We likewise are concerned about the ability of the inflatable seals used in the personnel airlocks to resist leakage. Finding 40. They are considered to have poor

resistance against severe accident conditions, and an inflatable seal used n drywell airlocks failed qualification testing. Id.

We are not persuaded by Applicants' arguments that hydrogen combustion conditions are expected to be less severe than design basis conditions, as this expectation is based on the CLASIX 3 code. Tr. 3363, 3623. Our disapproval of the CLASIX 3 code is addressed below.

Finally, we must address the ability of Applicants' analytical techniques to describe the containment response to the structural loads involved. At the hearing, Applicants argued that questioning the validity and applicability of their analytical methods was in effect challenging the rule. Tr. 3384-85. We disagree. The rule does not require the use of the ASME code, but permits other analyses. The evidence in our record indicates that the ASME code may not be the most conservative method for all parts and conditions encountered in the PNPP containment. The ASME method does not consider leakage from the pressure-unseating equipment hatch.' A fracture mechanics analysis is the

appropriate methodology for evaluating defective welds. And the ASME code would allow the dome knuckle to reach 82 psig (57000/31203 x 45, from Table 8 of Applicants' Ex. 8-4, based on minimum specified yield stress), which is above the buckling pressure (determined from methods which may not be conservative) at which shell fracture may occur. Finding 16.

We therefore find no prohibition to questioning the validity of Applicants' analysis. We are particularly concerned that Applicants used finite element techniques in their analyses of

containment vessel penetrations. Finding 21. Experiments conducted by the Sandia National Laboratory to qualify such techniques have shown that they may be nonconservative. Finding 22. Although Applicants claimed that their techniques are applicable for the linear, elastic stress ranges under consideration (Tr. 3393), we are not sure that their analyses are in fact confined to these stress ranges. The ASME code is based on redistribution of stresses. Finding 23. Even the less severe service level A or B loadings may produce allowable stresses exceeding yield and approaching ultimate strength. Id. Applicants' early analyses of penetrations showed that local yielding occurs at 22.3 psig for the main steam penetration. Applicants' Ex. 8-4 at Table 6A; Alley, Tr. 3359. We thus cannot find that the finite element techniques used by Applicants are qualified for the stress ranges considered.

In conclusion, we do not accept Applicants' analysis of containment vessel capacity or their finding that the PNPP containment vessel can withstand an internal pressure of 50 psig. We are not convinced that their analyses demonstrate structural integrity with margin at this pressure, let alone that they properly evaluate containment leakage. (We do not consider Applicants' assertions that their service level D calculations establish margin. Applicants' Testimony at 28. This analysis used actual material strengths, again without margins, for the equipment hatch and personnel airlock and a plastic analysis for P205. Applicants' Ex. 8-4 at 22.)

Until the uncertainties addressed above are resolved, we

must consider the PNPP containment capacity to be indeterminate. We note that the ability of the containment to withstand its design basis pressure has not yet been established. Finding 36. This too must be rectified before we can approve operation above 5% power.

(ii) Negative Internal Pressure

Applicants did not evaluate the negative pressure capacity of the containment, but relied upon a conservative FSAR analysis of vacuum breaker action for design basis events to show that design negative pressure is not exceeded. Finding 34. We would find this acceptable, but for the uncertanty as to the ability of the vacuum breaker to survive hydrogen burn pressures, as discussed below.

(b) Drywell Capacity

Applicants did not perform an analysis of the PNPP drywell, but relied on an analysis of Grand Gulf. Finding 41. The Grand Gulf analysis did not consider the effects of voids in the drywell wall; voids have been found in the PNPP drywell wall. Finding 42. Although the voids have been repaired, and Applicants claim that they are reasonably sure that there are no others in places not inspected for them (Tr. 3417), we cannot accept this without further scrutiny. We cannot be sure that Applicants' evaluation of the voids determined the cause of the problem, which apparently narrowed the inspection process. We are also concerned that there may not be effective nondestructive examination methods for detecting voids in the drywell wall. Alley, Tr. 3416.

The Grand Gulf analysis is not available for our scrutiny or for that of other parties. The Sandia National Laboratory, in its review of the Grand Gulf igniter system, did not advocate taking GGNS calculations at face value, but recommended obtaining independent estimates. OCRE Ex. 21 at 10. Given the importance of drywell integrity (Finding 43), we cannot accept the GGNS analysis on faith either. We find Applicants' approach on this matter to be unacceptable.

4. Containment Response Analysis

We examine herein the adequacy of the containment response code used by Applicants, CLASIX 3 (Finding 45), and assumptions and models used therein. We will concentrate first on the input parameters assumed for the scenarios analyzed, then on the code itself, and finally on variations of these scenarios which may represent a more severe challenge to the containment than was analyzed.

Our assessment relies heavily on the review of the Grand Gulf igniter system performed by the Sandia National Laboratory contained in NUREG/CR-2530, OCRE Ex. 21. Finding 48. We find this document to be the most comprehensive and scholarly evaluation of the igniter system in our record. At the hearing Appliants and Staff tried to discredit this document by claiming that it was out of date and that the HECTR computer model therein was crude and did not model a Mark III containment. Notafrancesco, Tr. 3724. We reject this reasoning for several reasons.[1]

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[1] Not the least of our reasons is the fact that Mr. Notafrancesco is obviously not a combustion expert (Tr. 3672).

We do not doubt that Sandia has been refining its HECTR models; indeed, this is to be expected in any field of scientific endeavor. Our record does not indicate that these later versions predict lower pressures and temperatures than that in OCRE Ex. 21; in fact, a later version of what Mr. Notafrancesco termed a realistic representation of the Mark III containment (Tr. 3738) produced higher peak pressures than in OCRE Ex. 21. Tr. 3741-42. In any event, when dealing with those fields of science that are rapidly changing, a line must be drawn somewhere; we cannot wait for the final version of HECTR before we render our decision. Applicants and Staff have both urged that that line
be drawn at Grand Gulf; i.e., nothing more should be expected of Applicants than was required of Grand Gulf for operation above 5% power. Notafrancesco, Tr. 3743; Applicants' Ex. 8-2 and 8-3. Since OCRE Ex. 21 (and no subsequent HECTR calculations) was relied upon by the Staff in approving power operation for Grand Gulf (Notafrancesco, Tr. 3744-46), we feel that it is appropriate to rely on the document for our decision.

Mr. Notafrancesco's arguments that HECTR overestimates flame speeds and combustion completeness (Tr. 3737) are also without merit. These are input parameters which were set to be identical to CLASIX 3 assumptions in some of runs in OCRE Ex. 21. Findings 80-84. While other HECTR cases we have examined use higher values for flame speed and combustion completeness, they are offset by use of a smaller hydrogen source term and less conservative propagation parameters. Findings 81, 85, 86,

87.

Mr. Notafrancesco's comments about HECTR lacking a drywell (Tr. 3738, 3742) are also unpersuasive. Sandia evaluated this through MARCH sensitivity studies, and found that presence of the drywell might lower peak pressures by less than 10 psi. OCRE Ex. 21 at 148. (We believe this is also verified by HECTR case A-1, which models CLASIX 3 case 5, which is a one-compartment containment model. Peak pressures from HECTR were only about 10 psi greater than CLASIX 3; this suggests that the low CLASIX 3 pressures result from other phenomena, such as compartmentalization, rather than the presence of the drywell. Findings 84, 88.)

Applicants finally tried to use the affidavit of Dr. Marshall Berman of Sandia filed in response to OCRE's motion to compel his appearance at the hearing, discussed above, to object to the use of OCRE Ex. 21. Tr. 3725, 3745. Dr. Berman, due to Staff and Applicant opposition to OCRE's motion, was not available at the hearing for cross-examination. We feel it is unfair and inconsistent for Applicants to object to Dr. Berman's appearance at the hearing and then try to use his statements in an affidavit (which is not part of our hearing record) to buttress their position.

As judges with considerable technical expertise, we are capable of evaluating the document ourselves. We have done so, and find the document competent.[2] Sandia recognized that none of

[2] We find this document to be far more credible than the testimony of Dr. Fuls, who claimed familiarity with the MARCH code (Tr. 3255) but did not recognize that a listing of input values was not for MARCH but for CONTEMPT-LT28, and was about to try to correlate the CONTEMPT listing with the MARCH manual (Tr. 3543-46), and that of Mr. Notafrancesco, who, as noted above, is not a combustion expert and whose testimony we found too protective of Applicants.

the available containment response models are sophisticated enough to predict burn pressures and temperatures with high accuracy. OCRE Ex. 21 at 10. Our purpose in relying upon this document is not to consider the HECTR results as absolutes, but rather to compare the effects of different input and modeling assumptions with CLASIX 3. We find the qualitative insights gained therefrom to be essential to our decision. (For ease of comparison, we have compiled in Appendix C to this decision the major

input parameters and results, in uniform units of measure, for the CLASIX 3 and HECTR runs for PNPP (from Applicants' Ex. 8-1) and GGNS (from OCRE Ex. 21).)

The first input parameter we will examine is ignition limit. The PNPP CLASIX 3 analysis used an ignition limit of 8% hydrogen by volume. Finding 51. Applicants claim that this is conservative. Applicants' Testimony at 46. However, we note that the the flammability limits for hydrogen are geometry-dependent. Id. 8% hydrogen concentration corresponds to the downward propagation limit, although Dr. Lewis later stated that this limit is 8.5-10%. Finding 60. The latter figures compare well with that used by Sandia (9%). OCRE Ex. 21 at 15.

Sandia found that the placement of the igniter assemblies at Grand Gulf close to ceilings (and the spray shield of the housing) would inhibit combustion effectiveness. Finding 60. Sandia felt that ignition would reliably occur at the downard propagation limit, but not at lower concentrations. Id. This has been verified by hydrogen combustion tests at the Nevada Test Site. Finding 61.

(We must reject Dr. Lewis' statements that the test which failed to ignite was due to incomplete mixing (Tr. 3520), and that proximity of the igniter assembly to ceilings has no effect on its effectiveness (Tr. 3523-14) as unsupported by the NTS data. It is obvious that Dr. Lewis has not kept up with the latest research in this field. Tr. 3517-18, 3522, 3627.[3] We also reject his assertions that the Fenwal tests proved the reliability of the ignition system. Tr. 3627. These tests used central ignition in a 3-foot diameter vessel (Karlovitz, Tr. 3639, 3649), which we do not consider applicable to conditions in the Mark III containment.)

Sandia reported that, at Grand Gulf, all but a few igniters are located within 2 feet of ceiling structures. OCRE Ex. 21 at 195. We note that the placement of igniters at Perry is similar to that at Grand Gulf. Applicants' Testimony at 34. OCRE Ex. 16 corroborates this. (We realize that much discussion was held at the hearing on whether this document could be relied upon to assess igniter locations. Although it has been superceded by the list of igniter locations in Applicants' Ex. 8-1 (Tr. 3504), the document provides valuable descriptions of the locations of igniters and of structures in close proximity, as the Staff found important (OCRE Ex. 19 at 2).

