September 16, 1982

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

APPLICANTS' ANSWER TO OHIO CITIZENS
FOR RESPONSIBLE ENERGY MOTION
FOR LEAVE TO FILE ITS CONTENTIONS 21-26

By its "Motion for Leave to File its Contentions 21 through 26", dated August 18, 1982, ("Motion"), Ohio Citizens for Responsible Energy ("OCRE") seeks leave to amend its petition for leave to intervene in this proceeding. OCRE's Motion sets out six proposed new contentions (Contentions 21-26), which Applicants address seriatim below. Because OCRE's Motion fails to demonstrate either adequate basis and specificity, or good cause, or both as to each of its late-filed new contentions, the Motion should be denied.

I. OCRE CONTENTION 21 (Turbine Missiles)

OCRE's first proposed contention states:

> [T]he placement and orientation of the PNPP turbine-generators are unacceptable because low trajectory turbine missiles could strike safety-related targets, thereby endangering the safe operation of the facility.

Motion at 1. The contention is totally untimely, with no showing of good cause. It also is totally lacking in basis.

A. Timeliness.

The five factors to be considered in determining whether a petition for a late-filed contention should be granted are set forth in 10 C.F.R. § 2.714(a)(1).

> (i) Good cause, if any, for failure to file on time.
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> (ii) The availability of other means whereby the petitioner's interest will be protected.
>
> (iii) The extent to which the petitioner's participation may reasonably be expected to assist in developing a sound record.
>
> (iv) The extent to which the petitioner's interest will be represented by existing parties.
>
> (v) The extent to which the petitioner's participation will broaden the issues or delay the proceeding.

Perhaps the most crucial of these factors is the requirement that intervenors must demonstrate good cause for their untimely filing.

In this contention, as in several others, OCRE's good cause argument is that the Staff's Safety Evaluation Report for the Perry facility (NUREG-0887, issued May 1982 ("SER")) "constituted OCRE's first notice of the concerns identified in" the contention. Motion at 7. There is no conceivable way that the SER was the first reasonably available notice of the contention. Indeed, OCRE itself cites a 1976 report by Gilbert Associates, Inc.1/ Motion at 1. OCRE also admits that the Gilbert Report is referenced in the Final Safety Analysis Report (see FSAR, § 3.5.1.3). In fact, this same Gilbert Report was referenced in the Preliminary Safety Analysis Report (see PSAR § 3.5.3.1). The Gilbert Report was discussed in the Safety Evaluation Report issued at the construction stage.2/ Even a cursory review of the construction permit documents shows that the issue now raised by OCRE, low trajectory turbine missiles, was extensively examined at the construction permit stage. See footnote 2 references. And the problem also is discussed in the FSAR (§ 3.5.1.3), citing to the Gilbert Report. Thus, by no stretch of the imagination can the operating license SER constitute good cause for this untimely

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1/ Gilbert Associates, Inc. Report No. 1848, "An Analysis of Low Trajectory Turbine Missile Hazard to the Perry Nuclear Power Plant, Units 1 & 2" (October 8, 1976)("Gilbert Report").

2/ See, e.g., Supp. 4 to the Construction Permit stage SER, § 10.2, p. 25 (January, 1977); Supp. 5 to the Construction Permit stage SER, § 10.2, p. 5 (February, 1977).

filing. The documentation on which OCRE bases this contention has existed for six years and was readily available. The issuance of the SER provides no excuse for the untimeliness.

Nor are OCRE's showings on the other lateness factors any stronger. OCRE states that it has only this forum in which to protect its interests, Motion at 7-8, but fails to show why the NRC Staff's review (both at the construction permit and operating license stages) is not adequate. OCRE claims that its participation will aid in developing a sound record, but provides no more than a paraphrase of the regulatory standard and a citation to the Licensing Board's July 12, 1982 Memorandum and Order, LBP-82-53, slip op. at 5. A commendation by the Licensing Board in one context does not necessarily establish a party's capabilities for all time and for all issues. See LBP-82-11, 15 N.R.C. 348, 352 (no basis shown for OCRE special competence on core catcher issue); cf. Memorandum and Order (Concerning Sunflower's Late-Filed Radiation-Dose Contention), September 15, 1982, slip op. at 3.

Finally, OCRE concedes that admitting this (and the other contentions) might cause delay, but opines that "this should not be of concern to any party." OCRE's cavalier disregard for delay is supported by its reference to Applicants' July 21, 1982 application to extend the construction completion dates set forth in the Perry construction permits. The application requests that the latest date for completion of construction be

changed from December 1982 and June 1984 for Units 1 and 2, respectively, to November 1985 and November 1991. OCRE apparently misunderstands the nature and purpose of this application. OCRE appears to believe that the requested extension means that the units will not be completed until the requested dates (i.e., November 30, 1985, for Unit 1, and November 30, 1991, for Unit 2). This simply is not correct. All that the requested extension does is seek to extend the latest date for completion of each of the units.3/ This is being done to account for possible time contingencies that may extend the completion dates beyond Applicant's present estimates.4/ It is important to note, however, that these are contingencies that may or may not cause substantial construction delay depending on factors not entirely under Applicants' control. Because Applicants believe that it would be both unnecessary and wasteful to file a series of amendments, they are now seeking to amend their permits to reflect the possible impact of these time contingencies. That is not the same as predicting (or conceding) that Unit 1 will not be completed

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3/ Section 185 of the Atomic Energy Act, 42 U.S.C. § 2235, requires construction permits to specify the earliest and latest dates for completion of construction. See also 10 C.F.R. § 50.55.

4/ As stated in the application, Applicants' scheduled dates of commercial operation remain May 1984, for Unit 1 and May 1988, for Unit 2.

until 1985 or Unit 2 until 1991. OCRE's misunderstanding of the latest completion date requirement cannot justify its desire to delay this proceeding.

OCRE, therefore, has failed to meet its burden justifying its untimely filing.

B. Basis and Specificity.

Aside from its total untimeliness, OCRE's proposed Contention 21 lacks basis and specificity. OCRE cites three references in support of the contention -- the Perry SER, the ACRS letter, and the 1976 Gilbert Report. None of these provide an adequate basis.

The SER discussion cited by OCRE states:

> The Perry turbine-generators were manufactured by the General Electric Company and the placement and orientation of each is unfavorable with respect to the station reactor buildings; that is, there are safety-related targets inside the low trajectory missile strike zone. Additional information that was requested from the applicant has been received by the staff and is currently under review. Until the staff completes the review, this will remain an open item.

SER § 3.5.1.3, p. 3-10.5/ The mere fact that NRC review of "additional information" is not yet complete can hardly constitute a basis for OCRE's concern. (In fact, the

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5/ The orientation of the turbine vis-a-vis station reactor buildings has, of course, been previously analyzed by Applicants and reviewed by the Staff. See n.2 supra.

"additional information" requested by the Staff was a copy of the 1976 Gilbert Report.) The SER provides no substantive information to support the contention.

Nor does the ACRS letter aid OCRE's case. The ACRS merely expressed concern with the pace of NRC Staff's "progress" in resolving the issue. ACRS letter at 3, in Supp. 1 to SER, NUREG-0887 at F-3. The discussion of this issue at the ACRS Meeting indicates that the resolution at issue is the Staff's generic resolution of the turbine missile issue, rather than a Perry-specific concern. ACRS Tr. 55-60 (July 8, 1982). No substantive information on the turbine missile issue is contained in the ACRS letter. Therefore, it cannot constitute a basis for OCRE's contention.

The final reference provided by OCRE is to the 1976 Gilbert Report. OCRE cites the report for showing that the control room, cable spreading room, HVAC equipment room, intermediate building, auxiliary building, electrical penetration area and Units 1 and 2 reactor buildings are within the "low trajectory missile strike zone". Motion at 2. OCRE does not provide a reference, but the data appears in the Gilbert Report, § 2.2. OCRE then claims that

> [t]he estimated damage to these structures resulting from turbine missile impact includes rendering the control room inoperable, the collapse of buildings on safety-related electrical cables and equipment, and penetration of the contention.

Motion at 2. Again, OCRE provides no citation to the Gilbert Report. In fact, the Gilbert Report does not estimate that a turbine missile impact would have these consequences. Rather the report assumes (conservatively) that these types of consequences might occur.6/

Notwithstanding these references, OCRE inexplicably has left out the heart of the Gilbert Report and its conclusions. The major portion of the Gilbert Report is devoted to assessing the probability for damage caused by turbine missiles. OCRE makes no mention of this fact, nor of the results of the analysis. Since OCRE must have had access to the Report to write its contention, this omission would appear to be deliberate. The conclusion expressed in the Gilbert Report, with which OCRE fails to take issue, states:

> The turbine missile hazard to an individual safety related target has been conservatively demonstrated to be less than 1.5 E-8 per year per turbine. This acceptably low value is within the limits prescribed by Regulatory Guide 1.115 and, therefore, redesign for additional turbine missile protection is unnecessary.

Gilbert Report, § 3.6. Since the Gilbert Report concludes that low trajectory turbine missiles are not a hazard, OCRE cannot use the report as the basis for a contention which alleges the

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6/ Gilbert Report, § 2.3 ("No credit is taken for the potential of penetrating a target without causing a loss of safety function.")

precise contrary. The construction permit SER, which OCRE neglected to cite, also reached the same conclusion:

> On the basis of our independent analysis, we conclude that the probability of a turbine missile causing unacceptable damage is within our acceptance criteria for this type of event and that, therefore, with regard to postulated turbine missiles, the proposed plant design is acceptable.

Supp. 4 to Construction Permit SER, § 10.2.

OCRE has provided no basis for challenging these conclusions and, therefore, no basis for its contention.

II. CONTENTION 22 (Mark III Containment Concerns)

As its Contention 22, OCRE submits sixty-six separate issues (the Humphrey concerns) which it would have this Licensing Board admit into this proceeding.7/ Admission of this contention would expand the number of issues to be adjudicated by this Licensing Board from the nine contentions now admitted, to a total of seventy-five separate contentions. As justification for such an extraordinary enlargement of this proceeding, OCRE does no more than attach a list of sixty-six concerns, with a prefatory statement that the sixty-six issues should be admitted *en masse* because "most" have not been "resolved" as yet with regard to Perry.

7/ The sixty-six separate Humphrey concerns fall within twenty-two major categories.

The contention should not be admitted. OCRE has failed utterly even to attempt to comply with the Commission's "basis and specificity" requirement for new contentions.8/ To the contrary, as to none of the sixty-six individual Humphrey concerns does OCRE demonstrate that the concern is a matter within the Licensing Board's jurisdiction (that is, related to safety or environmental considerations), or that the concern is at all relevant to Perry,9/ or that the concern in fact is unresolved.10/

OCRE's motion to admit Contention 22 simply ignores the Commission's "basis and specificity" requirement. The Licensing Board is asked effectively to admit sixty-six new contentions without any indication that any one of these contentions is within the Licensing Board's jurisdiction, is relevant to Perry, or is unresolved. Simply put, OCRE wants

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8/ See 10 C.F.R. § 2.714(b) (requiring that contentions be supported by "bases . . . set forth with reasonable specificity"). The mandatory nature of the requirement recently was reaffirmed in Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), ALAB-687 (Aug. 19, 1982).

