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October 3, 1984

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED
USNRC

Before the Atomic Safety and Licensing Board

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

OCRE RESPONSE TO APPLICANTS' MOTION FOR SPECIFICATION OF A CREDIBLE ACCIDENT SCENARIO UNDER ISSUE #8

On September 18, 1984 Applicants moved that the Licensing Board require Intervenor Ohio Citizens for Responsible Energy ("OCRE") to specify a credible accident scenario "so that Issue #8 can be fairly resolved on a reasonable schedule." (Motion at 2) Applicants claim that this measure (rejected by the Board in March 1983)[1/] is now necessary to avoid delay of the proceeding[2/] and that it is no longer reasonable to assume that the hydrogen control rule is imminent. OCRE agrees with Applicants that consideration of Issue #8 should not be suspended any longer. Indeed, it was in this spirit that OCRE filed its 13th Set of Interrogatories to Applicants and motion

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1/ This rejection is based on the fact that when the rule issues, CLI-80-16 will in essence overturned, as the premise of that decision, that postulating hydrogen generation in excess of that in the present 10 CFR 50.44 guidelines challenges that regulation, will become non-existent. What will be at issue is whether Applicants will comply with the rule, which is always litigable.

2/ Applicants apparently believe that this proceeding must be expedited at all costs because they "currently plan to be ready to load fuel in Unit 1 by mid-1985." Motion at 4. First of all, this attitude is contrary to the clear statements of the Appeal Board: when important safety issues are yet to be considered,

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to reopen discovery on July 30, 1984. However, the remedy proposed by Applicants, the specification (and presumably litigation) of a "credible accident scenario" would amount to a colossal waste of time for the Board and parties. Far from reducing delay, this step would only compound it; e.g., one can only imagine the delay engendered by the mere definition of the highly subjective term "credible". OCRE believes this delay is totally unnecessary; by agreeing to the standards and criteria for the litigation of Issue #8, the Board and parties can avoid unnecessary delay of the proceeding, focus on the real safety issues involved, and satisfy the requirements of all the applicable law.

I. Status of the Hydrogen Control Rule

Since Applicants claim that the issuance of the hydrogen control rule for BWR Mark III containment does not appear to be imminent, an examination of the status of that rule is necessary.

The proposed rule on hydrogen control for BWRs with Mark III containments and PWRs with ice condenser containments was published in the Federal Register for public comment on December 23, 1981. The public comment period closed on April 8, 1982. In June 1983 the proposed rule was reviewed by the Committee to Review Generic Requirements. The current thinking on the hydrogen control rule appears to be embodied in SECY-83-357, dated August 26, 1983. This document contains a summary of the public comments, Staff

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2/ continued. delay resulting therefrom is proper because the facility is not ready to operate. Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-124, 6 AEC 358, 365. Secondly, it should be noted that the NRC's Caseload Forecast Panel has concluded that a fuel load of late 1985 is attainable

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responses thereto, the draft final rule, a regulatory analysis, ACRS correspondence, and an analysis of manual vs. automatic actuation of hydrogen igniter systems.

OCRE has attached as Exhibit 1 Enclosure F to SECY-83-357, the draft Notice of Final Rulemaking. This rule would require for BWRs with Mark III containments:

> that a hydrogen control system, justified by a suitable program of experiment and analysis, be provided that can handle, without loss of containment structural integrity, the hydrogen resulting from a 75% metal-water reaction;
>
> that containment structural integrity be demonstrated by proper analytical techniques;
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> that systems and components required for safe shutdown and the maintenance of containment integrity are operable after exposure to hydrogen burn environments, including local detonations (unless local detonations can be shown to be unlikely).

This proposal came before the Commission on November 9, 1983; the Commission had questions for the Staff and thus deferred action on the rule. The rule was again scheduled for Commission action in March of this year, but was again deferred, due to the need for evaluation of recent data from hydrogen combustion tests conducted at the Nevada Test Site. A recent conversation between OCRE and Morton Fleishman (contact person for the hydrogen rule and source of the information in this paragraph) revealed that the rule may come before the Commission within the next few weeks.

It is important to realize that, even though the rule is not finalized, the NRC Staff is in fact applying it in its licensing

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2/ continued. for Unit 1; however, not much contingency is built into Applicants' schedule, and preoperational testing will have to be problem-free to meet this goal. See the August 28, 1984 memorandum for B J. Youngblood from J.J. Stefano.

review of PNPP and other Mark III plants. Hydrogen control for degraded core accidents is identified as a license condition in the Perry SER, NUREG-0887. Exhibits 2 through 6, attached hereto, reveal the similarity between the draft final rule and the Staff's evaluation criteria.