Applicants' witness Buzzelli first said that an igniter-by-

3/ This is also apparent from the transcript of the McGuire OL proceeding, in which Dr. Lewis testified. We take official notice of the following portions of that transcript: Tr. 3295-96 (February 26, 1981, where Dr. Lewis insisted that detonations are impossible in hydrogen concentrations less than 18%, contrary to the statements of other experts); Tr. 5055 (March 19, 1981, where Dr. Lewis admitted he was not familiar with the work of Dr. John H.S. Lee until learing of it from other witnesses at the hearing); Tr. 5062 (March 19, 1985, where Dr. Lewis was not familiar with the results of tests of glow plug igniters conducted at Lawrence Livermore National Laboratory).

igniter comparison could be performed to determine whether the exhibit locations were representative of the current locations. Tr. 3506. Then, persons who were not witnesses informed Applicants' counsel that locations with the same elevation, azimuth, and radial dimension may not be identical. Tr. 3512. Ms. Buzzelli subsequently, on redirect, changed her testimony to match counsel's statements. Tr. 3607-08. Later, she admitted that specifying the 3 coordinates identified a precise location. Tr. 3659. We agree with the latter point, and believe the document is accurate enough for our purposes.)

Comparing OCRE Ex. 16 to Applicants' Ex. 8-1 (Table 2.4-1), we find that the following igniters have identical locations and are located close to the identified structures:
1M56-054 through -060, 1M56-098, -100, and -101 (room ceiling); 1M56-067 through -075, 1M56-077 and -078 (underside of polar crane support ring); 1M56-080 through -090 (containment vessel dome).

The following igniters do not have identical locations but are very close to the identified structures:
1M56-0076 (same elevation and radial dimension, but 6 degrees difference azimuthal, underside of crane support ring); 1M56-016 through -022 (all located within 1 foot of location identified as inside face of drywell top slab); 1M56-024 through -030 (all located about 1 foot below the 620' elevation, the HCU floor (see Finding 62)); 1M56-032 and -033 (within a foot of the underside of floor slab for refueling pool); 1M56-040 (within a foot of steam tunnel ceiling); 1M56-042 through -045, -093

through -096, and -099 (within a foot of room ceilings); 1M56-091 (within 2 ft. of room ceiling).

Thus, we find from the information available that 61 of the 102 PNPP igniters are located in close proximity to structures which could inhibit upward flame propagation. (It is not possible to determine this for the remaining igniters.) Sandia concluded that such an arrangement will reliably ignite hydrogen in the 10% range, but not at lower concentrations. OCRE Ex. 21 at 195; Findings 59, 60. We therefore believe that an appropriate, conservative ignition limit for containment response analyses is at the downward propagation limit range of 8.5 to 10%. Finding 60. This is somewhat greater than the value used in CLASIX 3. Finding 51.

The second input parameter we examine is propagation limits. This refers to the concentration of hydrogen needed for flame propagation into an adjacent compartment. OCRE Ex. 21 at 94. The CLASIX 3 model assumes a propagation limit of 8% hydrogen. Finding 51. Sandia considered a more realistic input for propagation limits to be the observed geometry-dependent limits (same as the ignition limits mentioned above). Finding 87. The CLASIX 3 value should be conservative.

The third input parameter is combustion completeness. As is expected, complete combustion results in higher pressures, as more hydrogen is consumed. Findings 79, 89. CLASIX 3 uses a combustion completeness of 85%. Finding 51. However, Nevada Test Site results show complete combustion for hydrogen concentrations above 7.7%. Finding 52. We therefore consider the CLASIX 3 value nonconservative.

The fourth parameter we consider is flame speed. Increasing the flame speed will increase peak burn pressures, as there is less time for heat loss during the burn. Finding 53. CLASIX 3 uses a flame speed of 6 feet/second. Id. Based on their experiments in the VGES tank, the Sandia researchers considered this value to be low by a factor of three or more. Id.; OCRE Ex. 21 at 15. Although Staff witness Notafrancesco claimed that the flame speed based on VGES data was too high (Tr. 3737), we believe that an important factor affecting flame speed has been neglected - the effect of ionizing radiation.

The action of ionizing radiation will increase flame speed. Finding 54. It even appears that ionizing radiation, through the creation of radicals, can initiate a detonation, although the levels of radiation in the containment are too low to achieve this. Lewis, Tr. 3526-27, 3615. (Detonations result in supersonic flame speeds. OCRE Ex. 21 at 187.) Sandia also believed that flame acceleration (to speeds greater than determined in VGES) is likely. OCRE Ex. 21 at 15, 17, 194. Given these uncertainties, we are convinced that a higher flame speed than that used in CLASIX 3 is necessary.

The fifth parameter is containment spray operation. Containment sprays are an important heat transfer mechanism that results in a significant reduction of pressure and temperature during a hydrogen burn. Finding 56; OCRE Ex. 21 at 17. CLASIX 3 assumes that the containment sprays are automatically actuated after the first burn. Finding 55.

The PNPP containment sprays are a subsystem of the RHR system, which also functions to cool the core. Finding 57. The

sprays share common pumps, piping, and electrical systems with an ECCS subsystem. Id. Since a degraded core accident is premised on the assumption of a loss of core cooling, it may not be conservative to assume sprays are available. Id.; OCRE Ex. 19 at 4. The Staff has stated that the sprays should not be considered if their availability is questionable. Finding 58. Although Applicants stated that it may not be inconsistent to assume spray operation in a degraded core accident if a valve failure is postulated (Tr. 3445), we believe that more than a valve

failure is necessary to cause a degraded core accident. We find spray availability to be questionable, and agree with the Staff that sprays should not be considered in Applicants' containment response analysis. CLASIX 3 is nonconservative in this respect.

The final input parameter we will consider is the wetwell spray carryover fraction. This of course assumes spray availability. CLASIX 3 sensitivty studies have shown that decreasing the wetwell spray carryover fraction will increase wetwell temperature. Finding 63. (The wetwell refers to the lower compartment in the containment and is bounded by the suppression pool and the HCU floor. Finding 62.) The PNPP CLASIX 3 model assumes a wetwell spray carryover fraction of 0. 9. Finding 63.

The containment spray headers are located far above the wetwell. Finding 65. The cross-sectional flow areas below the refueling floor in the Perry Mark III containment are quite low. Finding 65. The fractions of total containment cross-sectional area at the various elevations are (assuming total cross-

sectional area to be 11,310 square feet, using an inside containment radius of 60 feet, Applicants' Ex. 8-4 at Figure 1): 689'-6" elevation, 2778/11310 = .246; 664'-7" elevation, .224; 642'-0" elevation, .271; 620'-6" elevation, .168. Thus, assuming spray droplets are not impeded by structures and components at elevations above the HCU floor, the maximum amount of direct spray impingement that could reach the wetwell is about 17%. The remainder of the assumed carryover fraction is composed of sheet flow, which is assumed to have half the heat transfer effectiveness as droplet flow. Finding 64. However, this assumption has no experimental basis. Id. We believe that it is improper to take so much credit for an unverified assumption. This too is nonconservative.

Of all the CLASIX 3 input parameters we have examined, we have found all but the propagation limits to be nonconservative.

We now turn to the adequacy of the CLASIX 3 code itself. We find that the appropriate standard by which to evaluate a computer model is that used by Applicants. This standard requires that the code either be available in the public domain and have a history of proven use or that it be validated by comparison with results of other accepted codes or experimental data. Finding 44. Since CLASIX 3 is a proprietary code (Finding 45), the first test is inapplicable.

Applicants claim that CLASIX 3 has been validated by comparison with results of hydrogen combustion tests at Fenwal and the Nevada Test Site. Finding 46. CLASIX 3 gave conservative predictions for all but one test. Id. However, we do not believe that these tests are similar enough to the Mark III containment to consider the code validated. The Fenwal

tests were conducted in a small (3-foot diameter vessel). Id. The Nevada Test Site vessel contained no compartments. Id.

We do not find it surprising that CLASIX 3 can handle such simple configurations, as the cause of its prediction of low burn pressures is apparently its compartmentalization. Finding 88. We note that the CLASIX 3 case 5 reported in OCRE Ex. 21 gave pressures only about 10 psi less than the HECTR prediction (and about three times higher than those predicted by the usual CLASIX 3 configuration). Finding 84. This case used a one-compartment containment configuration. Id. This indicates that "validation" with such simple test configurations is not indicative of proper code performance for the Mark III containment.

Comparison with other accepted codes is the last alternative. The appropriate codes for comparison are HECTR and CONTEMPT. Findings 49, 50, 73.

The NRC Staff conducted comparisons of CLASIX 3 with CONTEMPT-LT28 and found that CLASIX 3 underpredicts atmospheric temperatures because it overpredicts heat transfer. Finding 73. The Staff found this to be nonconservative and stated that CLASIX 3 should not be used. Id. (Applicants in their findings urged that this issue be deferred to the final analysis. We believe, for reasons discussed below, that it is appropriate to consider it now.)

Comparisons with HECTR show that CLASIX 3 greatly underpredicts peak burn pressures. Finding 80. HECTR case B-1 modeled CLASIX 3 case 1. Finding 81. These cases used a higher preburn pressure, as would occur in the DWB accident from the

added drywell air mass. Finding 79. This was the only HECTR case using the same hydrogen source term as CLASIX 3; all other HECTR runs assumed less hydrogen was released. Finding 81. Ignition and propagation limits were 10%, with 100% complete combustion, 6 ft/sec flame speed, and automatic spray action. CLASIX 3 produced a peak pressure of 11.1 psig; HECTR produced a peak of 67.6 psig. Id.

CLASIX 3 case 2 and HECTR cases B-2 and B-2' used input parameters similar to those in case 1, except the HECTR hydrogen source term is less and the preburn pressure is not elevated. Finding 90. CLASIX 3 predicted a peak pressure of 7.4 psig, while HECTR predicted peaks of 56.5 (B-2) and 59.4 (B-2') psig. Id.

CLASIX 3 case 3 and HECTR case B-3 assume the same input parameters as were used in the PNPP analysis. Finding 82. CLASIX 3 produced a peak pressure of 6.7 psig; HECTR's peak was 41.6 psig. Id. CLASIX 3 case 6 and HECTR case B-4 used the same assumptions, except the sprays were assumed off. Finding 83. CLASIX 3 produced a peak of 10.3 psig; HECTR, 53.4 psig. Id.

Other HECTR runs were conducted with input parameters considered more realistic by Sandia, i.e., flame speed and propagation limits. Finding 85. These runs (with the exception of case B-6, the significance of which is discussed below) gave peak pressures ranging from 57 to 69 psig. Id.