9/ OCRE's failure in this regard is all the more astounding in light of its concession that some of the concerns do not even apply to Grand Gulf, the BWR 6/Mark III to which they were addressed by Mr. Humphrey. Motion at 2.

10/ OCRE contends only that "most [of the concerns] are still unresolved." Motion at 2. No distinction is made between the concerns that have been resolved and those that have not. OCRE does not even identify which of the concerns fall in each category.

the Licensing Board to admit sixty-six new issues with the "basis and specificity" of each issue to be determined sometime later in this proceeding. Admission of such unsupported contentions, however, clearly is prohibited by the recent ruling of the Appeal Board in Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), ALAB-687 (August 19, 1982). As unequivocally stated there, "a licensing board is not authorized to admit conditionally, for any reason, a contention that falls short of meeting the specificity requirements."11/ Id., slip op. at 11 (emphasis in original).

OCRE's failure to provide "basis and specificity" for any of the sixty-six concerns is all the more egregious in light of the treatment accorded to the Humphrey concerns by the NRC Staff and the Advisory Committee on Reactor Safeguards ("ACRS"). The Humphrey concerns are addressed by the NRC Staff in § 6.2 of Supplement No. 1 to the Perry SER.12/ That section

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11/ Although OCRE does not ask the Licensing Board to admit the individual issues "conditionally," any admission would have to be conditioned on OCRE being able to show that particular issues are within the Licensing Board's jurisdiction, are related to Perry, and are unresolved. None of these showings have as yet been made as to any of the sixty-six Humphrey concerns.

12/ By letter of July 14, 1982 (A. Schwencer to D. Davidson), the NRC Staff has requested Applicants to answer certain questions raised by some of the Humphrey concerns. Some of the questions merely are confirmatory in nature, while many others only ask for details regarding analyses that have already been provided to the Staff. The letter does not, as OCRE appears to suggest in its Motion, state that the Humphrey concerns raise unresolved safety issues.

indicates that the Staff has made a preliminary assessment of the Humphrey concerns, and has concluded that, of the twenty-three major items, only two have not already had prior consideration or do not represent significant safety concerns.13/ These two items, discussed in greater detail *infra*, are "(1) the effects that structural encroachments over the suppression pool might have on pool swell and impact loads and (2) the response of the residual heat removal (RHR) system, when it is used in the steam condensing mode, to loads produced by the steam condensation phenomenon." NUREG-0887, Supp. 1, § 6.2.

The NRC Staff's conclusion is consistent with that of the ACRS, which also reviewed the Humphrey concerns. On July 29-30, 1982, the ACRS Subcommittee on Fluid Dynamics held a hearing on the concerns. A portion of the transcript is attached. As is plain from the transcript, there is a "consensus" (Tr. 371) among the ACRS Fluid Dynamics Subcommittee that the Humphrey concerns do not raise serious safety questions (Tr. 360-62, 365 and 373-74), and "are receiving far more attention than they deserve" by the NRC and the industry (Tr. 370-71). Indeed, Dr. Butler expressed the view, which appeared to be shared by members of the Subcommittee, that "many of these new areas are really design questions rather than safety

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13/ The Staff's conclusion was first stated in § 6.1 of Supplement No. 3 to the Grand Gulf SER (NUREG-0831, July 1982).

questions," and that "[t]here are no real technological questions at hand associated with these issues" (Tr. 373-74). Finally, the Subcommittee strongly recommended to the NRC Staff "that the issues . . . be resolved generically rather than case by case . . ." (Tr. 361-62, emphasis added).

Applicants do not cite the above language from the Perry SER Supplement and the ACRS Subcommittee hearing for the purpose of litigating here the substantive merits of the Humphrey concerns. Rather, Applicants wish to emphasize that the substantial analysis of the concerns already undertaken makes it imperative that OCRE, at minimum, be required to demonstrate the requisite "basis and specificity" as to each of the concerns it asks the Licensing Board to admit. Specifically, OCRE must demonstrate as to each concern that the concern is a safety related concern, that the concern in fact is relevant to Perry, and that the concern is unresolved.14/

14/ In this regard, it should be noted that OCRE's list of concerns appears to be taken from an attachment to a June 8, 1982 letter from J. McGaughy, MP&L, to H. Denton, NRC (attached). It perhaps is understandable that OCRE did not identify the source of its list, for an accompanying memorandum to the letter (also attached) indicates that as early as June 8, 1982 many of the Humphrey concerns had already been considered resolved.

OCRE may wish to challenge the way in which the NRC Staff has resolved a particular concern. But in order to demonstrate the requisite "basis and specificity", OCRE should at least have identified the Staff resolution, and the reasons why it believes the resolution to be inadequate. OCRE, of course, has done nothing of the sort, but has chosen instead simply to ignore the fact that many of the concerns have been resolved.

Anything less is in clear violation of Commission regulations.15/ See 10 C.F.R. § 2.714(b).

As noted above, Supplement No. 1 to the Perry SER identifies two items that "warrant priority attention" in the Staff's opinion. As to the first of these concerns -- the effects of structural encroachments over the suppression pool -- Applicants have provided the NRC Staff with an analysis showing that the effects of the structural enroachments are not significant. See Perry SER Supplement 1, at 6-1. OCRE has not suggested that this analysis is deficient. Moreover, OCRE has provided no basis for concluding that the encroachments, to the extent they cause any problems, raise safety related considerations. Most importantly, however, OCRE has demonstrated no "nexus" between this concern and Perry.16/ Thus, to admit this

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15/ Such a holding would be consistent with the Licensing Board's discussion with regard to the control systems contention submitted by Sunflower Alliance, Inc. See Memorandum and Order (Concerning Late-Filed Contentions: Quality Assurance, Hydrogen Explosion, and Need for Increased Safety of Control Systems Equipment), LBP-82-15, 15 N.R.C. 555, 557-60 (March 3, 1982). There, quoting from Gulf States Utilities Co. (River Bend Station, Units 1 and 2), ALAB-444, 6 N.R.C. 760 (1977), the Licensing Board observed that "a party . . . must do more than present what amounts to a check list of items contained in [a technical safety analysis report] or in regulatory guides." 15 NRC at 558-59. Specifically, a "nexus" must be demonstrated between the alleged "generic" deficiency and the particular plant. Id. OCRE has not established such a "nexus." All it has done is identify a number of generic concerns that may or may not be relevant to PNPP. As the Licensing Board already has held, that simply is not enough upon which to base a contention.

16/ The structural encroachments concern was raised by Mr. Humphrey with regard to the Grand Gulf design. OCRE has not even shown that these same design concerns apply to Perry.

concern would be to admit a contention in search of a basis -- something clearly prohibited by the Appeal Board's recent ruling in Catawba. See supra.

As for the second of the two priority items identified by the Staff -- the response of the RHR system to loads produced by the steam condensation phenomenon -- the Staff's resolution (pending further analysis) is for Applicants to commit not to use the RHR system in the steam condensing mode. Perry SER Supplement 1, at 6-1. Applicants have so committed. OCRE has demonstrated absolutely no basis for concluding that this interim response presents any serious safety concern.17/ Thus, as with the structural encroachments concern, OCRE is asking the Licensing Board to admit this concern in the hope that it can develop the requisite "basis and specificity" sometime in the future.

For the same reasons OCRE has failed to meet the Commission's "basis and specificity" requirement, the Motion is deficient under the timeliness factors set forth in 10 C.F.R. § 2.714(a)(1). See supra. Even assuming that OCRE can demonstrate "good cause" for this late filed contention, on the basis of its Motion, OCRE cannot seriously contend that its

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17/ The only potential issue is whether alternative plant safety systems will perform the same function as the RHR system. OCRE has not even suggested, much less demonstrated any basis for concluding, that alternative plant safety systems cannot perform this function.

"participation may reasonably be expected to assist in developing a sound record." 10 C.F.R. § 2.714(a)(1)(iii). After being on notice of the Humphrey concerns since the time of the SER publication in May (see Motion at 7), all OCRE has done to "assist" the Licensing Board is to attach to its Motion a photocopied list of the concerns.18/ OCRE has not even done the elementary analysis of determining which of the concerns remain unresolved and which are at all relevant to Perry. As noted, the Licensing Board has held that, at minimum, intervenors must demonstrate a "nexus" between a submitted contention and the specifics of the plant at issue. See LBP-82-15, 15 N.R.C. at 558-59 (discussed supra). Having failed to even attempt to meet this minimal standard, it is difficult to see how OCRE reasonably can contend that its participation as to this issue would assist in developing a sound record. In fact, OCRE's Motion "shows only a superficial understanding of the issue, . . . and an ignorance of the entire previous history" of the concerns OCRE asks the Licensing Board to admit as contentions. Id. at 558. In addition, as with Sunflower's proferred control systems contention, Contention 22 is a generic contention, see n.15, supra, making OCRE's participation in this proceeding even less likely to assist in the development of a sound record.19/ Id.

18/ The Licensing Board, of course, was already aware of the Humphrey concerns. See July 15, 1982, letter with attachments to Licensing Board regarding the Humphrey concerns.

19/ As the Licensing Board noted with regard to Sunflower's control systems contention, the "nexus" requirement is particu-

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at 559. In sum, for the same reasons the Licensing Board rejected Sunflower's control systems contention, it should reject Contention 22.

Finally, it is indisputable that admission of this contention would broaden and delay this proceeding. Indeed, it is readily apparent that admission of this contention -- in fact sixty-six separate contentions -- would radically transform this proceeding.

III. OCRE CONTENTION 23 (Core Thermal-Hydraulics)

In this proposed contention, OCRE alleges that

> [T]he Applicants' seismic analysis (and the NRC Staff's review of same in the SER) is deficient because this analysis totally neglects the response of the core thermal-hydraulic design to a seismic event. Because the BWR uses a two-phase moderator/coolant, it is inherently susceptible to power excursion transients resulting from events affecting void distribution. An earthquake could cause sloshing of the water in the reactor vessel, thus resulting in void collapse and/or redistribution. See Dr. Richard E. Webb, The Accident Hazards of Nuclear Power Plants (University of Mass., 1976) at 28.

Motion at 3. The proposed contention is woefully untimely, has no basis and involves an impermissible challenge to NRC regulations.

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larly crucial when intervenors are raising generic issues already being considered by the NRC Staff since litigation of such non-plant specific problems "is merely redundant." LBP-82-15, 15 N.R.C. at 559.

A. Timeliness.

OCRE's good cause argument in its entirety reads, "Contention 23 is based on the deficiency of the Staff's analysis in the SER." Motion at 7. In fact, the contention has nothing whatsoever to do with the SER. It stands (and falls) on one phrase taken from a six year old book. OCRE cannot bootstrap its failure to raise this issue at the appropriate time by alleging that the SER should have considered it.