Because of the inevitability of the issuance of the hydrogen control rule,[3/] and since the Staff is in essence requiring Mark III owners to meet the rule's requirements before finalization, OCRE proposes, to avoid delay and to focus on the real issues, that the litigation of Issue #8 be according to the draft final rule as given in Enclosure F to SECY-83-357. I.e., Applicants must have a hydrogen control system, justified by proper experiment and analysis, capable of handling a 75% metal-water reaction without loss of containment structural integrity or damage to equipment needed for safe shutdown or maintaining containment integrity. It is OCRE's contention that Applicants cannot meet these requirements. Under this standard, specification of a "credible accident scenario" is unnecessary as a degraded core accident with 75% metal-water reaction is postulated.

II. Legal Standards

A. CLI-80-16

Applicants base their request that OCRE specify a scenario largely on Metropolitan Edison (TMI-1 Restart), CLI-80-16, 11 NRC 674 (1980). This decision, rendered in a special proceeding,

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3/ The evaluation of NTS data mentioned above will not impact on the rule for Mark IIIs; rather the "hold-up" appears to be whether additional hydrogen control measures are necessary for PWRs with large dry containments.

not an operating license case, reached three major points:

- that no "special circumstances existed that warranted waiving 10 CFR 50.44 for the TMI facility;

- that, apart from 10 CFR 50, hydrogen control could be litigated under Part 100 if it were shown that there is a credible LOCA scenario entailing hydrogen generation, hydrogen combustion, containment breach or leakage, and offsite radiation doses in excess of Part 100 guidelines;

- that the Commission planned to address the hydrogen issue in a generic rulemaking.

Two Commissioners dissented from CLI-80-16. A motion for reconsideration of CLI-80-16 resulted in a 2-2 vote (see unpublished Commission Order of September 26, 1980). In their strong dissent in Duke Power Co. (Wm. B. McGuire Nuclear Station, Units 1 and 2), CLI-81-15, 14 NRC 1 (1981) former Commissioners Gilinsky and Bradford demonstrate that 10 CFR 100 (which actually uses the word "credible" to describe accidents involving substantial core meltdown), had it been written post-TMI, would have assumed the presence of large quantities of hydrogen gas along with the fission product inventory. 14 NRC at 10. The dissenting Commissioners said that "(t)here is no need, in applying the test of Part 100, to require a detailed accident sequence." Id., emphasis in original. The Commissioners concluded that "(t)o continue to require the parties, including the staff if a licensee should choose to contest the point, to prove the "credibility" of given accident sequences, when the Commission itcslf requires the installation of hydrogen control systems without such proof, is an exercise in futility." 14 NRC at 11.

The above indicates that CLI-80-16 has been on shaky legal ground from its issuance. That decision, reached on May 16, 1980, also needs to be examined in light of its historical context. The Commission's Order was issued little more than a year after the TMI-2 accident which catapulted the hydrogen issue into public and scientific controversy. Considering the multitude of TMI issues, of which hydrogen control is just one, and the considerable uncertainty surrounding hydrogen behavior in reactor accidents, the Commission's decision to address the hydrogen issue generically had some merit.

Now it over 4 years after the issuance of CLI-80-16. Much research has been conducted. Regulatory decisions have been reached. It has been determined that small containments (BWR Mark I and II) should be continuously inerted. PWRs with large dry containments apparently need no modifications for hydrogen control, other than recombiners and high-point vents. The intermediate size, low design pressure, pressure suppression containments (BWR Mark III and PWR ice condensers), being the most vulnerable to the effects of hydrogen combustion, will have to meet more stringent requirements. What was in 1980 considered a generic issue has since been broken up into a number of design-specific remedies.

Even these cannot be considered truly generic solutions. Unlike an "ideal" generic issue which affects every plant in exactly the same way (e.g., environmental effects of the nuclear fuel cycle), hydrogen control in the remaining, viable Mark III plants (Perry, Grand Gulf, Clinton, and River Bend) can by no means be treated generically. These four plants are quite diverse.

They differ in basic containment design and materials of construction (e.g., Perry, using the standard Mark III design, has a free-standing steel vessel surrounded by a concrete shield building, while Grand Gulf, using the alternate Mark III design, has a reinforced concrete vessel) and internal features and configurations (e.g., ratio of containment volume to core power, containment spray flow rate, and the River Bend fan coolers, which the other 3 plants do not have). These differences demand a plant-specific evaluation of the adequacy of hydrogen control measures.

Clearly, time and events have overtaken CLI-80-16. The proper way to treat hydrogen control in Mark III containments in 1984 is neither to consider it a generic issue nor to require a party (including the NRC Staff, if Applicants choose to contest the matter) to prove that accidents can happen. Rather, plant-specific evaluation and litigation, according to the reasonable standards of the draft final rule in SECY-83-357, are mandated.