The cause of CLASIX 3's prediction of low burn pressures is its compartmentalization model and heat transfer assumptions. Finding 88. Wetwell burns will give small pressure excursions,

while burns in the containment compartment cause higher pressures. Id. CLASIX 3 overpredicts oxygen transport back into the wetwell, thus avoiding wetwell inerting. Id. In reality, hydrogen would also be pushed into the containment compartment. OCRE Ex. 21 at 93; Notafrancesco, Tr. 3749. I.e., intercompartmental mixing is more rapid than assumed. OCRE Ex. 21 at 93. Better mixing makes global burns, with their attendant high peak pressures, more likely. Id. at 17, 93.

We must conclude that the CLASIX 3 code is unacceptable, as it does not compare favorably with the accepted codes HECTR and CONTEMPT. Indeed, comparison with these codes has identified serious nonconservatisms in the CLASIX 3 approach. CLASIX 3 fails to predict more than minimal pressure rises even assuming conservative input parameters. We therefore concur with the Staff (OCRE Ex. 20) that CLASIX 3 should not be used.

We finally turn our attention to conditions which could present a more severe threat to containment integrity than was previously examined. Combustion in the drywell for the DWB accident only occurred when the transient was extended beyond the end of the hydrogen release. Finding 75. More severe combustion might occur if core reflood was assumed at an earlier time. Finding 77. E.g., recovery at 5500 seconds into the transient would condense the large steam fraction in the drywell, raising the concentrations of hydrogen and oxygen. Id. Action of the drywell vacuum breakers would admit more oxygen and limit the degree of depressurization. Id. A simplified calculation of this

event showed that pressures in the range of 50-55 psi could be

attained, assuming preburn atmospheric pressure. Id. This is more severe than previously calculated for the DWB scenario (see Finding 47) and exceeds design basis pressures. We believe that the effects of early recovery on drywell combustion should be investigated using a reliable analytical model, and special attention should be paid to the resultant wetwell hydrodynamic loads from suppression pool swell (OCRE Ex. 19 at 5).

We are also concerned that the base cases may have been terminated too early. The SORV transient was terminated when the hydrogen release went to zero. Finding 74. At the end of the transient, wetwell hydrogen concentration was 26%; oxygen, 3%; steam, 23%. Id. In the containment, there is 5% hydrogen, 2% oxygen, and steam is 31%. Id. (Similar conditions exist at the end of the DWB transient, terminated after the drywell burn. Finding 75.) Eventually, the steam will condense, raising the hydrogen and oxygen concentrations. OCRE Ex. 21 at 29. The containment vacuum breakers (not modeled by CLASIX 3; Finding 76) will open to equalize pressure and will admit oxygen, like the drywell vacuum breakers in the case just discussed. Finding 34.

The effect of terminating these transients early may be significant. HECTR cases B-6 and B-6' illustrate this. Both cases used identical input parameters, except that case B-6 was terminated at the end of the hydrogen release, as was the CLASIX 3 SORV case. Finding 86. Case B-6' was allowed to continue past that time. Id. Case B-6 produced a peak pressure of 21.6 psig, similar to the results obtained from CLASIX 3. Id. Case B-6' produced a peak pressure of 69 psig. Id. The early termination of these transients may be another factor

contributing to CLASIX 3 nonconservatism.

Sandia stated that a more severe threat to containment integrity would be posed by hydrogen releases entering the containment or wetwell without first traversing the suppression pool. OCRE Ex. 21 at 197-198. The PNPP analyses assume that hydrogen flows through the pool and enters at the bottom of the wetwell for both SORV and DWB cases. Finding 66. Drywell bypass leakage is not modeled in CLASIX 3. Finding 68. The maximum drywell leakage allowed by technical specifications is over 5 times the rated capacity of the drywell purge compressors. Findings 69, 70; Richardson, Tr. 3498. In some conditions, drywell leakage may be great enough to prevent flow through the suppression pool. Finding 72. Applicants admit that leakage of hydrogen through the drywell wall leak paths could occur. Finding 71. They assumed that 14-19% of the total hydrogen could bypass the pool. Id. They also claim that this effect would not affect the operation of the hydrogen control system or their analytical conclusions (Tr. 3500, 3649), but no confirming calculations have been performed. Finding 71.

We are not convinced that the amount of leakage will be limited to 14-19%, as neither we nor the other parties have been able to review the GE calculations. We also believe that leakage of hydrogen to the containment compartment will alter the mixing and concentration assumptions of the containment response analysis, making global burns more likely. The effects of drywell leakage on containment response to a degraded core accident must be thoroughly and formally evaluated, as the Staff

has required. OCRE Ex. 19 at 5; OCRE Ex. 20 at 2.

5. Equipment Survivability

Applicants' evaluation of equipment survivability is based on containment pressure and temperature profiles determined by use of the CLASIX 3 code. Finding 92. Since we found above that the CLASIX 3 code is inadequate and produces unrealistically low pressures and temperatures, we must find any analysis of equipment survivability relying on it to be likewise deficient.

We also have problems with the approach used by Applicants, who would have us summarily affirm their analysis because it is bound by Grand Gulf's, which was accepted by the Staff but is not part of our record. Id. We have no way to confirm that their heat transfer code, HEATING-6, is conservative, or that appropriate modeling assumptions have been used. We must conclude that equipment survivability during and after hydrogen deflagrations has not been proven.

(Though not necessary for our decision, we would state that the apparent survival of electrical cables at the Nevada Test Site, despite burning and cracking of cable insulation (Tr. 3716-17) may not be indicative of survival in an actual application. We have no information on whether the cables were aged to an end-of-life condition, as required in equipment qualification tests by 10 CFR 50.49(e)(5). Equipment qualification and survivability are considered by the Commission to be essentially the same, and 10 CFR 50.49 is referenced by the rule. 50 FR 3501.)

We are concerned that certain components have very low qualification pressures. Finding 93. Applicants' justification

for accepting these is apparently not confirmed by any testing. Id. We find the use of these components, not qualified to the higher pressures we believe wll occur from hydrogen burning, to be unacceptable, especially since the vacuum breakers are relied upon in determining containment negative pressure capacity. Finding 34.

We believe that a showing that essential equipment can survive detonations is unnecessary, as detonations are unlikely to occur, even if detonable concentrations were to form. Finding 94.

Diffusion flames pose another form of thermal threat to equipment. Tests conducted by the Hydrogen Control Owners Group ("HCOG") in a 1/20 scale facility showed that diffusion flames exist when the hydrogen release rates exceed 0.4-0.5 lb/sec. Finding 95. Preliminary results from these tests indicated that diffusion flames resulting from a 75% metal-water reaction will result in unacceptable thermal loading to equipment. Id.

The 1/4 scale tests planned by HCOG will define thermal environments resulting from diffusion flames. However, HCOG will not consider diffusion flames resulting a release history corresponding to a 75% metal-water reaction. Finding 96. The reason for this is that the hydrogen release rates needed for diffusion flames are supposedly not realistic for a 75% metal-water reaction. Richardson, Tr. 3568, 3622-23.

Although we recognize that the 1/4 scale tests will be addressed in the final analysis, we cannot defer this issue. 10 CFR 50.44(c)(3)(v)(B) requires that evaluations of equipment survivability consider a 75% metal-water reaction. Applicants have announced their defiance of this regulation; we cannot

ignore such a blatant violation of the Commission's regulations.

We are not convinced by Applicants' arguments about the hydrogen release rates leading to diffusion flames being unrealistic for a 75% metal-water reaction (as a sustained hydrogen release rate at that level would supposedly lead to an unrecoverable degraded core accident). Richardson, Tr. 3622-23. We believe that this logic challenges the Commission' rule. The Commission specifically states that the 75% metal-water reaction is a limiting case degraded core accident, and that oxiuation beyond that point will be unrecoverable. 50 FR 3499. In the rulemaking process the Commission rejected comments which sought a reduction in the extent of metal water reaction to be considered. Id.

We also find the HCOG approach illogical. If the 1/20 scale tests indicated that equipment survival is a problem for diffusion flames from a 75% metal-water reaction, then the logical approach is to test these conditions in the 1/4 scale facility. Rather than resolving the problem, HCOG appears to be hiding from it.

We therefore rule that Applicants' approach to this issue is illegal. Applicants must commit to evaluating the thermal environment, and equipment response thereto, resulting from diffusion flames using a hydrogen release history resulting from a 75% metal-water reaction as a condition of licensing.

c. Other Effects of System Operation

In this section we consider whether other effects of hydrogen control system operation will aggravate the course of an accident. 50 FR 3500.

(a) Drywell Pool Loads

Sensitivity studies at Grand Gulf using the CLASIX 3 code have predicted violent overflow of the suppression pool into the drywell for some cases and assumptions. Finding 97. Sandia likewise predicted pool surge into the drywell. Id. It is not clear from Mr. Richardson's evasive responses (Tr. 3485-96) whether the effects of these loads on equipment and components within the drywell have been evaluated. In any event, there is no evidence in our record for Perry on these loads (as calculated from an appropriate containment response code) and their effects. This problem must be evaluated. See OCRE Ex. 19 at 5.

(b) Decay Heat Removal

The design basis for the Mark III containment assumes that decay heat is added to the suppression pool and is transfered to the ultimate heat sink by the suppression pool cooling mode of the Residual Heat Removal ("RHR") system. Applicants' Ex. 8-1 at 25-26. In a degraded core accident, hydrogen combustion adds heat to the containment atmosphere, in addition to the decay heat addition to the pool. It is thus appropriate to examine decay heat removal capability in this situation.

The evidence in our record indicates that long term decay heat removal is essential to maintain the core in a safe condition, and that operation of a loop of the RHR system in the pool cooling mode is necessary to ensure decay heat removal. Finding 98.

When containment atmosphere pressures exceed 9 psig, containment sprays, a subsystem of the RHR system, automatically actuate. Finding 99. If pressure remains above 9 psig, the

sprays remain operating; sprays take precedence over other RHR modes. Id. Operator action is necessary to align the system to other modes. Id.

Operation of both spray trains will degrade pool cooling and mixing. Finding 100. Apparently no calculations of pool temperature response with both RHR loops in the spray mode have been performed, at least not on our record. Holtzclaw, Tr. 3470. Although Applicants later claimed that the pool temperature peaks about 4 hours after a design basis accident (while hydrogen burning, as analyzed by Applicants, occurs within 1-2 hours) (Richardson, Tr. 3611), this may not be true for other scenarios not considered by Applicants; nor is it clear that that the design basis calculation did not take credit for RHR pool cooling.

We find that the effects of hydrogen combustion on decay heat removal capability must be evaluated. The effects predicted by Mr. John Humphrey, a former GE containment systems engineer, should receive special consideration. (See Tr. 3479-85.) The effects of pool temperature on makeup systems, including the potential for pump cavitation, must be evaluated for PNPP, considering any locally elevated pool temperature due to the proximity of the suction strainers to safety-relief discharge quenchers. 41 (See OCRE Ex. 15; Richardson, Tr. 3469.)

41 Although Applicants claimed that calculations have been performed which show that pump cavitation is not a problem for high pool temperatures (Tr. 3606), these calculations are not available for our scrutiny (or that of the parties, Tr. 3659-59), nor is it apparent that the effects of taking suction from the vicinity of the safety-relief valve quenchers was considered in that calculation. We cannot accept on faith analyses which we have not seen.