OCRE has not argued that it only recently became aware of the Webb book. Nor could it do so. Reasonable diligence would have disclosed it long ago. Webb himself is no stranger to NRC proceedings.20/

Having failed to raise any Webb issues in a timely manner, OCRE now seeks to inject these issues at this late date. To use the SER as the tool to raise a six year old issue would

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20/ He appeared in the Salem spent fuel pool proceeding, Public Service Electric and Gas Co. (Salem Nuclear Generating Station, Unit 1), LBP-80-27, 12 N.R.C. 435, 452-3 (1980) (Webb testimony characterized as unsupported, ill-organized, difficult to follow; portions were stricken from the record without objection), aff'd ALAB-650, 14 N.R.C. 43, 60-62 (1981); in the Public Service Co. of Oklahoma (Black Fox Station) construction permit proceeding, and in the Commonwealth Edison Co. (Zion Station) spent fuel pool proceeding. His book was cited as the basis for numerous contentions in the Ohio Edison Company (Erie Nuclear Plant) proceeding. Interestingly, in that proceeding he was cited by, and appeared on behalf of, Evelyn Stebbins, who is one of the intervenors with Sunflower Alliance in this proceeding. See Sunflower Alliance, Inc. et al., Petition for Leave to Intervene, dated March 15, 1981.

wipe out the Commission's requirements for filing contentions at the start of a proceeding. 10 C.F.R. § 2.714(b). An intervenor or petitioner would merely await the publication of the last Staff document to be issued. Then, he would submit a contention, alleging that it had not been dealt with in the SER (or FES). The fact that the substance of the contention was (or should have been) known to the intervenor for years would be irrelevant. So, too, would be any consideration of whether the intervenor had any reason to expect that the issue would be treated in the SER. OCRE had no reason to expect that the Webb allegation would be treated in the SER. This process would be in direct conflict with the rule established by the Commission in Wisconsin Electric Power Co., et al. (Koshkonong Nuclear Power Plant, Units 1 and 2), CLI-74-45, 8 A.E.C. 928 (1974). There, the Commission unequivocally rejected intervenors' argument that they should not have to file their contentions until after issuance of the DES and the SER. Koshkonong thus expressly prohibits what OCRE here attempts to do through the more indirect route of asserting that late safety contentions always can be justified through a citation to a recently issued SER. Cf. Memorandum and Order (Concerning Sunflower's Late-Filed Radiation-Dose Contention), September 15, 1982, slip op. at 2.

B. Basis and Specificity

The only support which OCRE supplies for its allegation that an earthquake can "cause sloshing of the water in the reactor vessel, thus resulting in void collapse and/or redistribution" followed by a "power excursion transient" is its reference to page 28 of the Webb book. The only relevant statement in the book reads as follows:

> Conceivable examples of autocatalysis are ... (5) sloshing of water coolant in the core of a BWR at power, which could displace steam bubbles and raise the reactivity rapidly -- a process which could occur in an earthquake or in any accident in which the reactor vessel is bounced or shaken. . . .

Webb at 28. Webb provides no references, citations or analyses in support of his theory that "sloshing" could occur from an earthquake, or that if it did occur it could "displace steam bubbles and raise the reactivity rapidly". Applicants are aware of no basis for this claim.21/

There are only two ways to collapse voids in a BWR's core: increasing the pressure in the reactor pressure vessel or

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21/ The lack of basis for Webb's claim is not surprising. As stated in a review of his book by the head of the Physics Department of DePauw University,

> Unfortunately, this reviewer does not believe that the author's assessment of the hazards [of nuclear reactors] is sufficiently valid for the book to be an important contribution to the subject. . . .

Nuclear Technology, vol. 33, mid-April 1977 at 237-8.

increasing core flow. There is no inherent reason why an earthquake would cause void collapse, since pressure and flow are constant. Furthermore, Applicants have already analyzed transients involving reactor pressure increase22/ and increased core flow,23/ with no indication of the void collapse and reactivity increase hypothesized by Webb.

Nor is there any conceivable way for Webb's "sloshing" to displace steam bubbles within the confines of the reactor core. The steam separators are close enough to the top of the water level to dampen "sloshing", even if it should occur. See FSAR Fig. 5.3-6. And with the top of the active fuel some 200 inches below normal vessel water level, FSAR Fig. 5.3-7, no "sloshing", even if it were to occur, could possibly uncover the core.

C. The Contention Involves An Impermissible Rule Challenge.

OCRE suggests that the analysis of "core thermal-hydraulic response" which it believes justified by the Webb book should "be based on an earthquake of greater severity than the SSE". Motion at 3. By its own terms, this constitutes a challenge to

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22/ For example, turbine trip (FSAR § 15.2.3) and main steam line isolation valve closure (FSAR § 15.2.4). These transients cause greater reactor pressure increases than accidents such as a LOCA which cause reactor depressurization.

23/ For example, the recirculation flow control failure with increasing flow (FSAR § 15.4.5).

the Commission's regulations. Under 10 C.F.R Part 100, Appendix A, the Safe Shutdown Earthquake ("SSE") is the most severe seismic event to which the plant need be evaluated. See, e.g., North Anna Environmental Coalition v. NRC, 533 F.2d 655, 660, 665 (D.C. Cir. 1976); cf. Southern California Edison Co., et al. (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-81-33, 14 N.R.C. 1091 (1981); see generally, Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-644, 13 N.R.C. 903 (1981). OCRE has not petitioned for waiver or exception under 10 C.F.R. § 2.758(b), and there is nothing to suggest that "special circumstances" exist in this "particular proceeding" that might provide the basis for a waiver or exception. See LBP-81-57, 14 N.R.C. 1037 (Licensing Board's denial of OCRE's § 2.758 motion concerning electromagnetic pulse).

OCRE cannot benefit from the recommendation of the Advisory Committee on Reactor Safeguards to the Commission that

> the Applicant and the NRC Staff conduct studies to evaluate the margins available to accomplish safe shutdown, including long-term heat removal, following an earthquake of somewhat greater severity and lower likelihood than the safe shutdown earthquake.

ACRS Letter at 2, in Supp. 1 to Perry SER, at F-2 (August, 1982). The ACRS operates under a statutory charter which directs it to review facility license applications, advise the Commission on their safety, and review the adequacy of proposed

safety standards. Section 29, Atomic Energy Act of 1954, 42 U.S.C. § 2039. The ACRS' advice is not restrained by current NRC regulations. Indeed one of its important functions is to advise the Commission on changes in regulatory standards which might be desirable.24/ The Licensing Board, on the other hand, is charged with conducting this proceeding in accordance with NRC rules, including 10 C.F.R. Part 100. OCRE's assertion constitutes such a challenge.

For all these reasons, the contention must be denied.

IV. OCRE CONTENTION 24 (In-core Thermocouples)

OCRE's proposed contention 24 states:

> Applicants should conform to the requirements of Regulatory Guide 1.97, Revision 2, and TMI Action Plan item II.F.2 by installing in-core thermocouples at Perry. In-core thermocouples provide an indication of inadequate core cooling (ICC) and are a redundant and diverse means by which to detect reactor coolant level.

Motion at 3-4. The proposed contention should be denied based upon OCRE's complete failure to show good cause for its untimely filing and also for its lack of basis.

24/ In this context, it should be pointed out that the ACRS' interest in seismic issues is a generic one. This is clear from the discussion of this issue at the ACRS meeting on Perry (ACRS Tr. at 60-63, July 8, 1982) as well as the similar statements set forth in other recent ACRS letters. *See*, *e.g.*, ACRS reports on Clinch River Breeder Reactor (July 13, 1982), Midland (June 8, 1982), Virgil Summer (March 18, 1981). *See* ACRS Subcommittee on Extreme External Phenomena, Transcript of August 11, 1982 meeting.

A. Timeliness.

OCRE's timeliness argument in its entirety reads as follows:

> Contention 24 was filed at this time because prior to the issuance of the SER, OCRE assumed that in-core thermocouples would be required at Perry. The Staff required them at Grand Gulf (Grand Gulf SER, NUREG-0831 at 22-22).

Motion at 7. OCRE's admission that it was aware of the in-core thermocouple issue before the SER was issued should, by itself, be enough to show OCRE's lack of good cause. The issue has existed for at least two years and OCRE has had actual knowledge of it since at least December 1981. In fact, OCRE had in its possession for some six months the very documents on which it now relies. OCRE's delay in proposing this contention is inexcusable.

The means to detect inadequate core cooling was one of the earliest issues to be raised after the Three Mile Island accident. The TMI-2 Lessons Learned Task Force recommended that licensees provide assurance that instrumentation exists to provide an "unambiguous, easy-to-interpret indication of inadequate core cooling". NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (July 1979), § 2.1.3.b, pp. 8, A-11 - A-12. The issue was further discussed in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident" (May 1980) (see Task II.F.2).

By November 1980, the inadequate core cooling issue had been further refined and clarified. The Staff set forth its position as follows:

> Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980), at II.F.2-1. NUREG-0737 did not explicitly require installation of in-core thermocouples in BWR's, although it did provide criteria for their design and qualification in PWR's. Id. at II.F.2-5. Since OCRE's Motion cites to TMI Task Action Plan item II.F.2, it must be assumed that OCRE was aware of NUREG-0737.

Finally, in December 1980, the Staff issued Revision 2 to Regulatory Guide 1.97 which announced a regulatory position that boiling water reactors should have in-core thermocouples to provide diverse indication of water level and to monitor core cooling. Thus, there have been numerous NRC pronouncements on the matter of inadequate core cooling and in-core thermocouples, all of which were readily available to OCRE.

OCRE has had actual knowledge of (and apparent interest in) in-core thermocouples dating back at least as far as December 1981. By letter dated December 22, 1981, Susan Hiatt filed a Freedom of Information Act ("FOIA") request with the NRC. Among other items, Ms. Hiatt requested "any documents concerning the use of in-core thermocouples in BWRs, particularly in BWR/6 reactors." As stated in the letter,

> Upon this requester's review and analysis of the requested documents, they will be used by intervenors in this case...

On February 22, 1982, the NRC responded to Ms. Hiatt's FOIA request, supplying four documents. These four documents (together with the Perry SER and the Webb book) make up the only bases cited by OCRE for the proposed contention. Thus, OCRE had in its possession the very documents on which it now relies for some six months before the contention was filed. OCRE made no effort to acknowledge or explain this fact in its motion. While OCRE's use of FOIA requests may show resourcefulness (it has filed at least 13 such requests since March 1981), it cannot hide behind the FOIA in establishing good cause. OCRE could easily have submitted this same contention a year ago or six months ago. It cannot justify its inaction.

OCRE claims to find good cause for its lateness in its "assumption" that the Staff would require in-core thermocouples for Perry because it required them for the Grand Gulf facility.