B. Court Cases

The U.S Court of Appeals has made it abundantly clear that the Commission cannot deny affected members of the public a hearing on an issue of material fact to the NRC's licensing decisions, as such action contravenes §189 of the Atomic Energy Act. Union of Concerned Scientists v. NRC, Case No. 82-2053, May 25, 1984. Although the mandate of this case will not issue until the expiration of the time period (November 1) in which the Commission may petition for certiorari to the Supreme Court, there is sound precedent for this principle.

In Independent Bankers Association of Georgia v. Board of Governors of the Federal Reserve System, 516 F.2d 1206 (D.C. Cir. 1975), it was held that where Congress has plainly given interested parties the right to a full hearing, an agency which claims that an evidentiary hearing would serve absolutely no purpose must show that the parties could gain nothing thereby because they are disputing none of the material facts on which the agency's decision could rest.

Closer to home, the Court in Siegel v. AEC, 400 F.2d 778 (D.C. Cir. 1968) stated that the hearing granted by the Atomic Energy Act must embrace all relevant matters; however, should the Commission decide that an issue is not relevant, the hearing need not consider the matter. In Siegel, the issue at hand was whether nuclear power facilities should be required to withstand enemy attack. The Court upheld the Commission, as it did not require and never had required licensees to meet such a standard, and had no intentions of doing so in the future. The matter was thus not an issue of material fact suitable for hearing.

However, hydrogen control at a Mark III plant is certainly an issue material to the issuance of an operating license, as demonstrated by Exhibits 2 through 6 and the fact that the Staff has made it a licensing condition for Perry. Using the reasoning of UCS and its predicessors, the Commission may not deny a party a hearing on the issue. Although the Commission in CLI-80-16 did not bar outright the litigation of hydrogen control, it did create a rather severe threshold for its consideration:

> specification of a credible LOCA scenario entailing hydrogen generation, combustion, containment breach or leakage, and offsite radiation doses exceeding Part 100 values.

Depending on the interpretation of "credible" and the standard of proof required, this threshold could easily become a de facto barrier to the litigation of hydrogen control, which is apparently what happened at TMI-1. But such an interpretation of CLI-80-16 here is obviously illegal as it denies a party the right to litigate an issue being pursued by the Staff as a licensing condition.[4/]

It is OCRE's interpretation of the law cited above that to require any standard beyond that for the normal admission of contentions (i.e., basis stated with specificity, 10 CFR 2.714(b)) as a prerequisite to the litigation of hydrogen control at a Mark III plant is plainly illegal.

C. ALAB-675 and Licensing Board Decisions

When admitting Issue #8, the Licensing Board wisely made these comments with respect to CLI-80-16:

> We find these recent Commission utterances [the proposed rule published December 23, 1981], proposed and tentative though they may be, to be inconsistent with the TMI decision on which we relied. The Commission now appears to be of the view that the assumptions of § 50.44 are unrealistic and that some additional steps may need to be taken. While we could adopt a wait-and-see attitude on this important matter, we believe it to be more prudent to proceed on the assumption that by the commencement of operation of Perry, the require-

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4/ Compare the concurring opinion of Chief Judge Bazelon in Citizens for Safe Power v. NRC, 524 F.2d 1291, 1303 (D.C. Cir. 1975): "to require of impecunious associations of private citizens a quantum of evidence beyond their financial means to marshall, as a prerequisite to examining the rule or its controlling effect, is to blunt the tools with which bad or outdated rules are discarded or limited."

> ments of 10 CFR 50.44 will be more stringent. Memorandum and Order (Concerning Late-Filed Contentions), March 3, 1982, slip op. at 8;

and on the litigation of specific accident scenarios:

> It seems to us that little purpose would be served by litigating the likelihood that any one of the suggested scenarios . . . could occur. There is little doubt that any one scenario, except perhaps for the occurrence of human error, would be highly unlikely to occur. However, we could embark on an endless search for multiple, unlikely events unless we assay that tortuous path in advance and refuse to enter. Id. at 11.

Applicants subsequently sought directed certification of this Order. In denying their motion, the Appeal Board in ALAB-675 stated that:

> different types of accidents, however, result in different rates and quantities of hydrogen generation. A given hydrogen-generating mechanism thus has obvious relevance to the efficacy of a hydrogen control system. In order to litigate meaningfully the adequacy of such a system, a particular accident or accidents should be specified. ALAB-675, slip op. at 17. 5/

Note that the Appeal Board's emphasis is not on having intervenors prove that accidents can happen, but rather it is on the effect of various accidents on the efficacy of the Perry hydrogen control system. Compare Exhibit 4, which states the Staff's position that a variety of degraded core accidents, with varying rates of hydrogen production with the total amount equal to a 75% metal-water reaction, be postulated for evaluating the hydrogen control system. See also p. 23 of Exhibit 1, draft final 10 CFR 50.44(c)(3)(vi)(B)(3), which requires the utilities to specify accident scenarios for analyzing the hydrogen control system. Under the hydrogen control rule as now written and