(c) Secondary Fires

Applicants have not evaluated the potential for secondary fires (burning of combustible material in the containment, ignited by hydrogen burning), although Applicants claimed that an analysis for Grand Gulf showed no potential for secondary fires. Richardson, Tr. 3580-81. Their logic is that since the CLASIX 3 containment response results for PNPP are bounded by GGNS, secondary fires won't occur at Perry either. Id.

We must reject this logic, as it relies upon the faulty CLASIX 3 code. Furthermore, we cannot accept on faith analyses performed for Grand Gulf which are not part of our record. The evidence in our record indicates that secondary fires are likely to occur, and will affect the containment pressure and temperature profiles. Finding 101.

7. Scope of the Preliminary Analysis

As discussed previously, the new hydrogen rule permits the final analysis of the hydrogen control system to be deferred if an acceptable preliminary analysis is submitted. But, there is no guidance as what constitutes an acceptable preliminary analysis. Applicants in their proposed findings urge us to interpret the rule such that the preliminary analysis includes only the general requirements of 10 CFR 50.44(c)(3)(iv)(A) and not the more substantive requirements of sections (iv)(B), (v), and (vi), claiming that the language of section (vii) indicates that the Commission meant to exclude these sections from the preliminary analysis. We cannot agree. The Federal Register notice

indicates that the language of section (vii) underwent last-

minute changes. 50 FR 3502. We therefore cannot infer from the wording of that the section what the Commission may have had in mind, as it is likely that its language was not chosen with care.

We believe that the Commission intended a preliminary showing to be made on all facets of the rule; this is consistent with what was required for ice condenser plants, Sequoyah and McGuire, where equipment survivability was demonstrated (50 FR 3501); the analyses for these plants are apparently equivalent to the preliminary analysis under Section (vii) of the rule. Id.

Applicants and Staff agreed on the scope of the preliminary analysis before Applicants submitted the document. Finding 102. Not surprisingly, both Applicants and Staff objected to many of the matters raised by OCRE as beyond what they considered the proper scope of the preliminary analysis.

We find guidance for resolving this issue in section (vii)(D) of the rule. This section lists the factors the Staff is to consider in setting the final schedule for compliance with the rule. These factors are the status of efforts to comply with the requirements, the impact of the requirements on other plant modifications, and the Commission's goal of compliance without undue delay. Thus, the standard is that of reasonableness. We find this to be the correct standard for our purposes.

It is obviously unreasonable to require Applicants to complete their final analysis (now scheduled for completion in mid-1986, Finding 104), including results of experimental

programs now in progress like the 1/4 scale tests, before operation above 5% power. It is also unreasonable for Staff and Applicants to choose only those areas relatively free of controversy for inclusion in the preliminary analysis, deferring the other, uncertain issues to the final analysis. This would be unfair to the intervenors, and is what has essentially happened here. We cannot condone such an action, as it is tantamount to deferring a contested issue to the Staff for resolution, a practice prohibited by the Commission in Consolidated Edison (Indian Point Unit 2), CLI-74-23, 7 AEC 951-52 (1974).

We believe that the areas addressed in Applicants' preliminary analysis must be satisfactorily resolved before operation above 5% power. Their inclusion of these areas indicates that they consider them applicable now rather than later. This means that our findings concerning containment and drywell integrity, containment response code, and deflagration equipment survivability must be adequately addressed as a condition for licensing. Since Applicants intend to have procedures for operating the hydrogen control system before operation above 5% power (Tr. 3426), we find this necessary as well, as discussed above.
The effect of igniter system operation in a degraded core accident on decay heat removal capability is also appropriate to address now, as Applicants admit that containment heat removal is relevant to the analysis of the hydrogen control system. Applicants' Ex. 8-1 at 24.

The Staff's judgement on the appropriate scope of the

preliminary analysis is based on what was required at Grand Gulf for operation above 5% power. Finding 103. The applicable GGNS SER supplement included a discussion of the 1/20 scale tests and comparisons of CLASIX 3 with CONTEMPT-LT28. Id. An earlier SER supplement addressed other matters, such as the Sandia evaluation. Id. We take official notice of the fact that Grand Gulf SSER 3 (NUREG-0831, Supplement 3, July 1982) also covered secondary fires (pp. 22-20 and 22-21). We therefore consider it appropriate to consider these matters now.

It is apparent from OCRE Ex. 19 that Applicants have been aware of most of the issues we have considered above (including drywell pool loads and drywell bypass leakage) since 1982. We feel there has been enough time for Applicants to address these matters. To defer them any longer would contradict the Commission's goal of prompt compliance with 10 CFR 50.44 (c)(3)(iv)-(vii).

With the exception of diffusion flames to be studied in the 1/4 scale tests, we find no valid reason to defer any of the matters we have addressed herein. We require a commitment from Applicants to address diffusion flames resulting from a 75% metal-water reaction in compliance with the rule, as discussed earlier, but for all other matters the deficiencies we have noted must be corrected before operation above 5% power.

II. FINDINGS OF FACT

1. Issue #8 was admitted as a contention in this proceeding by Memorandum and Order (Concerning Late-Filed Contentions; Quality Assurance, Hydrogen Explosion, and Need for Increased Safety of Control System Equipment), LBP-82-15, 15 NRC 555 (1982). Issue

#8 was subsequently reworded by Memorandum and Order (Motions on Hydrogen Control Contention), LBP-85-__, 21 NRC ___ (March 14, 1985). OCRE is the lead intervenor on the issue and the only intervenor which participated in its litigation.

2. Issue #8, as litigated, reads as follows:

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

This wording basically alleges that the hydrogen control system does not comply with 10 CFR 50.44(c)(3)(iv)-(vii). March 14 Memorandum and Order at 7.

3. The PNPP hydrogen control system used for design basis events is the combustible gas control system. Applicants' Ex. 8-1 at 3; Applicants' Testimony at 35-37. This system consists of the hydrogen mixing or drywell purge subsystem; two 100% capacity recombiners; the hydrogen analysis subsystem; and a backup containment purge subsystem. Id.

4. The degraded core hydrogen control system is a distributed igniter system. Applicants' Ex. 8-1 at 5-12, 22-25; Staff Ex. 8 at 6-2; Notafrancesco Testimony I at 3-4; Applicants' Testimony at 29-34. This system consists of 102 thermal gl w plug igniters placed throughout the drywell, wetwell, and upper containment. Id. at 32-34. The system is powered from 120 VAC, 60 Hz, Class 1E power distribution panels capable of being powered by the emergency diesel generators. Id. at 32; Applicants' Ex. 8-1 at 9. The system is designed to burn hydrogen at low concentrations (8%), thereby preventing high concentrations of hydrogen, which might ignite randomly and threaten containment integrity.

from accumulating. Id. at 31; Notafrancesco Testimony I at 3-4.

5. The PNPP igniter system is of the same design, and uses the same igniter assemblies and locations, as the Grand Gulf Nuclear Station ("GGNS"). Applicants' Ex. 8-1 at 23; Applicants' Testimony at 34. The PNPP containment and engineered safety system designs are also similar to those at GGNS. Applicants' Ex. 8-1 at Section 5.

6. The igniter system is manually initiated, in accordance with the emergency procedure guidelines, when the reactor water level reaches the top of the active fuel. Applicants' Ex. 8-1 at 12; Applicants' testimony at 34; Buzzelli, Tr. 3424.

7. The PNPP emergency instructions for operation of the igniter system, and the generic emergency procedure guidelines on which they will be based, are currently under development. Buzzelli, Tr. 3425-27.

8. The PNPP emergency instructions will include provisions for venting the containment in the event of overpressure. Richardson, Buzzelli, Tr. 3441-43. The vent path for PNPP (and resultant flow rates) has not been established. Buzzelli, Tr. 3443.

9. The degraded core accident scenarios considered by Applicants in assessing the igniter system are (1) a small break in the drywell ("DWB") with extended ECCS failure, and (2) a transient with a stuck-open relief valve ("SORV") with extended ECCS failure. Applicants' Testimony at 39; Notafrancesco Testimony I at 5.

10. For these scenarios, no ECCS flow is assumed until just prior to reaching a metal-water reaction equivalent to 75% of the active fuel cladding, at which time recovery of coolant

makeup systems is assumed to occur and the transient is terminated. Applicants' Testimony at 38.

11. The hydrogen and steam release rates used for these scenarios were calculated from the MARCH computer code. Applicants' Testimony at 40; Applicants' Ex. 8-1 at 19. Identical mass and energy releases were used for both scenarios; the steam and hydrogen were assumed released to the drywell as well as the suppression pool (through the safety relief valves) for the DWB case. Id. The MARCH analysis was for an unmitigated accident; to model a degraded core accident, the release history was modified such that the maximum hydrogen release rate achieved prior to core slump was held constant until a 75% metal-water reaction was reached. OCRE Ex. 21 at 11; Applicants' Testimony at 40-41; Notafrancesco Testimony II at 6.

12. The igniter system would not be operable during a station blackout degraded core accident. Buzzelli, Tr. 3432. The hydrogen would accumulate in containment. Id. The hydrogen concentration in containment resulting from a 75% metal-water reaction is about 28% by volume. Buzzelli, Tr. 3438. Although a detonable mixture, it would not detonate if power were restored and the igniters actuated. Lewis, Tr. 3439. However, the deflagration of this amount of hydrogen would produce high overpressures of about 50-110 psi. Lewis, Tr. 3440.

13. In such a situation, the emergency procedures would instruct the operator to determine the hydrogen concentration prior to actuating the igniter system. Buzzelli, Tr. 3441. Containment venting might be used in such circumstances.

Buzzelli, Tr. 3442.

14. Applicants performed an analysis of containment ultimate structural capacity for internal pressurization. Applicants' Ex. 8-4; Applicants' Testimony at 25.

15. The analysis determined static containment vessel shell capacity using actual material strengths. Applicants' Ex. 8-4 at 4-7. Values were computed for mean and lower bound values of material properties. Id. at 3. Lower bound properties are computed by subtracting three standard deviations from the mean. Alley, Tr. 3283. Fifty percent of the containment vessel materials are expected to have strengths greater than the mean values, and 50% are expected to have strengths less than the mean, while 99% of the materials are expected to have strengths greater than the lower bound values. Alley, Tr. 3284.

16. The dome knuckle controls the capacity of the containment shell. Applicants' Ex. 8-4 at 6. At 78 psig, hoop buckling occurs in the knuckle (assuming lower bound material properties). Id. At this pressure, waves form periodically around the circumference of the dome. Id. The shell may fracture where the waves appear. Id. The methodologies used for calculating the buckling pressure are based on perfect ellipsoidal shells (containing no residual stresses and geometric imperfections) and have no factors of safety. Alley, Tr. 3347-49.

17. The NRC requested that Applicants perform an analysis of the PNPP containment demonstrating that the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits are satisified for a

loading combination of dead load and internal pressure. The minimum pressure was to be 45 psig. Applicants' Ex. 8-4 at 13.