(Mississippi Power & Light Co. (Grand (Grand Gulf Nuclear Station, Units 1 and 2), Docket Nos. 50-416, 50-417). OCRE has provided no support for this self-serving argument. It cannot withstand scrutiny. In the Grand Gulf Safety Evaluation Report, (NUREG-0831, September 1981, at 22-22 - 22-23), as cited by OCRE, the NRC Staff stated that it would require in-core thermocouples, and would condition the operating license accordingly. However, OCRE failed to acknowledge that, subsequent to September 1981, the Staff changed its position in Grand Gulf. Supplement 2 to the Grand Gulf SER, issued June 1982, p. 22-6, modified the SER requirement and instead committed the utility to submit a report on inadequate core cooling instrumentation generally and to implement the Staff's requirements after the Staff completed its review of the report. And, indeed, the Grand Gulf operating license contains a condition specifically incorporating the position set forth in the SER Supplement 2. See Operating License No. NPF-13, § 2.C(44)(f), dated June 16, 1982. Since OCRE has apparently been following developments in the Grand Gulf docket,25/ it is not unreasonable to expect that OCRE was or should have been aware of these developments.

25/ OCRE's proposed contention 17 was based upon the Final Environmental Statement for Grand Gulf. See "Ohio Citizens for Responsible Energy Motion for Leave to File Its Contentions 17, 18, and 19", dated April 22, 1982.

Nor was this revised Staff position unique to Grand Gulf (or Perry). Safety evaluation reports for the Clinton, WNP 2, and LaSalle facilities, issued in February, March and April 1982 respectively, required generic inadequate core cooling reports, instead of installation of in-core thermocouples.26/ OCRE thus should have had ample notice that the Staff's September 1981 position in Grand Gulf was subject to change.

Certainly, OCRE had or should have had actual knowledge of Applicants' position on in-core thermocouples. In a letter dated August 12, 1981 from the NRC Staff to Applicants (a copy of which was served on OCRE), the Staff summarized Applicants' and its positions on in-core thermocouples:

> The applicant takes the BWR Owner's Group position that no additional instrumentation is needed. However, the Regulatory Guide 1.97 requires that the incore thermocouples be installed. Therefore, we require that the applicant commit to install incore thermocouples in accordance with Regulatory Guide 1.97 and to provide the documentation required by NUREG-0737 Section II.F.2 for Staff review.

Applicants' response to the August 12, 1981 letter was sent to the NRC by letter dated October 1, 1981. The response stated, in part, that

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26/ NUREG-0853, Safety Evaluation Report for Illinois Power Company's Clinton facility, § 4.4.2.2, 4.4.2.3 (February, 1982); NUREG-0519, Supp. 2 to Safety Evaluation Report for Commonwealth Edison Company's LaSalle facility, Table 22.1 (February, 1982) (see also Operating License No. NPF-11, § 2.C(20)(i), issued April 17, 1982); NUREG-0892, Safety Evaluation Report for Washington Public Power Supply System's WNP-2 § 4.4.7 (March 1982).

> Cleveland Electric Illuminating Company supports the BWR Owner's group position that no additional instrumentation is needed to monitor inadequate core cooling at the Perry Nuclear Power Plant. Further, we feel that incore thermocouples may provide the operator with ambiguous information in the event of an accident.

The same conclusion appeared in Amendment 5 to the Perry FSAR, filed in November 3, 1981. FSAR Q&R p. 4.4-31. See also Amendment 7 to FSAR, filed May 27, 1982, FSAR App. 1A, p. 1A-59. All of these documents were readily available to OCRE.

In short, there can be no doubt that OCRE has been aware of the generic issue and of Applicants' specific position for a long time. OCRE has provided no reason why it should be permitted to rely on a position taken by the Staff in another docket a year ago, particularly when the Staff publicly abandoned that position at least seven months ago.

As discussed above, OCRE's showing on the remaining good cause factors is, at best, pro forma.

B. Basis and Specificity.

OCRE's proposed contention starts off incorrectly by referring to "the requirements of Regulatory Guide 1.97, Revision 2, and TMI Action Plan item II.F.2." Neither of these establish requirements for in-core thermocouples and therefore neither can provide a basis for the contention. As noted above, Task Action Plan item II.F.2 does not require or even recommend in-core thermocouples for BWR's. Regulatory Guides such as Revision 2 of Regulatory Guide 1.97 are, of course, not requirements. As each Regulatory Guide states:

> Regulatory Guides are not substitutes for regulations, and compliance with them is not required.

The role of Regulatory Guides as guidance, not requirements, has been emphasized by Appeal Boards27/ and by the courts.28/ Thus, OCRE's citations to these documents cannot provide a basis for the contention.

27/ As stated in Gulf States Utilities Co. (River Bend Station, Units 1 and 2), ALAB-444, 6 N.R.C. 760, 772-3 (1977)(citations omitted):

> For their part, and as their title suggests, regulatory guides are issued for the basic purpose of providing guidance to applicants with respect to, *inter alia*, acceptable modes of conforming to specific regulatory requirements. But they are not regulations *per se* and are not entitled to be treated as such; they need not be followed by applicants; and they do not purport to represent that they set forth the only satisfactory method of meeting a specific regulatory requirement. Indeed, quite the contrary is true; the cover page of each guide states that
>
> > Methods and solutions different from those set out in guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.
>
> In other words, a guide sets forth one, but not necessarily the only, method which an applicant may choose to employ in order to conform to a regulatory standard. While the staff will accept such a method, an applicant is not precluded from utilizing some other method which it can demonstrate is appropriate in the particular case. Nor are other parties precluded from demonstrating that the prescribed method is inadequate in the particular circumstances of the case.

28/ Porter County Chapter of the Izaak Walton League of America, Inc. v. AEC, 533 F.2d 1011, 1016 (7th Cir.), *cert. den.* 429 U.S. 945 (1976).

Nor do OCRE's other arguments provide a basis. OCRE first sets up a straw man. According to OCRE, the BWR Owner's Group believes that in-core thermocouples have no advantage in monitoring inadequate core cooling or reactor water level because the thermocouples would have an excessive time lag. OCRE then tries to knock down its straw man by citing a Battelle Laboratories calculation showing a time lag of only 1 to 1-1/2 minutes. As shown by the very documents cited by OCRE, time lag is not a prime factor in opting against in-core thermocouples. The General Electric report cited by OCRE, "Evaluation of the Need for BWR Thermocouples," shows that in-core thermocouples are not necessary because the BWR has highly reliable and redundant reactor water level measuring systems, in-core thermocouples would probably be erratically indicating lower temperatures due to subcooling effects of ECCS, monitoring of reactor water level and water makeup system performance provides the capability of monitoring core cooling, as well as engineering and maintenance considerations. Furthermore, the Levy document29/ cited by OCRE for its discussion of the "excessive time constant" [sic, "constraint?"], Motion at 4, provides a detailed explanation of the differences between the Battelle calculation and the Levy

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29/ Gillis, Hench, Adams, Eddleman & Beckett, "Thermal Analyses of In-Core Thermocouples in Boiling Water Reactors" (S. Levy, Incorporated, November 1981).

calculation. See Levy report, § VI. OCRE's failure to acknowledge the existence of this explanation in a document which it cites is at best disingenuous. Its failure to provide a rationale for challenging the Levy results should be dispositive.

OCRE then asserts that "GE does admit that thermocouples could be used in one situation: loss of coolant inventory with no water makeup systems available". Motion at 4. OCRE implies that this supports the need for thermocouples. Yet, even here, OCRE mischaracterizes the very report it relies upon. The GE report (§ 2.1) states that the situation described by OCRE is the "only ... condition for the BWR that incore temperature measurement would provide unambiguous and definitive information"; even in this situation,

> the BWR not only has other diverse indication of such water level but also unique symptom based procedures to direct the operator whether he does or does not have water level indication.

Id. at § 2.2. OCRE has provided no basis for challenging these conclusions. Thus, this "narrow case of very low probability", id. at § 1.0, cannot, therefore, support the need for in-core thermocouples.

OCRE's final allegation is that in-core thermocouples can provide "vital information" in the event of a fuel bundle blockage accident. OCRE acknowledges that the GE report in its possession analyzes this accident, but asserts that "several

key assumptions" which support GE's "conclusion that thermocouples are of no value" are "arbitrary and unproven". Motion at 4-5. OCRE cites pages 59-61 of the Webb book as support for this latter assertion. Nowhere on these pages (or, to our knowledge, elsewhere in the book) are in-core thermocouples mentioned. The Webb reference is simply irrelevant. Nor does OCRE even identify the "key assumptions" to which it refers. The portion of the GE report relied on by OCRE (App. B, "Detection of Propagating Core Damage") shows that in-core thermocouples would not detect a fuel bundle blockage accident, even where there had already been 5-10% core damage and an uncoolable geometry and the instrumentation was directly adjacent to the bundles in the damaged core region.30/ The cited portions of the Webb book do not deal with any of these facts.

OCRE, therefore, has failed to supply a basis for its claims.

V. OCRE CONTENTION 25 (Steam Erosion)

Proposed Contention 25 states that:

> OCRE contends that Applicants are not prepared to prevent, discover, assess, and mitigate the effects of steam erosion on components of PNPP which will be subjected to steam flow. Steam erosion has been

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30/ Since Regulatory Guide 1.97, Rev. 2, only recommends four thermocouples in each quadrant of the core (p. 1.97-8 n.2), it is unlikely that an in-core thermocouple (even if installed) would be near any given fuel bundle.

> identified as the cause of recent failures of valves and piping (MSIVs and turbine exhaust lines: see NRC Information Notices 82-22 and 82-23). The NRC Staff has identified Applicants' lack of an inservice testing program for pumps and valves and leak testing of valves as an open item in Section 3.9.6 in the SER.

Motion at 5.

The two IE Information Notices cited in the contention are dated July 9 and 16, 1982, respectively. Although some of the events (and Licensee Event Reports) underlying the Information Notices date back as far as 1979, the Notices appear to be the first generic statement on the issue. Applicants thus do not question the timeliness aspect of OCRE's good cause showing.

Nonetheless, OCRE has still failed to justify admission of the contention. An examination of each of the three cited references shows that the contention lacks any basis. IE Information Notice 82-22 deals with a pipe rupture in a turbine exhaust line caused by steam erosion and four other steamline failures apparently resulting from steam erosion. It did not call for any immediate action. Pursuant to 10 C.F.R. § 50.55a(g), Applicants must conduct specified inservice inspection programs for ASME Class 1, 2 and 3 components, including pumps, valves, and piping. The inspection programs must meet the requirements of ASME Section XI. As stated in the SER, at 6-37, Applicants have not yet submitted their inservice inspection program; the Staff will review the program

before the first refueling outage. (Under 10 C.F.R. § 50.55a(g)(4)(i), the applicable edition of the ASME cannot be determined until 12 months before issuance of the operating license.) Applicants' inspection program will include all piping subject to steam erosion, including those identified in the Information Notices. Since OCRE has not alleged that an inspection program meeting ASME requirements is inadequate to detect steam erosion, there is no basis for a contention on the inadequacy of Applicants' efforts in this area. Nor can OCRE rely on the fact that the inservice inspection program has yet to be submitted. A contention based on such an allegation would be the type of conditional contention prohibited by the Appeal Board. Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), ALAB-687 (August 19, 1982).