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5/ This view is accepted by the Licensing Board. See the December 23, 1982 Memorandum and Order at 2.

under the Staff's current evaluation standards, it is not OCRE's job to specify accident scenarios; rather, it is Applicants'. Of course, OCRE may challenge the completeness and assumptions of the scenarios Applicants choose, but it is clear that neither ALAB-675, the Staff's evaluation standards, nor the draft final rule require OCRE to prove that accidents involving substantial hydrogen generation are "credible" as a condition of examining the efficacy of Applicants' hydrogen control system.

III. Conclusion

Applicants' request that OCRE specify a "credible" accident scenario is essentially an anachronism. Their logic is more appropriate for 1980 than for 1984. They neglect the applicable case law, the clear pronouncement of the Appeal Board, the status of the hydrogen control rule, the vitiated state of CLI-80-16, and the fact that the Staff is already requiring hydrogen control measures for degraded core accidents at Perry.

Applicants should stop hiding behind the tired cliché that "accidents can't happen" and face the facts. The fact is that nuclear accidents can and do happen. The fact is that additional hydrogen control requirements are now being imposed on PNPP. If Applicants have confidence in their hydrogen control system, they should not be afraid to litigate the issue on its merits.

If Applicants were truly concerned with the avoidance of delay, they would agree with OCRE, instead of urging the Board to waste its time on the absurd.

The solution proposed by OCRE for the litigation of Issue #8, that the criteria of Exhibit 1, the draft final rule on hydrogen control for intermediate containments (which appears to be implemented already by the Staff in its evaluation of hydrogen control at Perry), be declared by the Board as controlling, is the only solution to this problem that is rational, reasonable, and consistent with law, fact, justice, fairness, and the Board's independent responsibility to ensure safety. OCRE therefore urges the Board to impose this standard.

Respectfully submitted,

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NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

EXHIBIT 1

Hydrogen Control Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: F nal Rule.

SUMMARY: The Commission is amending its regulations to improve hydrogen control capability for boiling water reactors with MARK III containments and for pressurized water reactors with ice condenser containments. The amendments require improved hydrogen control systems that can handle large amounts of hydrogen during and following an accident. For those of the above reactors not relying upon an inerted atmosphere for hydrogen control, the rule requires that certain systems and components be able to function during and following hydrogen burning. The rule also requires affected licensees to submit analyses to the Commission in support of the previous two requirements. The rule is needed to improve the capability of the indicated types of nuclear power reactors to withstand the effects of a large amount of hydrogen generation and release to containment from an accident, as occurred at Three Mile Island. The new requirements will result in greater assurance that nuclear power reactor containments and safety systems and components will continue to function properly so that the reactors can be safely shut down following a Three Mile Island-type of accident.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone 301-443-7616.

SUPPLEMENTARY INFORMATION:

Background

The Commission has taken numerous actions to correct the design and operational limitations that were revealed by the accident at Three Mile Island, Unit 2 (TMI-2), which resulted in a severely damaged or degraded reactor core, in a concomitant release of radioactive material to the primary coolant system, and in a fuel cladding-water reaction causing the generation of a large amount of hydrogen. Included in these actions are several rulemaking proceedings intended to improve the hydrogen control capability of light-water nuclear power reactors.

On December 23, 1981, the Commission published in the Federal Register (46 FR 62281) a notice of proposed rulemaking on "Interim Requirements Related to Hydrogen Control," inviting written comments or suggestions on the proposed rule by February 22, 1982. A notice extending the comment period for an extra 45 days to April 8, 1982, including editorial corrections, was published in the Federal Register on February 25, 1982 (47 FR 8203). The notice concerned proposed amendments to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which would have required that:

a. Each boiling water reactor (BWR) with a Mark III type containment and each pressurized water reactor (PWR) with an ice condenser type containment be provided with a hydrogen control system capable of handling an amount of hydrogen equivalent to that which would be generated if there were at least a 75 percent fuel cladding-water reaction without loss of containment integrity;

b. Each boiling water reactor and each pressurized water reactor that does not rely on an inerted atmosphere for hydrogen control be provided with safety systems needed to establish and maintain safe cold shutdown and maintain containment integrity that can function after the burning of substantial amounts of hydrogen; and

c. Analyses be performed for the reactor categories mentioned above to justify the hydrogen control systems selected and to assure containment structural integrity and survivability of needed safety systems during a hydrogen burn.