18. In response to this request Applicants analyzed the containment vessel cylinder and dome and penetrations. Id. The stress intensities at 45 psig for the containment vessel cylinder and dome were less than the allowable stress intensities. Id. at Table 8. The analyses used minimum specificied yield strengths. Id. at 13. The pressure capacity, pursuant to ASME Service Level C limits, for the cylindrical shell is 79 psi. Alley, Tr. 3596-97.

19. Analyses of the personnel airlock and equipment hatch indicated that Service Level C limits would be reached at 50.2 psig and 52.6 psig, respectively. Applicants' Ex. 8-4 at 15.

20. Analyses of the other penetrations neglected dead load, as dead load contributes a maximum of 4.7% of the stress intensity contributed by internal pressure. Id. at 16. The limiting penetration is P414, with a pressure capacity of 49.7 psig, when considering the effect of an adjacent penetration. Id. at 17, Table 10. This calculation is based on minimum specified material properties. Id. P205 has an allowable pressure capacity of 54.8 psig, taking credit for material certifications. Id. at 16, Table 10. The maximum internal pressures to meet Level C limits were calculated by factoring up the stresses to reach the allowable stress. Id. at 15.

21. Applicants used finite element techniques in their analyses of penetrations to meet ASME Level C limits. Id. at 14, Figures 7-14; Alley, Tr. 3406.

22. The Sandia National Laboratory has conducted experiments

involving internal pressurization of scale models of containment vessels for the purpose of qualifying analytical methods. Alley, Tr. 3383. One of the models failed catastrophically at 195 psig internal pressure. Alley, Tr. 3401. Finite element analyses of that model predicted general membrane yielding at 185 psig, leakage at 205 psig, and catastrophic failure at 226 psig. Alley, Tr. 3404-06.

23. The ASME code relies upon a redistribution of stresses caused by material yielding at discontinuities or penetrations. Alley, Tr. 3386-87. At service limit A or B loadings, allowable stresses may exceed yield strengths and may approach ultimate strengths. Id.

24. The material strengths of the containment vessel will be decreased about 10% due to increased temperatures from hydrogen burns. Alley, Tr. 3286. This effect is not included in Applicants' analyses. Id.

25. The PNPP containment vessels, as built, are out-of-tolerance with respect to their specifications. Alley, Tr. 3344. An analysis was performed, for 15 psig internal pressure, to determine the stresses resulting from these as-built dimensions. Alley, Tr. 3345. The analysis showed that circumferential stresses at one point were 20% greater than ideal conditions. Alley, Tr. 3345, 3347. The analysis concluded that the largest increase in circumferential stress was 4.37%, with the largest increase in vertical stress being 9%. Alley, Tr. 3596. The stresses at higher pressures were not evaluated. Alley, Tr. 3347. Geometric imperfections will decrease the shell's resistance to

buckling. Id.

26. The PNPP containment vessels contain welds deviating from ASME code requirements. OCRE Ex. 13 at 1-1 to 1-3; OCRE Ex. 18 at 1. These defective welds are inaccessible for repair. Id. The deficiencies were discovered upon a re-review of weld radiographs. Staff Ex. 6, SSER 4, at 3-2. These radiographs are of poor quality, and the weld surfaces are rough. OCRE Ex. 18 at 1.

27. Applicants commissioned Aptech Engineering Services to perform an evaluation of the rejectable welds. Alley, Tr. 3304. The Aptech evaluation included fracture and fatigue analyses to determine crack growth and fracture performance. OCRE Ex. 13 at 2-1 to 2-9.

28. Only design basis loadings were evaluated by Aptech. Id. at 3-7 to 3-12; Staff Ex. 6 at 3-3. Credit was taken for the annulus concrete. OCRE Ex. 18 at 2. Fatigue crack growth due to multiple hydrogen deflagrations was not considered. Alley, Tr. 3306, 3325.

29. Aptech determined the flaw size of the weld defects by a digital radiographic enhancement technique. OCRE Ex. 13 at 6-12. Approximation of flaw depth by this methodology is a guide or aid in judging flaw depth at this time. Staff Ex. 6 at 3-2. Further demonstrated accuracy of the technique is needed before it can be considered a standard technique for flaw depth measurement. Id. The technique is acceptable as an alternative to repeated radiography provided that the original radiographs

meet minimum specifications as to image quality. Id.

30. The depth of a surface-connected crack is an important factor affecting its propagation; the deeper the crack, the less stress needed to initiate fracture. Alley, Tr. 3326; OCRE Ex. 13 at 2-1.

31. The materials used in the PNPP containment vessels have lower resistance to a propagating fracture than to fracture initiation. Id. at 2-3. Under constant loads, continued crack propogation is expected once a fracture has initiated. Id.

32. The PNPP containment weld flaws are located in horizontal or circumferential welds and are oriented such that longitudinal stresses apply. OCRE Ex. 13 at 3-4, 3-6. If a propagating circumferential crack occurred in the lower containment vessel, the internal pressure would lift the vessel up, to be restrained only by penetrations, attachments, and external structures. Alley, Tr. 3337-38.

33. The Aptech analysis, which included conservatisms as to crack growth rates and fracture toughness values, found that, for the design loads considered, very small crack growth would occur over the life of the plant, and that the final flaw size would not propagate by a fracture mechanism, given a maximum one-time stress loading. OCRE Ex. 13 at 7-1; Staff Ex. 6 at 3-2 to 3-3. The ratio of the fracture toughness, Kic, to the applied stress intensity factor, Ka, ranged from 1.34 to 1.54 for the flaws and load combinations analyzed. OCRE Ex. 13 at 7-2.

34. The negative internal design pressure for the PNPP containment is -0.8 psig. Applicants' Ex. 8-1 at 14. Two 24-inch vacuum relief lines are included in the PNPP design to

assure that negative pressure inside the containment does not exceed the design value. Id. Two additional 24-inch vacuum relief lines are provided for redundancy. Id. FSAR design basis calculations, taking credit for only two of the 4 lines, showed that for conservative assumptions, a negative pressure of -0.72 psig was reached. Id. Analyses of the ultimate negative pressure capacity of the PNPP containment were not performed. Id. at 15.

35. The function of the containment vessel is to serve as a low-leakage barrier to limit fission product escape into the environment in the event of an accident. Alley, Buzzelli, Tr. 3264.

36. The containment structural integrity test, conducted at 17. psig, has not yet been performed. Buzzelli, Tr. 3281. The purpose of this test is to demonstrate the capability of the containment to meet specified pressures and to exceed design pressure. Id. The structural integrity test is a one-time test. Alley, Tr. 3282. 17.25 psig is the highest internal pressure to which the containment will be tested. Buzzelli, Tr. 3282. An integrated leak rate test has not been conducted yet. Buzzelli, Tr. 3281. The integrated leak rate test is conducted at 11.31 psig. Buzzelli, Tr. 3282.

37. The PNPP equipment hatch is pressure unseating. Alley, Tr. 3272. The preload on the hatch closure bolts is the only force rasisting the internal pressure. Alley, Tr. 3273. The original bolt preload was based on an internal pressure of 15 psig, and cannot be increased. Alley, Tr. 3274. The internal pressure at

which the closure bolt preload is overcome is 21.6 psig. Alley, Tr. 3277.

38. Applicants' analyses of the equipment hatch assumed that no leakage would occur at 45 or 52.6 psig as the deflections of the hatch cover were less than the precompression of the O-rings, which are assumed to have sufficient spring-back to prevent leakage. Applicants' Ex. 8-4 at 14-15; Alley, Tr. 3275.

39. Compression set is a phenomenon wherein the O-rings would retain some or all of their precompression when the load is removed; i.e., there is little or no spring-back. Alley, Tr. 3277. Compression set will increase due to thermal and radiation aging. Alley, Tr. 3278.

Roughness or irregularity of the hatch seating surfaces will also cause or increase leakage. Alley, Tr. 3278-79.

40. The PNPP personnel airlocks use inflatable seals to prevent leakage. Alley, Tr. 3362. The Argonne National Laboratory considers inflatable seals to have poor resistance against severe accident conditions. Buzzelli, Tr. 3370. Inflatable seals for the drywell personnel airlocks failed qualification testing at a temperature of 465 degrees Fahrenheit. OCRE Ex. 14 at 1. The seals were redesigned and requalified (to lower temperatures, 330 degrees-F). Id. at 2. The internal seal pressure had to be decreased from 90-120 psig to 60 psig, as increased temperatures would increase the pressure within the seals beyond that to which they were qualified. Id.

41. The ultimate capacity of the drywell was not analyzed for Perry; Applicants relied upon an analysis of the Grand Gulf drywell and upon design similarities between the PNPP and GGNS drywells. Applicants' Ex. 8-1 at 15-17. The ultimate pressure

capacity of the Grand Gulf drywell was found to be +67 psid; the negative pressure capability of the GGNS drywell head was found to be -89 psid. Id. at 17.

42. The GGNS analysis did not evaluate the effects of voids in the concrete drywell wall. Richardson, Tr. 3414. Voids have been found in the PNPP drywell wall. Alley, Tr. 3415. They have been repaired, and areas of the drywell wall containing heavily congested reinforcement (conditions in which the voids had been found) were inspected for other voids. Alley, Tr. 3416-17. Inspection of the entire wall was not performed. Alley, Tr. 3416.

43. Loss of drywell integrity can cause containment failure by steam bypasss of the suppression pool in those accidnets involving a pipe break in the drywell. Holtzclaw, Tr. 3420.

44. Applicants' standards for computer codes is that either they are available in the public domain and have a historyof use which has proven their applicability and validity, or they are used to analyze sample problems (within the range of applicability for the actual problem to be analyzed) which have known solutions from other accepted programs, classical theory, or experimental data. Alley, Tr. 3382.

45. Applicants used the CLASIX 3 computer code to evaluate the temperatures and pressures resulting from hydrogen deflagrations within the containment and drywell. Applicants' Ex. 8-1 at 20-21, Appendix A; Applicants' Testimony at 41-42. CLASIX 3 is not available in the public domain. Fuls, Tr. 3547.

46. CLASIX 3 results have been compared with hydrogen combustion tests conducted at Fenwal and the Nevada Test Site; except for one instance, CLASIX 3 predicted conservative

pressures and temperatures. Fuls, Tr. 3621. The Fenwal tests were conducted in a 3 foot diameter vessel. Karlcvitz, Tr. 3649. The Nevada Test Site vessel was a large open vessel with no compartments. Fuls, Tr. 3662.

47. The Perry CLASIX 3 analysis for the SORV accident scenario indicated that 32 wetwell burns and 2 containment burns would occur, with a peak pressure of 21.2 psig (containment) and a peak temperature of 1762 degrees-F (wetwell). Applicants' Ex. 8-1 at Table 17 of Appendix A. CLASIX 3 results for the DWB accident showed 38 wetwell burns and one drywell and one containment burn. Id. Peak pressure was 19.4 psig (wetwell and containment), and peak temperature was 1763 degrees-F (wetwell). Id.