IE Information Notice 82-23 does not specifically deal with steam erosion. Its subject is main steam isolation valve ("MSIV") leakage from all causes. The notice reports on a survey of MSIV performance at BWRs. In a few cases, the observed MSIV leakage is attributed to "steam erosion". The Notice states:

> This information indicates that some MSIVs may not adequately limit release of radioactivity to the environment if called upon to do so. NRC is considering the need for improved MSIV maintenance, more frequent MSIV testing or installation of leakage control systems. (emphasis added)

Applicants have already committed to an MSIV leakage control system. That system is described in detail in the FSAR § 6.7. The Staff has reviewed and approved it. SER § 6.7. OCRE has provided no basis for questioning the adequacy of the Perry MSIV leakage control system; indeed, it does not even acknowledge the system's existence. Thus, IE Information Notice 82-23 provides no basis for the contention.

OCRE's final reference is § 3.9.6 of the SER, entitled "Inservice Testing of Pumps and Valves." This section notes that Applicants have not yet submitted their inservice testing program for pumps and valves and that the Staff's review of the program will appear in a supplement to the SER. OCRE has offered no explanation of the relevance of this program to steam erosion. Neither of the Information Notices relates to pumps; the only relevant valves are MSIVs and, as discussed above, possible leakage from steam erosion or any other cause has been provided for by a system which NRC has already reviewed and approved.

For these reasons, OCRE has failed to set forth an adequate basis for this contention.

VI. <u>OCRE CONTENTION 26 (Control Room Fire Suppression)</u>

OCRE's final late-filed contention states:

> OCRE contends that all advantages and disadvantages of [carbon dioxide and Halon 1301 as control room fire suppressants] should be thoroughly evaluated before choosing a particular system, especially in regard to toxicity.

Here, too, OCRE is raising an issue which has been on the docket since the construction permit stage, and has failed to supply an adequate basis.

A. Timeliness.

OCRE claims that its "first notice of the concerns identified in" Contention 26 was the SER. Motion at 7. The claim cannot withstand scrutiny. The control room fire suppression system, and specifically the choice between carbon dioxide and Halon, has been on the Perry docket since the construction permit stage. If OCRE was not aware of the issue, it can only be attributed to its failure to exercise reasonable diligence on this matter.

Applicants initially intended to install a Halon 1301 fire suppression system for the Perry control rooms. In the Construction Permit Stage SER, issued in July 1974, the Staff raised questions about the reported detrimental effects of Halon on operating personnel, as well as concerns with artificial aging or disabling of safety-related charcoal bed. Construction Permit Stage SER § 9.4.1. In response to the Staff's concerns, Applicants in Amendment 17 to the Preliminary Safety Analysis Report, issued July 15, 1974, proposed to replace the Halon system with a carbon dioxide fire suppression system. The Staff reviewed the CO_2 system and approved it.

> We find that the proposed use of CO_2 in the fire fighting system of the control room complex will not adversely decrease the habitability of the control room since CO_2 is stable and will not produce any toxic decomposition products when used as a fire extinguishing agent. We conclude, therefore, based on our review, that the proposed design criteria for the fire fighting system in the control room complex are acceptable.

Supplement No. 1 to the Construction Permit Stage SER, § 9.4.1, issued December 1974.

OCRE cannot therefore argue that it took the issuance of the Operating License Stage SER to alert it to the advantages and disadvantages of CO_2 and Halon fire suppression systems. These have been discussed on the Perry docket since 1974. All that the SER did was to identify areas where the Staff sought additional information. This cannot possibly constitute good cause.

OCRE has cited to no new information on which to base its contention. The only reference other than the SER is National Fire Prevention Association ("NFPA") Standard 12. This standard sets forth general information, hazards and safety requirements for many types of fire suppression systems. The same Standard was cited in the Construction Permit Stage SER at 9-15, Amendment 17 to the PSAR, Supplement 1 to the Construction Permit Stage SER at 54, and the FSAR, § 9.5.1.2.3 at 9.5-4. Thus, OCRE cannot use NFPA 12 to support its untimeliness.

As to the remaining good cause factors, OCRE's pro forma showing, discussed above, is little more than a restatement of the regulatory requirements. OCRE has failed to meet its timeliness burden.

B. Basis and Specificity.

Proposed Contention 26 asks for an evaluation of all the disadvantages and advantages of the CO_2 and Halon systems. This does not even constitute a contention. It does not advocate a position. It does not assert that Applicants' design is unsafe. It merely asks for more study. Unless OCRE can demonstrate, with basis and specificity, that Applicants' design is somehow inadequate, there is nothing to litigate and therefore no contention.

OCRE points out only two "disadvantages" of CO_2 which it argues "should be thoroughly evaluated". These two, taken from NFPA 12, are reduced visibility and possible oxygen deficiency. Motion at 6. Yet OCRE nowhere provides a nexus between these generic concerns and the Perry design. OCRE fails to acknowledge the significant design information concerning Perry which is on the docket and which responds to these "disadvantages". See, e.g., FSAR § 9.5.1.2; letter from Applicants to NRC dated March 12, 1982; letter from Applicants to NRC dated May 19, 1982. Most recently, Applicants responded to additional NRC Staff questions by letter dated August 31,

1982. Something more must be provided than picking two generic concerns out of a generic standard without either addressing the methods outlined in the standard to resolve those concerns (OCRE ignores the safety requirements for CO_2 systems set by NFPA 12) or explaining why the particular design at Perry is susceptible to these concerns. OCRE has failed to link its allegations of hazard with the Perry design. Therefore, it has failed to provide a basis for the contention.

The contention also sets forth a list of potential hazards from Halon. However, these allegations are simply irrelevant. Applicants do not intend to utilize a Halon system at Perry.

For all these reasons the contention should be denied.

VII. CONCLUSION

For all the reasons set forth above, Applicants request that OCRE's late-filed proposed Contentions 21-26 not be admitted as issues in this proceeding and that OCRE's Motion be denied.

Respectfully submitted,

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had not been proven acceptable as far as the integrity of that unit. I think since that time there is a little better understanding of this and I think that now it would be most likely accepted by the Committee, so that in a sense, those plants are all pretty much on an equal footing.

Then of course now, we've had the concerns raised by Mr. Humphrey and the question is, would this change the view of the -- first the Grand Gulf subcommittee and second, the view of the full committee? And I think that what this subcommittee can do is forward it's views to both of those -- the Grand Gulf subcommittee and to the full committee.

Now I'm going to call on the others for their opinions but not to influence them, give my own first.

My feeling is, that I see no reason why these plants and Grand Gulf in particular cannot go ahead and receive a full power operating license. There is nothing that's come forward since the reports that I mentioned have been prepared that would change my view on this question.

Now, it's true that there were some concerns raised by Mr. Humphrey that have occupied the Staff and the applicant. For example, things like encroachment. I'm very optimistic about that. I don't think it's going to make any difference to the safety of the plant, but

that's just an optimistic forecast. I'm sure that the Staff will investigate this. And in my view, recommending that Grand Gulf in particular get a full power license doesn't mean that the Staff won't do some more work. I am sure they will but like other items in an application, there are usually many points, some of them generic that they have to straighten out to their own satisfaction. They have a responsibility in that direction and I don't see where there's anything new which really changes this picture. Maybe it's painted on a little broader horizon, but that to me is not a particularly essential item. So that's my view. I would recommend that the Grand Gulf subcommittee and the full committee accept this situation and proceed as usual with the licensing of the plant. We've already done that for Perry and Clinton. It's just a matter of Grand Gulf and that's a relatively small step in my mind from the 4% approval to the full power.

Now, there's a spectrum of people up here and they may give you other views so let me go down the table. Spence, would you like to --

DR. BUSH: Sure. Dr. Plesset knows that I don't influence that easily so his prior comments haven't really introduced a bias since I've already written my comments.

I would hope that the issues could be resolved

generically rather than case by case for a variety of reasons, certainly Staff load and I think also as you see in the tenor of my remarks, I don't consider them as having that major an impact that we need to overload the industry.

With regard to a degradation of safety function which I think is the important thing, I see no significant losses in the Grand Gulf design and I suspect this to be true for the STRIDE design but I would reserve final judgement pending a little more information. Quite frankly, I'm not that familiar with the STRIDE design.

The preceding comments consider the effects of loads rather than the subtleties and thermo-hydraulics since I don't consider myself very expert in that area.

I have no reservations in permitting Grand Gulf to go to full power on the basis of these issues that we've been discussing. Obviously other issues may control this decision. That's been the case in other plants and I think that the decision of a 5% license on the first plant was a very logical one.

At this time I reserve judgement on the other Mark III designs simply because of -- I haven't had a chance to look at them and there may be some subtleties that would affect the plant specific areas, though I suspect this may not be the case.

With regard to Mark I's and II's, I feel that most of the issues are either inapplicable or insignificant with regard to safety margins. There may be a few applicable issues that need further examination and I hope again that these could be generic.

I go back to my original plea and feel that some level of instrumentations that measure a critical pressure temperature, stresses or strains could be valuable in a Mark III design and possibly in a Mark II to confirm the loads are comfortably within the design envelope which would hopefully minimize the continuous discourse on design margins in this particular area.

DR. PLESSET: Thank you, Spence. I didn't want to imply that you would be at all malleable. We know otherwise. A good metallurgical man.

Before I call on Dr. Schrock, I should mention that Mr. Ray, a committee member, indicated his concurrence with the views that I've expressed regarding this situation and Dr. Zudans did likewise. I'm going to make life easy for Virgil by letting him have the microphone.

DR. SCHROCK: I'll keep it only briefly.

With regard to the Humphrey Issues, my view is that I heard nothing that would lead me to have any misgivings about proceeding with a full power license for Grand Gulf.

I think that many of the things that were discussed here were certainly worth discussing. I have some severe reservations about whether we have set an unreasonable precedence for raising issues of this level of importance in the way that they eventually evolved in a meeting of the subcommittee of the ACRS. I think that there is some risk in our proceedings here in following this path.

With regard to the responses from the utilities and from the General Electric Company and Bechtel, I think it appears to me that the answers that are being sought will be obtained in a satisfactory way. I have no real concern that there will be serious questions remaining after all of the things that we've heard to be done will be accomplished.

One point I would make with regard to the assurance with respect to design margins is that I don't like to see design margins essentially misrepresented. I don't mean to say that they were intentionally misrepresented but I don't like to see them carelessly misrepresented. I think the question of what a design margin is is a serious question and it should be dealt with very carefully.

Frequently, usually, I think we do not know very well what our design margins are and to overstate them

is not a good practice in general. With regard to the encroachments as a specific issue, it seems to me that we have had a lot of controversy in the hydrodynamics of the pool responses, and that it would be very desirable to have some of these calculations confirmed by other than the designer of the system and for that reason, I was pleased to hear that there will be some additional supporting calculations submitted to the staff that will be done by other AE's using different codes.

That concludes my comment.

DR. PLESSET: Thank you, Virgil. Mr. Etherington?

DR. ETHERINGTON: I think everyone is addressing these concerns in a responsible manner and I see nothing in the unresolved items that would warrant withholding a license, full power operating license.