It should be noted that the proposed rule was not part of the separate, long-term rulemaking on degraded or melted cores (the "severe accident rulemaking") for which an advance notice of proposed rulemaking was published on October 2, 1980 (45 FR 65474) and which was the subject of the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," published in the Federal Register on April 23, 1983 (48 FR 16014).

General Comments

Twenty-eight persons submitted comments regarding the proposed amendments. The comments and the SECY paper noted above are part of the public record and may be examined and copied, for a fee, in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. A summary of the comments and a comment analysis are also available for inspection and copying, for a fee, in the Public Document Room.

The comments received have been carefully reviewed and evaluated during preparation of this final rule. The final rule contains revisions to the proposed rule that reflect consideration of these comments. The commenters generally provided many specific comments on all aspects of the proposed amendments. The following discussion represents a distillation of the more significant comments.

Numerous commenters suggested that the implementation of the Hydrogen Control Rule should be deferred until the severe accident rulemaking (see above) when applicable research and probabilistic risk analyses (PRAs) will be completed. The Commission agrees with these comments relative to PWRs with large dry containments. Dry containment designs have a greater inherent capability to accommodate large quantities of hydrogen because of their high design pressure and large volume; therefore, for these designs the Commission believes that rulemaking with regard to hydrogen control can be safely deferred pending completion of NRC- and industry-sponsored research. Furthermore, with regard to systems and components that must be able to function during and following hydrogen burning, the fact that TMI-2 was shut down and maintained in a shutdown condition indicates that such systems and components did generally perform their functions following the burn event. In addition, design improvements that have been implemented as a result of NRC directives have served to reduce the likelihood of a degraded core accident.

With regard to BWRs with Mark III containments and PWRs with ice condenser containments, the Commission believes that these containments can safely accommodate the burning in a single event of the hydrogen from about a 25 percent metal-water reaction.[1] However, since the TMI-2 accident showed that a 45-50 percent metal-water reaction was possible, the Commission believes that it is necessary to enhance the hydrogen control capability for reactors with these types of containments and that new regulations are required to ensure that the proper design features are incorporated. Adoption of the final rule will also formalize Commission regulatory decisions currently being applied on a case-by-case basis in individual licensing proceedings and will provide the needed basis for regulatory actions that cover licensing and continued operation of the affected plants.

Several commenters stated that the 75 percent metal-water reaction required to be assumed for design and analysis is unreasonably high based on evaluation of the TMI-2 accident and analyses of recoverable degraded core accidents.[2] The 75 percent metal-water reaction chosen by the Commission is greater than that which occurred during the TMI-2 accident; however, the primary intent of

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[1]The basis for this belief is contained in SECY 80-107, "Proposed Interim Hydrogen Control Requirements for Small Containments," February 22, 1980, which is available for inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

[2]See the following studies, available for inspection at the Commission's Public Document Room at 1717 H Street, NW, Washington, D. C. Also NUREG and NUREG/CR publications may be purchased from the NRC/GPO Sales Program by calling (301) 492-9530.

NUREG/CR-2540, "A Method for the Analysis of Hydrogen and Steam Releases to Containment During Degraded Core Cooling Accidents," February 1982

NUREG/CR-1219, "Analysis of the Three Mile Island Accident and Alternative Sequences," January 1980

"Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements," Rev. 1, July 1982 - Submitted by the BWR/6 MARK III Hydrogen Control Owners Group.

the rule is to require containment designs that can accommodate accident sequences in which hydrogen combustion poses a significant threat to containment integrity. Consequently, the Commission believes it is prudent to specify a value sufficiently greater than that which was estimated to have occurred at TMI-2 so that there will be an appropriate margin of safety. The Commission feels confident that the 75 percent value is representative of a limiting case degraded core accident (beyond which a core melt is likely to occur). Finally, the Commission sees no significant benefit in reducing the metal-water reaction to a level such as 50 percent for those plants having Mark III and ice condenser containments since the basic design of the heretofore chosen igniter system would not change.

A number of commenters recommended that the requirement for a hydrogen control system be revised to permit licensees the option of demonstrating analytically that additional hydrogen control systems are not necessary because of intrinsic design capabilities that reduce the likelihood of hydrogen generation. While it is true that design features to reduce hydrogen generation are necessary and desirable, the Commission still believes that, in order to cope with unexpected events, there should be a solution to the hydrogen issue that involves design features that ensure containment integrity, even if a large amount of hydrogen is generated. Thus, while measures to prevent the the generation of large amounts of hydrogen are necessary and desirable, the Commission believes that it is also necessary, depending upon containment design, to provide measures to mitigate the effects of large amounts of hydrogen.