48. The Sandia National Laboratory conducted an evaluation of the Grand Gulf igniter system, including comparison of CLASIX 3 results with other computer codes, MARCH and HECTR. OCRE Ex. 21; Notafrancesco, Tr. 3688.

49. HECTR is a more advanced containment response code than the MARCH subroutine MACE. Pratt, Tr. 3700. Version 2.0 of MARCH has incorporated certain elements of the HECTR model. Pratt, Tr. 3689.

50. HECTR has been evaluated against Nevada Test Site and was found to give conservative results. Notafrancesco, Tr. 3737.

51. The CLASIX 3 code used hydrogen burn parameters of 8 volume-% for ignition and propagation with 85% combustion completeness. Applicants' Ex. 8-1 at Appendix A, Table 4.

52. Hydrogen combustion tests at the Nevada Test Site showed combustion was virtually complete for hydrogen concentrations greater than 7.7%. Lewis, Tr. 3516.

53. CLASIX 3 used a flame speed of 6 ft/sec. Applicants' Ex. 8-1 at Appendix A, Table 4. Sandia considered this flame speed to be significantly low, by a factor of three or more. OCRE Ex. 21 at 16, 17, 93. Higher flame speeds result in less time for heat transfer, and consequently, higher pressures and temperatures. Id.

54. Ionizing radiation has the effect of promoting accelerated combustion and increasing flame speed. Lewis, Tr. 3528-29.

55. The CLASIX 3 analysis assumes that containment sprays are available during the hydrogen burn, and are assumed to be actuated after the first hydrogen burn. Richardson, Tr. 3444; Applicants' Ex. 8-1 at Appendix A, Table 9.

56. Containment sprays are the dominant containment atmosphere heat transfer mechanism, and spray operation is very important to plant safety for hydrogen burn scenarios. OCRE Ex. 21 at 12, 29.

57. Containment sprays at PNPP are a subsystem of the residual heat removal system; another function of that system is low pressure coolant injection, an ECCS subsystem. Richardson, Tr. 3444; OCRE Ex. 19 at 4. A basic postulate of degraded core accidents is that coolant is unavailable. Id.; Richardson, Tr. 3445; Notafrancesco Testimony II at 3. The containment spray systems share common pumps, piping, and electrical power supplies. Richardson, Tr. 3445.

58. The Staff has stated that sprays should not be considered in Applicants' containment response analysis if their availability is questionable. OCRE Ex. 19 at 4.

59. Numerous igniter assemblies at PNPP are located near

ceilings or under other obstructions. OCRE Ex. 16; Applicants' Ex. 8-1 at Table 2.4-1; Buzelli, Tr. 3502.

60. Sandia found that the design of the igniter housing (the spray shield) and the placement of the igniters close to ceiling structures would inhibit combustion effectiveness, such that the system would reliably ignite hydrogen at the downward propagation limit, but not at lower concentrations. OCRE Ex. 21 at 195-96. The downward propagation limit for hydrogen is 8.5 to 10%. Lewis, Tr. 3514.

61. Hydrogen combustion tests at the Nevada Test Site showed that ignition at lean concentrations (6%) could not be achieved for an igniter location at the top of the vessel. Richardson, Tr. 3627.

62. The CLASIX 3 model for the PNPP Mark III containment uses 3 compartments, drywell, wetwell, and containment. Applicants' Ex. 8-1 at 20. The wetwell consists of the volume bounded by the suppression pool and the HCU floor (620' elevation). Notafrancesco Testimony II at 3, 5. The volume of the containment compartment is 959,388 cubic feet; wetwell volume is 181,626 cubic feet. Applicants' Ex. 8-1 at Appendix A, Table 5.

63. CLASIX 3 sensitivity studies showed that decreasing wetwell spray carryover fraction resulted in higher temperatures from hydrogen burns. Fuls, Tr. 3549, 3550-51. The PNPP CLASIX 3 analysis assumed a wetwell spray carrover fraction of .4669. Fuls, Tr. 3550; Applicants Ex. 8-1 at Appendix A, Table 9.

64. The wetwell spray carryover fraction of .4669 includes sheet flow, which is assumed to have half the heat transfer

effectiveness as droplet flow. Fuls, Tr. 3550; Applicants' Ex. 8-1 at Appendix A, pp. 3-4. There is no experimental basis for this assumption. Fuls, Tr. 3550.

65. The cross-sectional flow area at containment elevation 689'-6" is 2778 square feet; at elevation 664'-7", 2534 square feet; at elevation 642'-0", 3070 square feet; at elevation 620'-6", 1900 square feet. Applicants' Ex. 8-1 at Figures 2.-12 through 2.4-15. The containment spray headers are above these elevations. Fuls, Tr. 3550.

66. The CLASIX 3 analyses assume that hydrogen is released through the suppression pool into the bottom of the wetwell compartment. Richardson, Tr. 3496, OCRE Ex. 21 at 18.

67. A drywell break accident first results in the expulsion of the drywell air mass by steam. Fuls, Richardson, Tr. 3533. The drywell atmosphere will then consist of steam and hydrogen. Fuls, Richardson, Tr. 3534. All of the hydrogen is at first assumed to be released to the drywell until 20 minutes into the transient, when half of the hydrogen is assumed to be released directly to the drywell, with the other half going to the suppression pool through the safety relief valves. Applicants' Ex. 8-1 at Appendix A, p. 1; Richardson, Tr. 3496-97.

68. The steam and hydrogen release into the drywell and the action of the drywell purge compressors are assumed to pressurize the drywell such that the first row of drywell LOCA vents is uncovered and the hydrogen bubbles through the suppression pool. Applicants' Ex. 8-1 at 26; Richardson, Tr. 3497. The CLASIX 3 model assumes bidirectional flow between wetwell and drywell (through the suppression pool), bidirectional flow between wetwell and containment, and

unidirectional flow from containment to drywell. Applicants' Ex. 8-1 at Appendix A, Figure 2.

69. There are two drywell purge compressors each rated at 546 scfm. Richardson, Tr. 3497; Applicants' ex. 8-1 at 26-27.

70. The maximum allowable drywell bypass leakage rate is 5843 scfm at 2.5 psig and 32,645 scfm at 30 psig. Applicants' Ex. 8-1 at Table 5.4-2; Richardson, Tr. 3497. These values are technical specification limits for drywell leakage and are equivalent to 0.168 square feet leakage area. Buzzelli, Tr. 3616. No action is required to be taken if the results of periodic drywell leak testing (conducted at approximately 3 psi) indicate bypass leakage values less than that allowed by the technical specifications. Richardson, Tr. 3499.

71. If drywell leakage occurs in a DWB accident, hydrogen would leak out through the drywell wall. Richardson, Tr. 3498. General Electric has done parametric calculations of bypass leakage assuming 0.168 square feet leakage area, and found that 14-19% of the total hydrogen generated could bypass the suppression pool. Holtzclaw, Tr. 3628-29. No evaluations have been performed of the effect of this leakage on hydrogen concentrations in the containment as a function of time. Richardson, Tr. 3500.

72. The Brookhaven National Laboratory performed calculations of bypass leakage for GESSAR which showed that, assuming a 2.7 psid is needed to activate suppression pool flow and a low internal drywell pressure, an opening as small as 4 inches in diameter (0.0873 square feet) will eliminate flow through the suppression pool. Pratt, Tr. 3708-12. The greater the pool

height, the greater the differential pressure needed to activate pool flow, and the smaller the bypass leakage area need to prevent pool flow. Pratt, Tr. 3712-13. The suppression pool makeup system is automatically actuated at 30 minutes after a LOCA signal, and would raise the level of the pool. Id.

73. Comparisons of CLASIX 3 with CONTEMPT-LT28, an acceptable and conservative computer code, showed that CLASIX 3 underpredicted atmospheric temperatures because it overpredicted convective heat transfer. Notafrancesco, Tr. 3686-87. Because of this, the Staff found CLASIX 3 to be in nonconformance with the provisions of NUREG-0588, and stated that it should not be used for determining the most severe compartment temperatures. Id.; OCRE Ex. 20 at 1.

74. The CLASIX 3 analysis of the SORV transient was terminated when the hydrogen release ended (75% metal-water reaction was reached). Applicants' Ex. 8-1 at Appendix A, p. 2. At the end of the transient, the gas concentrations in the various compartments are:

Drywell, [O2] = 17.45%; [N2] = 72%; [H2] = 0.94%; [H2O] = 9.5%.
Wetwell, [O2] = 3%; [N2] = 49%; [H2] = 26%; [H2O] = 23%;
Containment, [O2] = 2%; [N2] = 62%; [H2] = 5%; [H2O] = 31%.
Applicants' Ex. 8-1 at Appendix A, Figures 10-21. Final containment and wetwell pressure at the end of the transient is 20 psia (5.3 psig). Id. at Figures 6, 7.

75. The DWB transient was continued past the end of hydrogen release to allow for a drywell burn, at which time the transient was terminated. Id. at 2, Figures 22-28. At the end of the transient, the gas concentrations in the various concentrations are:

Drywell, [O2] = 0%; [N2] = 29%; [H2] = 16%; [H2O] = 50%; Wetwell, [O2] = 0.5%; [N2] = 65%; [H2] = 3%; [H2O] = 28%; Containment, [O2] = 3%; [N2] = 67%; [H2] = 13%; [H2O] = 28%. Id. at Figures 29-40. Wetwell and containment pressure at the end of the transient is 21 psia (6.3 psig). Id. at Figures 26, 27.

76. The containment vacuum relief valves are not modeled in the Perry CLASIX 3 analysis. Id. at 3.

77. Recovery of a DWB accident before the time postulated in the analyses, e.g., at 5500 seconds into the transient, would result in condensation of the large steam fraction (about 75%) in the drywell at that time. Fuls, Tr. 3535-36; Buzzelli, Tr. 3537. Simplified calculations of this event showed that the concentrations of oxygen and hydrogen would rise from 2% and 13% to 8% and 57%, respectively. Fuls, Tr. 3534, 3541; Lewis, Tr. 3541. The mixture, originally nonflammable, then becomes combustible. Fuls, Tr. 3535, 3540-41. The pressure rise from the combustion of this mixture would be 50-55 psi, assuming initial atmospheric pressure, 23 psia if the drywell were assumed at an initial

pressure of 6 psia. Lewis, Tr. 3542. The mixture is rich, oxygen limited. Id.; Fuls, Tr. 3541. In actuality the action of the drywell vacuum breakers would limit the extent of depressurization and admit more oxygen into the drywell. Fuls, Tr. 3542.