DR. PLESSET: Dr. Garlid?

DR. GARLID: Well, I think the issues that were raised were real ones but were for the most part second order with respect to safety. MP&L has been responsive to the concerns that were raised, and that the Staff has developed a reasonable plan, although if anything it's on the conservative side of how to deal with them.

I don't think the issues should cause any delay and finally, I think the question of interface is whether

they are interfaces between organizations or interfaces of problems between one discipline and another, that these are generic issues and not unique to these plants.

DR. PLESSET: Thank you. Jesse?

DR. EBERSOLE: Yes. May I ask G.E. a question about the containment structural design and the limitations on it? It's always concrete, I take it for N16 and other shielding purposes. Does it have a membrane liner on either side? Do you give freedom to the AE's to put liner skin on the structural wall? Do you know?

MR. DAVIS: This is Mac Davis from General Electric. We place no requirements at all on the AE as to whether he can or cannot put liners on.

DR. EBERSOLE: Are any of these equipped with liners? Membranes on either side?

MR. McGAUGHY: We have what's -- well it's not a Q type liner. We have concrete steel forms that are welded together. In essence, a liner but it's not a --

DR. EBERSOLE: Is it on both sides?

MR. McGAUGHY: It's on the inside.

DR. EBERSOLE: THen I would only ask one question. When that particular wall is subjected to negative pressure and therefore gas in-leakage, how do you retain that liner in the structural context? How do you keep it from peeling off?

MR. McGAUGHY: It's designed to 3PSID, to withstand that pressure.

DR. EBERSOLE: In otherwords, it's anchored at sufficient intervals to --

MR. McGAUGHY: Yes. See, well, it's not Q, it's got to be seismic. We've got to show that it won't fall off in an earthquake and it will withstand the amount of 3PSID.

DR. EBERSOLE: I'm talking about due to in-leakage from the high pressure side. How do you keep it from peeling off and flying inward into the containment?

MR. McGAUGHY: It has anchors on the back of it, into the concrete.

DR. EBERSOLE: So it's periodically anchored?

MR. McGAUGHY: That's correct.

DR. EBERSOLE: On the inner face.

MR. McGAUGHY: Yes, sir.

DR. EBERSOLE: Is it designed to permit atmospheric penetration and to carry the structural load at the liner face? On the inner face of the liner, next to the concrete?

MR. McGAUGHY: I'm not sure I understand the question.

DR. EBERSOLE: Okay. The gas, the atmosphere on a reverse pressure mode will be carried inward through the leakage of the concrete and the pressure gradient will

occur on the liner. Are you with me?

MR. McGAUGHY: Yes, I think so.

DR. EBERSOLE: Okay, the pressure gradient being almost all contained on the liner, how do you support it against the buckling load?

Somebody is holding their hand up.

MR. BROSE: I'm Tom Brose from Bechtel in Los Angeles. The generic Bechtel design of a liner plate is not a structural member.

DR. EBERSOLE: That's what I was afraid of. So now what's going to keep it from flying all over the place if you apply an external atmospheric load on it.

MR. BROSE: It is anchored to the concrete containment by three by two by quarter-inch channels spaced every fifteen inches. It's designed as a membrane only. Okay, your question as to the differential pressure across the liner through diffusion through the concrete would not occur because the liner is continous to the outside surface, and by that the liner is attached to the penetration -- you wouldn't -- I don't foresee a differential pressure occurring across the liner.

DR. EBERSOLE: You do not put any pressure in your design against the exterior face of the liner, that is the face between the the concrete and the steel?

MR. BROSE: No.

DR. EBERSOLE: You don't look at the permeation of atmospheric pressure against that face?

MR. BROSE: No, but the liner itself is designed for a negative load due to the other new loads which create such loads on the liner and it has the capability on Grand Gulf -- I can't give a specific number, but whatever the negative pressure is from SRV.

MR. McGAUGHY: He's not -- we're talking about inside the drywell.

MR. BROSE: No, no, he's talking about containment.

MR. McGAUGHY: I'm sorry.

MR. BROSE: He's talking about the containment liner and the containment liner is capable of withstanding the negative pressure from an SRV discharge which would suck on the liner in the order of magnitude --

MR. McGAUGHY: At least 5PSI.

DR. EBERSOLE: You follow me -- I'm just looking at the anchor mode to the concrete and hoping it won't scallop and come off.

MR. BROSE: It's designed for a suction load.

DR. EBERSOLE: In otherwords, you do then put atmospheric pressure on the back face?

MR. BROSE: Yes.

DR. EBERSOLE: You have to.

MR. McGAUGHY: That's the only way you can get it, I guess.

DR. EBERSOLE: And you put what, 5PSI?

MR. BROSE: Whatever Grand Gulf's design criteria are.

MR. McGAUGHY: The negative loads from the SRV actuation are at least five.

DR. EBERSOLE: So you're anchored at sufficient intervals per square foot to hold it together.

MR. BROSE: Yes.

DR. EBERSOLE: Okay, that's one question I had. Other than that, I have no reservations, Dr. Plesset about this containment. If I have any reservations about thermo-hydraulic loads in other contexts such as the drive, controller drive units -- sorry, not the CRU's but the tubes and instrumentation and other thermo-hydraulic loads that may be imposed on safety equipment which we haven't pinpointed here as we have the HCU's on this floor. But those will come up in another context rather than a containment context.

DR. PLESSET: Thank you, Jesse. Arthur?

DR. CATTON: I've been involved, I guess, with the Mark I, II and III in the suppression pool loads and so forth. And it's my view that the Humphrey Issues are receiving far more attention than they deserve by NRC,

G.E., and MP&L. I have no reservations regarding the Mark III containment scheme. I have some residual questions that I've raised through the two day period. I've had some promises with respect to experimental data and answers and I'll just await receiving them.

DR. PLESSET: Thank you. Let me -- do you want to make another comment? I think we've heard from the wise men at this table and I'm not including myself in that category, but you see there's kind of a consensus here. I'd like to follow up on a couple of points that were made by Dr. Bush, Dr. Catton that one has only a certain amount of resource at one's disposal and one has to use this wisely. The question is, are you using these resources for the most efficiency for safety? And it's been indicated or hinted at that maybe you aren't by paying so much attention to these particular issues that we've been talking about the past two days. And this disturbs me as well as the other members up here, that you may be not helping safety by disregarding other items and concentrating on these and this I think, you have to think about and I think along this same line, the kind of a cost benefit approach to safety.

Dr. Bush mentioned his distress at Mark I and II being drawn into this and this seemed to me particularly non-productive. We indicated it was not productive for

Mark III's but to get the Mark I's and II's in it is really a little bit well, more than unfortunate and I wanted to stress those points to you, Jack in this connection.

Now, unless the people up here at the table want to make more comments, I'd be glad to have you respond to what we've just been saying. Jack or Dr. Butler, either one. Both maybe.

MR. KUDRICK: We appreciate your frankness in your positive comments relative to the concerns that have been raised. Since we have been informed of the Humphrey concerns, we have taken about trying to resolve those as quickly as possible and hopefully we have given the subcommittee the impression that we do not feel that the majority of the concerns are significant safety issues, and I hope that we have made that point earlier yesterday. We have, however, believed that there are one or two items that deserve our attention and that based on the information that we've gotten, we believe that we will be getting a satisfactory response. However, we will be awaiting judgement until we get those responses. In a similar fashion, we are waiting final acceptance of the response. Until we get the necessary background on which the judgements were made, that these loads were indeed secondary, I don't believe that we are that

significantly differing from the subcommittee. We have asked the various elements of the industry to respond to those comments. The magnitude of effort that that industry responds to would be indicative of the magnitude of safety concerns that they feel those concerns justify. We are perfectly -- in fact we have indicated rather strongly that generic efforts be established wherever possible, so I don't believe that we are inconsistent in that manner.

DR. PLESSET: Thank you, Jack. I don't want to appear to abrasive in discussing the staff's work but evidently we do have a fair amount of agreement which is unusual between us and you. Dr. Butler?

DR. BUTLER: Let me just add a little bit more. I agree with Jack. We have pretty strong consensus with the views expressed by the subcommittee.

On the matter of margins that Dr. Schrock hit on, I agree with that, that many of the margins depicted during the presentation were relying on what I'll call margins generally looked at for degraded core considerations. When we're dealing with design basis accidents, these different margins have a specific function and we don't want to lean too heavily on them for these new areas.

The other point that I wanted to make is that many

of these new areas are really design questions rather than safety questions. And if you delegate the responsibility to do a good engineering job, you would expect that these issues would be suitably dealt with. There are no real technological questions at hand associated with these issues.

To reinforce Jack's earlier statement, we intend to moderate the amount of resources obligated to resolving these issues. To the extent practical, we will push for generic treatment of them so as to minimize the utilization of resources. Thank you very much.

DR. PLESSET: Thank you, Dr. Butler. I appreciate that and I might say that all the members of the subcommittee received a lot of literature, reports, from meetings of the NRC and maybe we got a little bit of an exaggerated idea of what effort went into this. Jack nods his head indicating concurrence.

MR. KUDRICK: No, I believe that I will be supported by MP&L by saying that there has been significant effort to date on these particular issues.

DR. PLESSET: Yes, and they seem to be getting a little out of hand if I may say so in the amount of effort and report writing and communications and so on and I know you've got a lot of other things you have to

work on, some of which you know, the ACRS thinks are very important that the Staff isn't pushing very hard. I don't need to mention them. You can think of them yourself.

Well, anyway, are there any other comments? Jesse, do you want to comment?

DR. EBERSOLE: No, I rest.

DR. PLESSET: Ivan, Virgil? Well, there's no use keeping you here any longer. We've found it very interesting. I was going to say profitable. I wouldn't go that far. And I presume that you will be meeting with the Grand Gulf subcommittee and with the full committee week after next, is that correct? Well, until then, let's let the subject go. We are adjourned.

(Whereupon, at 1:40 p.m., the meeting was adjourned.)

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NUCLEAR PRODUCTION DEPARTMENT

June 8, 1982

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket Nos. 50-416 and 50-417
File: 0260/L860.0
Justification for Fuel Load and Low Power Testing Pending Resolution of John Humphrey's Concerns
Ref: AECM-82/237
AECM-82/250

As a result of concerns raised by Mr. John Humphrey regarding the design of the Grand Gulf Nuclear Station (GGNS) Mark III containment, representatives of Mississippi Power & Light Company (MP&L), General Electric Company (GE) and Bechtel Power Corporation met with members of your staff and Mr. Humphrey on May 27, 1982 to discuss the concerns. On May 28, 1982, MP&L submitted letter AECM-82/237 which provided a substantial discussion of those concerns and their applicability to the Grand Gulf design. As a result of the May 27, 1982 meeting and the information submitted via the above referenced letter, the concerns have been categorized as "resolved" or "open" to the satisfaction of your staff and Mr. Humphrey. The status of each item is listed in Attachment One. A full list of the concerns is provided in Attachment Two to this letter.