Some commenters indicated that, since the primary function of the containment is to prevent excessive radiation dose to the public, the rule should be modified to preclude the loss of containment function rather than to preclude the loss of containment integrity. The Commission appreciates the fact that some nuclear plants are designed with a multi-building, multi-barrier concept that is intended to prevent the leakage of radiation by diverse methods such as filtering and scrubbing mechanisms, plate-out mechanisms, and containment sprays. However, the Commission's safety philosophy remains the same; namely, the containment should be designed to remain intact following a recoverable degraded core accident in order to provide additional assurance that excessive radiation will not be released. In other words, the Commission reaffirms its policy that the prevention of excessive radiation dose to the public can best be assured by maintaining a leak tight containment and that this, in turn, can be provided by assuring that there is structural integrity with margin.

Some commenters stated that the criterion for containment structural integrity is unnecessarily restrictive. They stated that it should not be limited to the provisions of the ASME Boiler and Pressure Vessel Code, but should permit the use of other methods such as realistic analyses using actual material properties. The Commission agrees with this comment and has modified the rule in this regard. Section 50.44(c)(3)(iv) has been changed to indicate that "containment structural integrity must be demonstrated by use of an analytical technique that has been accepted by the NRC staff." The rule includes two alternative methods as examples but does not preclude other methods that may be shown to be acceptable to the Commission. Finite element analysis would be acceptable for use with the methods considered.

It was suggested by some commenters that the rule should address only non-inerted, small-volume, low-pressure containments and should not impose requirements for the remaining containments since, for these containments, it would provide, at best, insignificant improvements in safety. The Commission agrees for the reasons indicated above; therefore, as indicated previously, it has revised the rule to apply only to Mark III BWRs and ice condenser PWRs.

A number of commenters stated that the rule ignores those post-TMI suggested improvements which have been implemented and which reduce the likelihood of a degraded core accident. In the case of PWRs with large dry containments, as discussed above, the Commission believes that the post-TMI improvements, along with the inherent strength of the containments, have indeed provided sufficient safety to permit the delay of any additional rulemaking until completion of ongoing research programs.

It has been recommended that in view of the small probability of occurrence of local detonations as a result of various design features, the rule should permit licensees the option of demonstrating that local detonations cannot occur in lieu of evaluating the effects of local detonations. The Commission agrees and has modified paragraphs 50.44(c)(3)(v) and (vi) of the rule appropriately.

Many commenters indicated that they believe the requirement that systems and components that can function after a hydrogen burn be provided for "safe cold shutdown" is unnecessary and is inconsistent with the licensing basis for most operating plants which requires only "safe shutdown". Those commenters felt that the safe shutdown criterion should not be an issue with regard to hydrogen control, but that it should be considered in another forum. Because of the fact that a degraded core accident is less likely than a design basis accident, the Commission agrees that the requirement for cold shutdown may be overly conservative. The licensing basis for most plants is, in fact, just safe shutdown. The reference to cold shutdown has been deleted from the rule; but the Commission notes that the issue of safe shutdown versus safe cold shutdown has not yet been resolved. The issue is expected to be addressed within the context of the resolution of Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," which is the subject of current NRC staff effort.

Several commenters have suggested that the implementation schedules should be made more realistic so that design changes logically follow after the required analyses are completed. The Commission agrees. The greatest relief, of course, has come by deferring implementation of the rule for PWRs with large dry containments. However, the rule has also been revised to specify that each applicant and licensee subject to the rule shall propose a schedule, to the Commission, for meeting the requirements. A final schedule for implementing the requirements shall be mutually agreed upon by the applicant or licensee and the NRC staff. The Commission anticipates that most applicants and licensees will be able to implement these requirements within two years. (See §50.44(c)(3)(vii).)

Some commenters noted that in the Supplementary Information accompanying the proposed rule it was stated that the selection of the hydrogen control system should be supported by comparative analyses of alternative systems to show their relative advantages and disadvantages. They stated that this guidance is inconsistent with Commission practice and is unnecessary. They felt that the only requirement should be a demonstration that the selected system is suitable for its intended application.

The Commission agrees that this guidance was inconsistent with Commission practice in the case of operating reactors and reactors for which operating licenses are about to be issued in the near-term. In the final rule, § 50.44(c)(3)(vi) has been modified to delete the implication that comparative analyses are required and to indicate that the analysis is intended to support the design of the hydrogen control system that is selected. Comparative analyses of alternative systems are not required.