78. Gas mixtures containing less than 5% oxygen or more than 55% steam (by volume) are inert. OCRE Ex. 21 at 15.

79. The pressure rise caused by a hydrogen deflagration is

mainly dependent on the initial pressure (pre-burn) and the number of moles of hydrogen consumed. OCRE Ex. 21 at 199. Small changes in initial pressure can produce significant changes in peak burn pressure. Id. at 17. Elevated initial pressures in containment are expected for the DWB scenario, in which the drywell air mass is pushed into the containment. Id. at 29, 94

80. Sandia performed a number of calculations with the HECTR code as a comparison with the GGNS CLASIX 3 models. OCRE Ex. 21 at 15-16, 28-29. Some of these cases used identical input parameters and compartmentalization models, while others used input values considered more realistic by Sandia. Id. In all cases HECTR predicted higher peak pressures than did CLASIX 3. Id.

81. HECTR case B-1 modeled CLASIX 3 case 1. Id. at 28. Identical burn parameters (10% ignition and propagation limits, 100% combustion completeness, 6 ft/sec flame speed) were used. Id. at 20-21. Containment sprays were assumed to initiate automatically after the first burn. Id. These cases modeled an elevated initial, pre-burn pressure, as would result from the DWB transient. Id.; Id. at 29. HECTR case B-1 used a hydrogen source term equal to that in CLASIX-3; all other HECTR runs used a hydrogen source term less than that in CLASIX 3. Id. at 18, 20. CLASIX 3 produced a peak pressure of 1.755 atm (11.1 psig); HECTR, 5.6 atm (67.6 psig). Id. at 30.

82. HECTR case B-3 models CLASIX 3 case 3, which used the same burn and spray parameters as in the PNPP SORV analysis. Id. at 20-21. CLASIX 3 produced a peak pressure of 1.456 atm (6.7 psig). HECTR's peak pressure was 3.830 atm (41.6 psig). Id. at

30.

83. HECTR case B-4 models CLASIX 3 case 6. These cases are the same as the previously considered case, except that sprays are assumed to be off. CLASIX 3 predicted a peak pressure of 1.701 atm (10.3 psig); HECTR, 4.63 atm (53.4 psig). Id. at 30.

84. CLASIX 3 case 5 considered a compartmentalization model consisting of one containment compartment (i.e., the wetwell was not considered a separate compartment); HECTR case A-1 also modeled this condition. Id. at 20-21, 28. CLASIX 3 predicted a peak pressure of 3.361 atm (34.7 psig); HECTR, 4.09 atm (45.4 psig). Id. at 30.

85. HECTR cases B-5, B-6, B-6', and B-7 used different burn parameters, but all modeled the CLASIX 3 compartmentalization. Id. at 29. All these cases assumed 100% combustion completeness, flame speeds which were a function of hydrogen concentration (using a correlation developed from turbulent flame experiments in the Sandia VGES tank), and realistic propagation limits (4.1% for upward propagation, 6% for horizantal, and 9% for downward). Id. at 15, 18, 22, 29. Case B-5 used an ignition limit of 8%, with automatic spray action. Id. This case gave a peak pressure of 4.877 atm (57 psig). Id. at 31. Case B-7 was similar, but used a 10% ignition limit. Id. at 22. This case had a peak pressure of 5.502 atm (66.2 psig). Id. at 31.

86. Cases B-6 and B-6' were identical, except that B-6 was terminated when the hydrogen release ceased, whereas B-6' was continued past that time. Id. at 29. Both assumed an ignition limit of 10%, and sprays were assumed on during the entire run. Id. at 22. Case B-6 produced a peak pressure of 2.467 atm (21.6

psig), while B-6' had a peak of 5.693 atm (69 psig). Id. at 31.

87. Realistic propagation limits such as used in HECTR cases B-5 through B-7 will cause upward propagating burns to occur, limiting the buildup of hydrogen in upper compartments. Id. at 18. The CLASIX 3 propagation model is unrealistic. Id.

88. In both CLASIX 3 and HECTR, wetwell burns have small pressure rises, while containment burns lead to high peak pressures. Id. at 16, 199; Applicants' Ex. 8-1 at Appendix A, pp. 6-7. This is because the small wetwell rapidly reaches the necessary conditions for combustion, and the resultant pressure rise is small due to the transport of gas from the wetwell to the containment. OCRE Ex. 21 at 199. In CLASIX 3 inerting of the wetwell (by oxygen depletion or by steam) is prevented because containment air is transported into the wetwell at an artifically high rate when the post-combustion gases cool. Id. CLASIX 3 predicts faster cooling than does HECTR. Id. at 16. The result of using CLASIX
3 with its present compartimentalization is an unrealistic calculation of the mixing and combustion process, leading to underestimated hydrogen burn pressures. Id. at 199.

89. Incomplete combustion results in lower pressure rises than complete combustion, which results in fewer burns with larger pressure rises accompanying each burn. Id. at 93.

90. HECTR cases B-2 and B-2' modeled CLASIX 3 case 2. OCRE Ex. 21 at 28. The input assumptions were the same as for CLASIX 3 case 1 (10% ignition and propagation limits, automatic spray option, and flame speed of 6 ft/sec), except that pre-burn pressure was not elevated. Id. at 28. The HECTR run used a

smaller hydrogen release than did CLASIX 3. HECTR case B-2 produced a peak pressure of 4.845 atm (56.5 psig), compared to the CLASIX 3 peak of 1.504 atm (7.4 psig). Id. at 30. (HECTR case B-2' examined the treatment of wetwell sprays, and found that wetwell inerting was delayed, but gave similar final results. Id. at 28.)

91. Applicants have complied a list of equipment required to survive a hydrogen burn. Applicants' Testimony at 50; Applicants' Ex. 8-1 at Table 5.6-1.

92. Applicants conducted a preliminary evaluation of equipment survivability based on a comparison of PNPP and GGNS containment pressure and temperature profiles calculated by the CLASIX 3 code. Applicants' Ex. 8-1 at 21A; Garg Testimony at 3-6. Applicants conducted an analysis, using the HEATING-6 heat transfer computer code, of the igniter assembly, using the CLASIX 3 temperature profile to confirm that the PNPP temperature profile results in lower equipment temperature than the GGNS profile. Applicants' Ex. 8-1 at 21C. Applicants base their conclusion of equipment survival on the similarity between GGNS and PNPP temperature profiles, the heat transfer analysis, the similarity of the PNPP and GGNS equipment lists, and the margins between the response calculated for GGNS and the PNPP equipment qualification temperatures. Applicants' Ex. 8-1 at 21D.

93. The containment vacuum breakers, 1M17F0010, 1M17F0020, 1M17F0030, and 1M17F0040, are only qualified to 15 psig. Applicants' Ex. 8-1 at 21D, Table 5.6-1. The hydrogen mixing compressors are qualified to 14.9 psig. Id. The compressor

check valves are qualified to 15.3 psig. Id. Applicants assumed that the check valves and vacuum breaker will not be exposed to peak burn pressures. Id. The compressors at GGNS were evaluated and shown to survive pressures of 24 psig. Id. These assumptions and evaluations have not been confirmed by testing. Buzzelli, Tr. 3570.

94. The likelihood of a detonation occuring is very small. OCRE Ex. 21 at 194.

95. Tests conducted in a 1/20 scale model of the Mark III containment found that diffusion flames exist for hydrogen release rates exceeding 0.4-0.5 lb/sec. Richardson, Tr. 3551. Preliminary results of these tests indicate that thermal loading to equipment resulting from a 75% metal water reaction will be unacceptable. Richardson, Tr. 3553-57.

96. The Hydrogen Control Owners Group ("HCOG") is conducting tests in a 1/4 scale test facility to better define the thermal environments resulting from diffusion flames. Richardson, Tr. 3552; Applicants' Testimony at 23-24. HCOG does not intend to evaluate the thermal environments caused by diffusion flames using a release history equivalent to a 75% metal-water reaction in the 1/4 scale tests. Richardson, Tr. 3568.

97. CLASIX 3 sensitivity studies (using input assumptions other than those in Applicants' preliminary analysis) have predicted violent overflow of the suppression pool into the drywell. Richardson, Tr. 3487, 3494. Sandia also predicted that significant pool surge may occur in the drywell. OCRE Ex. 21 at 195.

98. Long term decay heat removal is needed to maintain the core

in a safe condition. Richardson, Tr. 3478. Successful decay heat removal depends upon a having an RHR loop operating in the suppression pool cooling mode. Richardson, Tr. 3576-77.

99. The PNPP RHR system includes suppression pool cooling and containment spray subsystems. Applicants' Ex. 8-1 at 25. Containment sprays are automatically actuated at 9 psig. Id. at 26. The spray subsystem takes precedence over other RHR functions, except for LPCI during the first 10 minutes following a LOCA signal. Richardson, Tr. 3448. If containment pressure remains above 9 psig, both trains of RHR will operate in the spray mode. Richardson, Tr. 3448. Operator action is necessary to align the RHR system from the spray to other modes, even after pressure drops below 9 psig. Richardson, Tr. 3449.

100. Having both RHR loops in the containment spray mode will degrade suppression pool cooling and mixing. Richardson, Tr. 3453-55, 3481-85.

101. Equipment survivability tests conducted at the Nevada Test Site resulted in extensive burning of electrical cable insulation during tests involving 13% hydrogen; burning was for shorter durations at a hydrogen concentration of 10%. Garg, Tr. 3717, 3748. Heat loads imposed on electrical cable insulation by hydrogen combustion can result in pyrolysis or gasification of the insulation compounds. OCRE Ex. 24 at 1201, 1205. The combustible gases thus liberated will affect the subsequent containment pressure and temperature histories. Id. at 1205-06.

102. Applicants submitted to the Staff on March 1, 1985 a preliminary analysis of the hydrogen control system. Applicants' Ex. 8-1; Applicants' Testimony at 19. The analysis included a description of the igniter system, an analysis of

containment capacity, a containment response analysis, and a comparison of the PNPP and GGNS design features. Id. Prior to submitting the analysis, Applicants and Staff agreed on its scope. Id.; Applicants' Ex. 8-2, 8-3.

On March 21, 1985 Applicants submitted an update to the preliminary analysis. This submittal included an analysis of equipment survivability. Applicants' Ex. 8-1 at 21A-21D.

103. The basis for the Staff's judgement of the scope of the preliminary analysis is what was required of Grand Gulf for operation above 5% power. Notafrancesco, Tr. 3742-43. The applicable SER supplement, GGNS SSER 5 (August 1984), included a discussion of the 1/20 scale tests, comparisons of CLASIX 3 with CONTEMPT-LT28, and referenced the findings of SSER 3, which relied upon HECTR calculations in NUREG/CR-2530. Notafrancesco, Tr. 3740, 3744-46. No HECTR calculations performed after the issuance of that NUREG were relied upon by the Staff in its evaluation of Grand Gulf. Id.

104. The final analysis of the PNPP hydrogen control system is scheduled for completion in mid-1986, and is dependent on experimental and analytical work to be performed by HCOG. Applicants' Testimony at 22.

III. CONCLUSIONS OF LAW

Applicants have failed to meet their burden of proof on Issue #8, as compliance with 10 CFR 50.44 (c)(3)(iv)-(vii) has not been demonstrated.

IV. ORDER

WHEREFORE, IT IS ORDERED

1. The application for licenses to operate the Perry Nuclear Power Plant, Units 1 and 2, at power levels greater than 5%, is DENIED.