Of the remaining concerns, MP&L has determined that none have the potential to impact the safe operation of the plant at power levels up to 5% of full power. Concern numbers 3.1 through 3.5 relate to use of the Residual Heat Removal System (RHR) for the Steam Condensing Mode. This mode of RHR will not be used until after receipt of a Full Power Operating License. The justification for all other open items is presented below.

On February 12, 1982, MP&L submitted letter AECM-82/55 which described the GGNS phased startup program. Phase I of the program, which would be conducted following fuel load and prior to exceeding 5% of full power, includes low power testing (conducted without the reactor vessel head in place) and non-nuclear heatup which includes tests done at rated temperature and pressure using pump heat as an energy source. During non-nuclear heatup, the reactor will be maintained subcritical at all times with control rod motion allowed for single rod stroke and scram tests only.

Under these circumstances, there is no possibility of a design basis LOCA which would require the GGNS containment to fulfill its safety function in protecting public health and safety. Any events which could be postulated under these conditions are substantially less severe than DBA events. Under these circumstances, there would be minimal risk to the health and safety of the general public and plant personnel for the following reasons:

1) Low power physics testing is conducted at low power levels with the vessel head removed and at temperatures less than 212°F.

2) The radiological source term which could contribute to a radiological health hazard is extremely small since the fuel will have been irradiated to only the very limited degree required for physics testing.

3) Due to the low degree of fuel irradiation, there is an insignificant amount of decay heat which could lead to fuel damage or require containment features to mitigate the effects of a large energy release in the containment.

Although it is our conclusion that all of the technical questions raised have been adequately addressed, MP&L will submit a description of a program to demonstrate the margins and conservatism in the GGNS containment design.

The proposed program will consist of a combination of additional analyses, technical specification and procedure reviews, in-plant testing, and design modifications, as appropriate, to completely resolve each open item. The program description will be submitted by July 16, 1982, and will identify each outstanding issue, the proposed method to resolve the issue, and the schedule for completion. Prior to receipt of the Full Power Operating License, a report will be submitted which includes the results of all analyses completed, test descriptions for any testing which may be required after 5% of full power, and a status of any ongoing analyses and evaluations. The report will be submitted by August 19, 1982 and will justify power operation above 5% of full power for any items which are expected to be open upon the receipt of the Full Power Operating License.

MP&L is confident that the Grand Gulf Nuclear Station is designed and constructed in full compliance with all applicable regulatory requirements and criteria and that the above described program, in combination with the ongoing Independent Design Review being conducted by CYGNA Energy Services, will provide additional proof to support this confidence. We also conclude that the remaining open issues discussed in Attachment Two may be resolved after issuance of a Low Power Operating License for GGNS Unit 1 authorizing fuel loading, testing and operation of the GGNS Unit 1 up to 5% of full power. During this phase of plant operation, none of these concerns has the potential to create an adverse impact to the safe operation of the plant or to pose a hazard to the health and safety of the general public or plant personnel.

Therefore, we request that you proceed with the issuance of the Low Power Operating License for the Grand Gulf Nuclear Station Unit 1.

Yours truly,

J.P. McGaughy

J.P. McGaughy
Assistant Vice President

JPM:rg

cc: Mr. N. L. Stampley
Mr. R. B. McGehee
Mr. T. B. Conner
Mr. G. B. Taylor

Mr. Richard C. DeYoung, Director
Office of Inspection & Enforcement
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Mr. J. P. O'Reilly, Regional Administrator
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta St., N.W. Suite 3100
Atlanta, Georgia 30303

ATTACHMENT ONE
AECM-82/250

STATUS OF HUMPHREY CONTAINMENT CONCERNS
June 7,1982

NOTE: CONCERN IS OPEN UNLESS STATUS SHOWS 'RESOLVED'

CONCERN NO.	STATUS	CONCERN NO.	STATUS
1.1		5.7	RESOLVED
1.2		5.8	
1.3		6.1	RESOLVED
1.4		6.2	RESOLVED
1.5		6.3	
1.6		6.4	RESOLVED
1.7	RESOLVED	6.5	
2.1		7.1	
2.2		7.2	
2.3		7.3	RESOLVED
3.1		8.1	
3.2		8.2	
3.3		8.3	
3.4		8.4	
3.5		9.1	
3.6		9.2	
3.7		9.3	
4.1		10.1	
4.2		10.2	
4.3		11	
4.4		12	RESOLVED
4.5		13	RESOLVED
4.6		14	
4.7		15	RESOLVED
4.8		16	
4.9		17	RESOLVED
4.10		18.1	RESOLVED
5.1		18.2	RESOLVED
5.2	RESOLVED	19.1	
5.3		19.2	
5.4		20	
5.5		21	RESOLVED
5.6		22	

E74sh1

ATTACHMENT TWO
AECM-82/250

HUMPHREY CONTAINMENT CONCERNS

1. <u>Effects of Local Encroachments on Pool Swell Loads</u>

1.1 Presence of local encroachments such as the TIP platform, the drywell personnel airlock and the equipment and floor drain sumps may increase the pool swell velocity by as much as 20 per cent.

1.2 Local encroachments in the pool may cause the bubble breakthrough height to be higher than expected.

1.3 Additional submerged structure loads may be applied to submerged structures near local encroachments.

1.4 Piping impact loads may be revised as a result of the higher pool swell velocity.

1.5 Impact loads on the HCU floor may be imparted and the HCU modules may fail which could prevent successful scram if the bubble breakthrough height is raised appreciably by local encroachments.

1.6 Local encroachments or the steam tunnel may cause the pool swell and froth to move horizontally and apply lateral loads to the gratings around the HCU floor.

1.7 GE suggests that at least 1500 square feet of open area should be maintained in the HCU floor. In order to avoid excessive pressure differentials, at least 1500 $ft.^2$ of opening should be maintained at each containment elevation.

2. <u>Safety Relief Valve Discharge Line Sleeves</u>

2.1 The annular regions between the safety relief valve lines and the drywell wall penetration sleeves may produce condensation oscillation (c.o.) frequencies near the drywell and containment wall structural resonance frequencies.

2.2 The potential condensation oscillation and chugging loads produced through the annular area between the SRVDL and sleeve may apply unaccounted for loads to the SRVDL. Since the SRVDL is unsupported from the quencher to the inside of the drywell wall, this may result in failure of the line.

2.3 The potential condensation oscillation and chugging loads produced through the annular area between the SRVDL and sleeve may apply unaccounted for loads to the penetration sleeve. The loads may also be at or near the natural frequency of the sleeve.

3. ECCS Relief Valve Discharge Lines Below the Suppression Pool Level

3.1 The design of the STRIDE plant did not consider vent clearing, condensation oscillation and chugging loads which might be produced by the actuation of these relief valves.

3.2 The STRIDE design provided only nine inches of submergence above the RHR relief valve discharge lines at low suppression pool levels.

3.3. Discharge from the RHR relief valves may produce bubble discharge or other submerged structure loads on equipment in the suppression pool.

3.4 The RHR heat exchanger relief valve discharge lines are provided with vacuum breakers to prevent negative pressure in the lines when discharging steam is condensed in the pool. If the valves experience repeated actuation, the vacuum breaker sizing may not be adequate to prevent drawing slugs of water back through the discharge piping. These slugs of water may apply impact loads to the relief valve or be discharged back into the pool at the next relief valve actuation and apply impact loads to submerged structures.

3.5 The RHR relief valves must be capable of correctly functioning following an upper pool dump which may increase the suppression pool level as much as five feet creating higher back pressures on the relief valves.

3.6 If the RHR heat exchanger relief valves discharge steam to the upper levels of the suppression pool following a design basis accident, they will significantly aggravate suppression pool temperature stratification.

3.7 The concerns related to the RHR heat exchanger relief valve discharge lines should also be addressed for all other relief lines that exhaust into pool. (p. 132 of 5/27/82 transcript)

4. Suppression Pool Temperature Stratification

4.1 The present containment response analyses for drywell break accidents assume that the ECCS systems transfer a significant quantity of water from the suppression pool to the lower regions of the drywell through the break. This results in a pool in the drywell which is essentially isolated from the suppression pool at a temperature of approximately 135°F. The containment response analysis assumes that the drywell pool is thoroughly mixed with the suppression pool. If the inventory in the drywell is assumed to be isolated and the remainder of the heat is discharged to the suppression pool, an increase in bulk pool temperature of 10°F may occur.

4.2 The existence of the drywell pool is predicated upon continuous operation of the ECCS. The current emergency procedure guidelines require the operators to throttle ECCS operation to maintain vessel level below level 8. Consequently, the drywell pool may never be formed.

4.3 All Mark III analyses presently assume a perfectly mixed uniform suppression pool. These analyses assume that the temperature of the suction to the RHR heat exchangers is the same as the bulk pool temperature. In actuality, the temperature in the lower part of the pool

where the suction is located will be as much as 7½°F cooler than the bulk pool temperature. Thus, the heat transfer through the RHR heat exchanger will be less than expected.

4.4 The long term analysis of containment pressure/temperature response assumes that the wetwell airspace is in thermal equilibrium with the suppression pool water at all times. The calculated bulk pool temperature is used to determine the airspace temperature. If pool thermal stratification were considered, the surface temperature, which is in direct contact with the airspace, would be higher. Therefore the airspace temperature (and pressure) would be higher.

4.5 A number of factors may aggravate suppression pool thermal stratification. The chugging produced through the first row of horizontal vents will not produce any mixing from the suppression pool layers below the vent row. An upper pool dump may contribute to additional suppression pool temperature stratification. The large volume of water from the upper pool further submerges RHR heat exchanger effluent discharge which will decrease mixing of the hotter, upper regions of the pool. Finally, operation of the containment spray eliminates the heat exchanger effluent discharge jet which contributes to mixing.

4.6 The initial suppression pool temperature is assumed to be 95°F while the maximum expected service water temperature is 90°F for all GGNS accident analyses as noted in FSAR table 6.2-50. If the service water temperature is consistently higher than expected, as occurred at Kuosheng, the RHR system may be required to operate nearly continuously in order to maintain suppression pool temperature at or below the maximum permissible value.

4.7 All analyses completed for the Mark III are generic in nature and do not consider plant specific interactions of the RHR suppression pool suction and discharge.

4.8 Operation of the RHR system in the containment spray mode will decrease the heat transfer coefficient through the RHR heat exchangers due to decreased system flow. The FSAR analysis assumes a constant heat transfer rate from the suppression pool even with operation of the containment spray.

4.9 The effect on the long term containment response and the operability of the spray system due to cycling the containment sprays on and off to maximize pool cooling needs to be addressed. Also provide and justify the criteria used by the operator for switching from the containment spray mode to pool cooling mode, and back again. (pp. 147-148 of 5/27/82 transcript)

4.10 Justify that the current arrangement of the discharge and suction points of the pool cooling system maximizes pool mixing. (pp. 150-155 of 5/27/82 transcript)

5. Drywell to Containment Bypass Leakage

5.1 The worst case of drywell to containment bypass leakage has been established as a small break accident. An intermediate break accident will actually produce the most significant drywell to containment leakage prior to initiation of containment sprays.