HYDROGEN CONTROL SYSTEMS [§ 50.44(c)(3)(iv)]

As originally proposed, applicants and licensees with boiling water reactor (BWR) facilities with Mark III type containments and pressurized water reactor (PWR) facilities with ice condenser type containments, for which construction permits were issued prior to March 28, 1979, are required to install hydrogen control systems capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75 percent of the fuel cladding (surrounding the active fuel region) with water, without loss of containment integrity. The particular type of hydrogen control system to be selected is left to the discretion of the applicant or licensee; however, the NRC must find it acceptable based upon suitable programs of experiment and analysis. The design of the selected system must be supported by the analyses which are to be submittted as part of the analyses required under § 50.44(c)(3)(vi). The system that is proposed and approved must safely accommodate large amounts of hydrogen, and operation of the system, either intentionally or inadvertently, must not further aggravate the course of an accident or endanger the plant during normal operations. As discussed previously, the amount of hydrogen to be assumed in the design of the hydrogen control system is that amount generated when 75 percent of the fuel cladding surrounding the active fuel region reacts with water.

As discussed above, the limited method proposed to demonstrate containment structural integrity has been expanded. Containment structural integrity may now be demonstrated by use of an analytical technique that has been accepted by the NRC staff. For example, finite element analysis is one acceptable technique for use with the methods considered. One of the acceptable methods is the use of the applicable ASME Boiler and Pressure Vessel Code. However, the Commission will accept other methods, provided that convincing evidence is presented regarding their suitability.

Other changes from the proposed rule are the relaxation of the implementation schedule to one that has been mutually agreed upon by the licensee and the NRC staff, and the elimination of the word "cold" in the phrase "safe cold shutdown."

SYSTEMS AND COMPONENTS [§ 50.44(c)(3)(v)]

At the time the proposed rule was issued for comment, the Commission indicated that it was considering a two-step approach to address "qualification" (as defined below) of those systems and components that must be able to function during and after a hydrogen burn. For the reasons explained below, the Commission did not choose this two-step approach. As the proposed first step, there would have been a demonstration that these systems and components could "survive" the hydrogen burn and continue to be able to perform their safety function. This step would not have entailed that these systems and components actually be qualified pursuant to NRC's qualification program. The proposed second step would have entailed the actual "qualification" of these systems and components. The conceptual differences between systems and components demonstrated to be "survivable" and systems and components demonstrated to be "qualified" were also described.

The Commission specifically sought comments on the use of the two-step approach for defining standards, on the "survivability" and "qualification" approaches themselves, and on proposals for implementation schedules. There were numerous comments in response to this request. The overwhelming reaction was that the two-step approach to reaching a survivability determination is unwarranted and will unnecessarily escalate the costs to industry. Many commenters felt that a straightforward survivability approach would be appropriate provided reasonable guidelines are specified. In view of the smaller likelihood of a degraded core accident as compared to a design basis accident, which has been reduced further by post-TMI improvements, the Commission has decided to forego the two-step approach previously described. The Commission now believes, in view of the recent issuance of 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety," that there is no significant difference between demonstrating survivability and demonstrating qualification. Paragraph (f) of § 50.49 describes several methods, one of which must be used, for qualifying electrical equipment important to safety. For example, for those licensees which have already demonstrated survivability, as described in the Supplementary Information of the notice of proposed rule-making for this rule on hydrogen control requirements (46 FR 62281, Dec. 23, 1981), the qualification methods described in paragraphs (f)(2) and (f)(4) of § 50.49 could be used to show that the systems and components have been qualified. In this regard, the margins considered adequate for a degraded core accident are less than those considered adequate for a design-basis accident due to the lower probability of occurrence of a degraded core accident.

The Commission now views "qualification" as the generation and maintenance of evidence using tests and analyses to assure that systems and components will operate on demand to meet system performance requirements. In the case of a hydrogen burn environment, this means that there must be adequate evidence that systems and components necessary to establish and maintain safe shutdown and to maintain containment integrity are capable of performing their functions during and after exposure to the environmental conditions created by the postulated accident, including the burning of hydrogen. Qualification may be demonstrated in a manner acceptable to the Commission using a combined approach of analysis and testing. Thus, an acceptable thermal analysis would have to be performed for the containment in order to determine the thermal response of the components during a hydrogen burn. This thermal response should then be compared to the thermal response the components had during their qualification testing. The licensee should then demonstrate that the qualification thermal response envelops the thermal response during a hydrogen burn. Selected tests should also be performed at predicted hydrogen burn conditions (or, other tests previously performed may be referenced if demonstrated to be applicable) to reasonably assure the Commission that the systems and components are qualified to perform their functions during and following a hydrogen burn.

Paragraph 50.44(c)(3)(v) applies to those Mark III BWRs and ice condenser PWRs that do not have an inerted containment atmosphere for hydrogen control. At present, this includes all Mark III BWRs and ice condenser PWRs, since no applicant or licensee has as yet elected to use the inerting option for these plants. The systems and components that must be qualified for a hydrogen burn are those needed (a) to shut down the reactor and bring it to and maintain it in a safe shutdown condition, and (b) to prevent loss of containment integrity. These systems and components can be further categorized as follows:

a. Systems and components mitigating the consequences of the accident;
b. Systems and components needed for maintaining integrity of the containment pressure boundary;
c. Systems and components needed for maintaining the core in a safe condition; and

d. Systems and components needed for monitoring the course of the accident.