2. Pursuant to 10 CFR 2.760(a), this is a partial initial decision that will constitute final action of the Commission forty-five (45) days from the date of issuance unless a notice of appeal is filed pursuant to 10 CFR 2.762 or the Commission directs that the record be certified to it.

3. A notice of appeal of this decision or designated portions thereof may be filed with the Commission, in the form required by 10 CFR 2.762(a), within ten (10) days after service of this decision.

4. To pursue an appeal, briefs in support of the appellant's position must also be filed, within thirty (30) days after filing the notice of appeal (forty days if the NRC Staff is the appellant). The brief must comply with the requirements of Section 2.762.

5. Within thirty (30) days (forty days for the NRC Staff) after the time has expired for the filing and service of all appellant briefs, parties may file opposing or supporting briefs that comply with the requirements of Section 2.762.

6. Filings that do not comply with the rule governing appeals may be striken.

Respectfully submitted,

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APPENDIX A

WRITTEN TESTIMONY RECEIVED INTO EVIDENCE

APPLICANTS' TESTIMONY

Following Tr. 3241:

Applicants' Direct Testimony of Eileen M. Buzzelli, John D. Richardson, Kevin W. Holtzclaw, Roger W. Alley, Bernard Lewis, Bela Karlovitz, and G. Martin Fuls on the Preliminary Evaluation of the Perry Nuclear Power Plant Hydrogen Control System (Issue #8), with statement of qualifications of said witnesses.

Following Tr. 3752:

Oral Testimony (without written testimony) of James H. Wilcox

STAFF'S TESTIMONY

Following Tr. 3676:

Testimony of Allen Notafrancesco Regarding Issue #8 (Hydrogen Control), with statement of qualifications ("Notafrancesco Testimony I")

Testimony of Li Yang Regarding Issue #8 (Hydrogen Control), with statement of qualifications

Testimony of Allen Notafrancesco on the Hydrogen Control Issues Contained in the Licensing Board Contention # 8 ("Notafrancesco Testimony II")

Testimony of Hukam C. Garg Regarding Issue #8 (Hydrogen Control), with statement of qualifications

Statement of qualifications of William Trevor Pratt (oral testimony without written testimony)

APPENDIX B
Exhibits

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page	Following Transcript Page
App. Ex. 8-1	The Cleveland Electric Illuminating Company Preliminary Evaluation of the Perry Nuclear Power Plant Hydrogen Control System, March 21, 1985	3219	3243	3243
App. Ex. 8-2	Letter from M. Edelman to B.J. Youngblood, dated February 5, 1985 re Perry Nuclear Power Plant Hydrogen Control Evaluation	3219	3243	3243
App. Ex. 8-3	Letter from B.J. Youngblood to M. Edelman, dated February 20, 1985 re Acceptability of the Scope of Hydrogen Control Design and Analytical Information to be Provided to Support Full Power Licensing of Perry Nuclear Power Plant, Unit 1	3219	3243	3243
App. Ex. 8-4	Letter from M. Edelman to B.J. Youngblood, dated February 11, 1985 re SER Confirmatory Issue (3) Containment Ultimate Capacity Analysis	3219	3243	3243

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page	Following Transcript Page
Staff Ex. 8	"Safety Evaluation Report, NUREG-0887, Supplement No. 6"	3675	3677	not bound into record
OCRE Ex. 12	Letter from Mr. C.O. Thomas to Mr. G.G. Sherwood, dated April 13, 1984 re Request for Additional Information Regarding the Severe Accident Review of Gessar II	3261	3263	3263
OCRE Ex. 13	"Analysis of Inaccessible and Potentially Rejectable Defects in Perry Nuclear Power Plant" authored by Warren P. McNaughton, Jeffrey R. Egan and Jeffrey D. Byron of Aptech Engineering Services, dated July 1983	3305	3343	3343
OCRE Ex. 14	Letter from M. Edelman to J. Keppler, dated March 7, 1985 re Drywell Airlock Door Seals	3377	3378	3378
OCRE Ex. 15	Letter from M. Edelman to Mr. B.J. Youngblood, dated May 29, 1984 re Piping Design Review	3465	3467	3467
OCRE Ex. 16	Table 2.2-1, Igniter locations from the Perry Nuclear Power Plant Units 1 & 2 Interim Report on the Hydrogen Control System	3508	3508	3508

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page	Following Transcript Page
OCRE Ex. 17	Attachment A, "Experimental Study of H_2 Diffusion Flames Burning Above a Pool of Water", from the Combex Study of Hydrogen Control at Grand Gulf Nuclear Station, dated 1981	3562	3562	3562
OCRE Ex. 18	NRC Memorandum from John Stefano to B.J. Youngblood, dated May 4, 1983 re Summary Report of meeting with the Cleveland Electric Illuminating Company (CEI) on Perry Containment Weld Deficiencies	3680	3681	3681
OCRE Ex. 19	Letter from A. Schwencer to D. Davidson, dated September 16, 1982 re Request for Additional Information Regarding Degraded Core Hydrogen Control for the Perry Nuclear Power Plant (Units 1 and 2)	3682	3683	3683
OCRE Ex. 20	Letter from B.J. Youngblood to M. Edelman, dated August 30, 1984 re Request for Additional Information Regarding Hydrogen	3685	3685	3685
OCRE Ex. 21	NUREG/CR 2530 Review of the Grand Gulf Igniter System	3691	3691	3691
OCRE Ex. 22	NRC Memorandum from Marc Wigdor,	3693	REJECTED, 3696	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page	Following Transcript Page
	through Jack Rosenthal, to Brian Sheron, dated October 24, 1984 re Hydrogen Control Owners' Group and NRC Meeting, October 3 and 4, 1984, discussing the use of BWR Heatup Code			
OCRE Ex. 23	Draft Report, "An Assessment of Postulated Degraded Core Accidents in the Grand Gulf Reactor Plant," by R.D. Gasser of Brookhaven National Laboratory, dated June 1982	3701	REJECTED, 3703	
OCRE Ex. 24	Paper presented at the Second International Conference on the Impact of Hydrogen on Water Reactor Safety, "Electrical Cable Insulation Pyrolysis and Ignition Resulting from Potential Hydrogen Burn Scenarios for Nuclear Containment Buildings," by A.L. Berlad, R. Jaung and W.T. Pratt	3714	3715	3715

APPENDIX C

COMPARISON OF CONTAINMENT RESPONSE ANALYSES

CLASIX 3

PLANT	CASE	IGN. LIMIT	PROP. LIMIT	% COMPLETE	FLAME SPEED, FT/SEC	SPRAYS	NO. OF BURNS CT	NO. OF BURNS WW	TMAX, °F	PMAX (PSIG)	NOTES
PNPP	SORV	8	8	85	6	AUTO	2	32	1762	21.2	1
NPP	DWB	8	8	85	6	AUTO	0	30	1201	13.8	1,3
GGNS	SORV	8	8	85	6	AUTO	1	59	1020	23.9	1
GGNS	DWB	8	8	85	6	AUTO	0	26	1110	11.9	1,3
GGNS	1	10	10	100	6	AUTO	0	18	1494	11.1	2,4
GNS	2	10	10	100	6	AUTO	0	43	1471	7.4	2,5
GNS	3	8	8	85	6	AUTO	0	58	1062	6.7	2,6
GNS	5	8	8	85	6	OFF	4	4	2023	34.7	2,7
GGNS	6	8	8	85	6	OFF	0	68	1575	10.3	2,8

NOTES:

1. TAKEN FROM TABLE 18 OF APPENDIX A OF APPLICANTS' EX. 8-1.
2. TAKEN FROM TABLES 2.1 AND 2.15 OF OCRE EX. 21.
3. TEMPERATURE AND PRESSURE MAXIMA TAKEN FROM WETWELL AND CONTAINMENT VALUES.
4. HECTR CASE B-1 MODELS CLASIX 3 CASE 1.
5. HECTR CASES B-2 AND B-2' MODEL CLASIX 3 CASE 2. HECTR CASE B-2' ALLOWED SPRAY CARRYOVER INTO THE WETWELL.

HECTR

PLANT	CASE	IGN. LIMIT	PROP. LIMIT	% COMPLETE	FLAME SPEED FT/SEC	SPRAYS	No. OF BURNS CT	No. OF BURNS WW	T_{max}, °F	P_{max} (PSIG)	NOTES
GGNS	B-1	10	10	100	6	AUTO	1	21	1877	67.6	10, 4, 14
GGNS	B-2	10	10	100	6	AUTO	1	24	1922	56.5	10, 5, 14
GGNS	B-2'	10	10	100	6	AUTO	1	21	1814	59.4	10, 5, 14
GGNS	B-3	8	8	85	6	AUTO	1	32	1877	41.6	10, 6, 14
GGNS	A-1	8	8	85	6	OFF	4	(NO WW)	1436	45.4	10, 7, 14
GGNS	B-4	8	8	85	6	OFF	1	30	1907	53.4	10, 8, 14
GGNS	B-5	8	G(% H_2)	100	F(% H_2) = 21.4	AUTO	1	30	1890	57.0	11, 9, 12, 14
GGNS	B-6	10	G(% H_2)	100	F(% H_2) = 25.3	ON	1	18	1890	21.6	11, 9, 12, 14
GGNS	B-6'	10	G(% H_2)	100	F(% H_2) = 25.3	ON	2	19	2078	69.0	11, 9, 12, 1[illegible] 14
GGNS	B-7	10	G(% H_2)	100	F(% H_2) = 25.3	AUTO	2	21	2032	66.2	11, 9, 12, 1[illegible]

NOTES, CONTINUED:

6. HECTR CASE B-3 MODELS CLASIX 3 CASE 3.
7. HECTR CASE A-1 MODELS CLASIX 3 CASE 5.
8. HECTR CASE B-4 MODELS CLASIX 3 CASE 6.
9. G(% H_2) = 4.1% FOR UPWARD PROPAGATION, 6% FOR HORIZANTAL PROPAGATION, 9% FOR DOWNWARD PRO[illegible]
10. TAKEN FROM TABLES 2.2 AND 2.16 OF OCRE EX. 21.
11. TAKEN FROM TABLES 2.4 AND 2.18 OF OCRE EX. 21.
12. FLAME SPEED CALCULATED FROM EQUATION 2.1 OF OCRE EX. 21, USING IGNITION LIMIT H_2 CONCENTRATI[illegible]
13. HECTR CASE B-6' IDENTICAL TO B-6, EXCEPT FOR LONGER RUN TIME.
14. H_2 SOURCE TERM FOR HECTR CASE B-1 IS EQUAL TO CLASIX 3 SOURCE TERM; ALL OTHER HECTR CASES HAVE A SOURCE TERM LESS THAN CLASIX 3.

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing were served by deposit in the U.S. Mail, first class, postage prepaid, this 13th day of JUNE, 1985 to those on the service list below.

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USNRC

Susan L. Hiatt
Susan L. Hiatt

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