5.2 Under Technical Specification limits, bypass leakage corresponding to $A/\sqrt{K} = 0.1$ ft.^{2} constitute acceptable operating conditions. Smaller-than-IBA-sized breaks can maintain break flow into the drywell for long time periods, however, because the RPV would be depressurized over a 6 hour period. Given, for example, an SBA with $A/\sqrt{K} = 0.1$, projected time period for containment pressure to reach 15 psig is 2 hours. In the latter 4 hours of the depressurization the containment would presumably experience ever-increasing overpressurization.

5.3 Leakage from the drywell to containment will increase the temperature and pressure in the containment. The operators will have to use the containments spray in order to maintain containment temperature and pressure control. Given the decreased effectiveness of the RHR system in accomplishing this objective in the containment spray mode, the bypass leakage may increase the cyclical duty of the containment sprays.

5.4 Direct leakage from the drywell to the containment may dissipate hydrogen outside the region where the hydrogen recombiners take suction. The anticipated leakage exceeds the capacity of the drywell purge compressors. This could lead to pocketing of hydrogen which exceeds the concentration limit of 4% by volume.

5.5 Equipment may be exposed to local conditions which exceed the environmental qualification envelope as a result of direct drywell to containment bypass leakage.

5.6 The test pressure of 3 psig specified for the periodic operational drywell leakage rate tests does not reflect additional pressurization in the drywell which will result from upper pool dump. This pressure also does not reflect additional drywell pressurization resulting from throttling of the ECCS to maintain vessel level which is required by the current EPGs.

5.7 After upper pool dump, the level of the pool will be 6 feet higher, and drywell-to-containment differential pressure will be greater than 3 psi. The drywell H_2 purge compressor head is nominally 6 psid. The concern is that after an upper pool dump, the purge compressor head may not be sufficient to depress the weir annulus enough to clear the upper vents. In such a case, H_2 mixing would not be achieved.

5.8 The possibility of high temperatures in the drywell without reaching the 2 psig high pressure scram level because of bypass leakage through the drywell wall should be addressed. (pp. 168-174 of 5/27/82 transcript)

6. RHR Permissive on Containment Spray -

6.1 General Electric had recommended that the drywell purge compressors and the hydrogen recombiners be activated if the reactor vessel water level drops to within one foot of the top of active fuel. This requirement was not incorporated in the emergency procedure guidelines.

6.2 General Electric has recommended that an interlock be provided to require containment spray prior to starting the recombiners because of the large quantities of heat input to the containment. Incorrect implementation of this interlock could result in inability to operate the recombiners without containment spray.

6.3 The recombiners may produce "hot spots" near the recombiner exhausts which might exceed the environmental qualification envelope or the containment design temperature.

6.4 For the containment air monitoring system furnished by General Electric, the analyzers are not capable of measuring hydrogen concentration at volumetric steam concentrations above 60%. Effective measurement is precluded by condensation of steam in the equipment.

6.5 Discuss the possibility of local temperatures due to recombiner operation being higher than the temperature qualification profiles for equipment in the region around and above the recombiners. State what instructions, if any, are available to the operator to actuate containment sprays to keep this temperature below design values. (pp. 183-185 of 5/27/82 transcript)

7. Containment Pressure Response

7.1 The containment is assumed to be in thermal equilibrium with a perfectly mixed, uniform temperature suppression pool. As noted under topic 4, the surface temperature of the pool will be higher than the bulk pool temperature. This may produce higher than expected containment temperatures and pressures.

7.2 The computer code used by General Electric to calculate environmental qualification parameters considers heat transfer from the suppression pool surface to the containment atmosphere. This is not in accordance with the existing licensing basis for Mark III environmental qualification. Additionally, the bulk suppression pool temperature was used in the analysis instead of the suppression pool surface temperature.

7.3 The analysis assumes that the containment airspace is in thermal equilibrium with the suppression pool. In the short term this is non-conservative for Mark III due to adiabatic compression effects and finite time required for heat and mass to be transferred between the pool and containment volumes.

8. Containment Air Mass Effects

8.1 This issue is based on consideration that some Tech Specs allow operation at parameter values that differ from the values used in assumptions for FSAR transient analyses. Normally analyses are done assuming a nominal

containment pressure equal to ambient (0 psig) a temperature near maximum operating (90°F) and do not limit the drywell pressure equal to the containment pressure. The Tech Specs operation under conditions such as a positive containment pressure (1.5 psig), temperatures less than maximum (60 or 70°F) and drywell pressure can be negative with respect to the containment (-0.5 psid). All of these differences would result in transient response different than the FSAR descriptions.

8.2 The draft GGNS technical specifications permit operation of the plant with containment pressure ranging between 0 and -2 psig. Initiation of containment spray at a pressure of -2 psig may reduce the containment pressure by an additional 2 psig which could lead to buckling and failures in the containment liner plate.

8.3 If the containment is maintained at -2 psig, the top row of vents could admit blowdown to the suppression pool during an SBA without a LOCA signal being developed.

8.4 Describe all of the possible methods both before and after an accident of creating a condition of low air mass inside the containment. Discuss the effects on the containment design external pressure of actuating the containment sprays. (pp. 190-195 of 5/27/82 transcript)

9. <u>Final Drywell Air Mass</u>

9.1 The current FSAR analysis is based upon continuous injection of relatively cool ECCS water into the drywell through a broken pipe following a design basis accident. The EPG's direct the operator to throttle ECCS operation to maintain reactor vessel level at about level 8. Thus, instead of releasing relatively cool ECCS water, the break will be releasing saturated steam which might produce higher containment pressurizations than currently anticipated. Therefore, the drywell air which would have been drawn back into the drywell will remain in the containment and higher pressures will result in both the containment and the drywell.

9.2 The continuous steaming produced by throttling the ECCS flow will cause increased direct leakage from the drywell to the containment. This could result in increased containment pressures.

9.3 It appears that some confusion exists as to whether SBA's and stuck open SRV accidents are treated as transients or design basis accidents. Clarify how they are treated and indicate whether the initial conditions were set at nominal or licensing values. (pp. 202-205 of 5/27/82 transcript)

10. <u>Drywell Flooding Caused by Upper Pool Dump</u>

10.1 The suppression pool may overflow from the weir wall when the upper pool is dumped into the suppression pool. Alternately, negative pressure between the drywell and the containment which occurs as a result of normal operation or sudden containment pressurization could produce similar overflow. Any cold water spilling into the drywell and striking hot equipment may produce thermal failures.

10.2 Describe the interface requirement (A-42) that specifies that no flooding of the drywell shall occur. Describe your intended methods to follow this interface or justify ignoring this requirement. (pp. 209-226 of 5/27/82 transcript)

11. Operational Control of Drywell to Containment Differential Pressures

Mark III load definitions are based upon the levels in the suppression pool and the drywell weir annulus being the same. The GGNS technical specifications permit elevation differences between these pools. This may effect load definition for vent clearing.

12. Suppression Pool Makeup LOCA Seal In

The upper pool dumps into the suppression pool automatically following a LOCA signal with a thirty minute delay timer. If the signal which starts the timer disappears on the solid state logic plants, the timer resets to zero preventing upper pool dump.

13. Ninety Second Spray Delay

The "B" loop of the containment sprays includes a 90 second timer to prevent simultaneous initiation of the redundant containment sprays. Because of instrument drift in the sensing instrumentation and the timers, GE estimates that there is a 1 in 8 chance that the sprays will actuate simultaneously. Simultaneous actuation could produce negative pressure transients in the containment and aggravate temperature stratification in the suppression pool.

14. RHR Backflow Through Containment Spray

A failure in the check valve in the LPCI line to the reactor vessel could result in direct leakage from the pressure vessel to the containment atmosphere. This leakage might occur as the LPCI motor operated isolation valve is closing and the motor operated isolation valve in the containment spray line is opening. This could produce unanticipated increases in the containment spray.

15. Secondary Containment Vacuum Breaker Plenum Response

The STRIDE plants had vacuum breakers between the containment and the secondary containment. With sufficiently high flows through the vacuum breakers to containment, vacuum could be created in the secondary containment.

16. Effect of Suppression Pool Level on Temperature Measurement

Some of the suppression pool temperature sensors are located (by GE recommendation) 3" to 12" below the pool surface to provide early warning of high pool temperature. However, if the suppression pool is drawn down below the level of the temperature sensors, the operator could be misled by erroneous readings and required safety action could be delayed.

17. Emergency Procedure Guidelines

The EPGs contain a curve which specifies limitations on suppression pool level and reactor pressure vessel pressure. The curve presently does not adequately account for upper pool dump. At present, the operator would be required to initiate automatic depressurization when the only action required is the opening of one additional SRV.

18. Effects of Insulation Debris

18.1 Failures of reflective insulation in the drywell may lead to blockage of the gratings above the weir annulus. This may increase the pressure required in the drywell to clear the first row of drywell vents and perturb the existing load definitions.

18.2 Insulation debris may be transported through the vents in the drywell wall into the suppression pool. This debris could then cause blockage of the suction strainers.

19. Submergence Effects on Chugging Loads

19.1 The chugging loads were originally defined on the basis of 7.5 feet of submergence over the drywell to suppression pool vents. Following an upper pool dump, the submergence will actually be 12 feet which may effect chugging loads.

19.2 The effect of local encroachments on chugging loads needs to be addressed. (pp. 251-252 of 5/27/82 transcript)

20. Loads on Structures Piping and Equipment in the Drywell During Reflood

During the latter stages of a LOCA, ECCS overflow from the primary system, can cause drywell depressurization and vent backflow. The GESSAR defines vent backflow vertical impingement and drag loads, to be applied to drywell structures, piping, and equipment, but no horizontal loading is specified.

21. Containment Makeup Air For Backup Purge

Regulatory Guide 1.7 requires a backup purge H_2 removal capability. This backup purge for Mark III is via the drywell purge line which discharges to the shield annulus which in turn is exhausted through the standby gas treatment system (SGTS). The containment air is blown into the drywell via the drywell purge compressor to provide a positive purge. The compressors draw from the containment, however, without hydrogen lean air makeup to the containment, no reduction in containment hydrogen concentration occurs. It is necessary to assure that the shield annulus volume contains a hydrogen lean mixture of air to be admitted to the containment via containment vacuum breakers.

22. Miscellaneous Emergency Procedure Guideline Concerns

The EPGs currently in existence have been prepared with the intent of coping with degraded core accidents. They may contain requirements conflicting with design basis accident conditions. Someone needs to carefully review the EPG's to assure that they do not conflict with the expected course of the design basis accident.

September 16, 1982

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing "Applicants' Answer To Ohio Citizens For Responsible Energy Motion For Leave To File Its Contentions 21-26" were served by deposit in the U.S. Mail, First Class, postage prepaid, this 16th day of September, 1982, to all those on the attached Service List.

Jay E. Silberg

JAY E. SILBERG

September 16, 1982

NUCLEAR REGULATORY COMMISSION

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