As discussed previously, these systems and components are described as bringing the reactor to "safe shutdown" rather than "safe cold shutdown." Furthermore, the schedule for implementation has been changed to one that has been mutually agreed upon by the licensee and the NRC staff. Finally, the rule has been revised to indicate that the environmental conditions to be assumed for a hydrogen burn do not have to include the effect of local detonations if it is shown to the Commission's satisfaction that local detonations are unlikely to occur.

ANALYSES [§ 50.44(c)(3)(vi)]

In the proposed rule, the Commission included a description of three different approaches concerning the supplementary guidance to be provided for performing the required analyses for the design of the hydrogen control system. These were (a) analyses of different accident scenarios, (b) analyses of a single accident scenario with variation of key parameters, and (c) analyses using an "envelope of time histories of hydrogen and steam release rates" to be supplied by the Commission. The Commission requested comments concerning which of the approaches was preferred as well as suggestions regarding improvements or other alternatives.

There was no preponderance of comments leaning toward a particular approach; however, the first two approaches appeared to have greater support. Furthermore, many commenters felt that there should be flexibility in the approach to be used and in the selection of the accident scenarios. It was also suggested that the accident scenarios should be considered in order of importance using PRA techniques.

Based on the comments received and in consideration of the improved calculational data base now available, the Commission has decided to adopt the second approach; applicants and licensees need not use the first or third approaches.

In the selected approach, a base sequence will be identified by the licensee or applicant based on the hydrogen threat to containment integrity. Key aspects of this sequence should then be parametrically varied by the licensee or applicant in determining the acceptability of the containment response. This will provide a wider range of parameters than that of the selected base sequence alone. The acceptability of the analyses used in this approach depends on the selection and range of the parameters being varied. A range must be chosen which includes the effects of recovery from the degraded condition. It is expected that each applicant or licensee will review its analytical approach with the NRC staff and arrive at a mutually agreeable method for performing the analyses.

As an example, in the recent Sequoyah case[3], the applicant based its initial analysis on an accident sequence involving a small break LOCA followed by loss of ECCS (S_2D), with a typical average hydrogen release rate of about 20 pounds per minute, which the NRC staff considered to be representative of the accident. However, several concerns remained open. Among these were the possibilities that: (1) other scenarios might present schedules of steam and hydrogen release not covered by the analysis chosen; (2) steam inerting might occur at some time during the sequence allowing large concentrations of hydrogen to develop; (3) the recovery period might produce an exceptionally large burst of steam or hydrogen; and (4) hydrogen might be released after the loss of the ice heat sink.

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[3]NUREG-0011, Supplement No. 6, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant, Units 1 and 2," November 1982. Available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C.

In the Sequoyah case, the applicant broadened the studies to include higher rates of steam and hydrogen release and releases after the ice melted. The broadened calculations included hydrogen release rates as high as 6 lb. per second under representative steam conditions, with and without ice. It was shown that a representative selection of scenarios would be bounded by the broadened release rates, including an intermediate break LOCA with loss of ECC (S_1D), a small break LOCA with loss of containment heat removal (S_2G), a transient loss of main feedwater and loss of all AC power (T_BB_2), and a transient loss of main feedwater, loss of auxiliary feedwater and loss of the ECC (T_BLD). The staff concluded that the coverage of these additional scenarios was sufficient to assure that the hydrogen associated with a representative group of degraded core situations could be managed acceptably using the ignition systems.

As another example, in the McGuire case[4], hydrogen release rates up to 4.3 lb. per second under representative steam conditions were considered and the S_2D releases were analyzed with and without ice. The results were considered acceptable by the staff.

The staff has accepted ac-powered igniters without requiring a backup power supply in the two examples cited above. This judgment was based upon the staff's perception that the incremental risk reduction associated with provision of the igniter system backup power supply did not warrant the additional cost at these particular facilities. Provision of a backup power supply is not required by this rule.

It is apparent that applicants and licensees with conceptually different reactors may have to address other scenarios. The appropriate details for MARK III BWRs, for example, are currently being worked out through interaction between the NRC staff and applicants.

[4]NUREG-0422, Supplement No. 7, "Safety Evaluation Report Related to Operation of McGuire Nuclear Station Units 1 and 2," May 1983. Available for inspection ... Public Document Room at 1717 H Street, NW, Washington D.C.

Previously approved generic or reference analyses may be employed in lieu of plant specific analyses where the generic analyses can be shown to be applicable. It is believed that the adoption of the above approach will eliminate the need for repetitive calculation of accident scenarios.

REGULATORY ANALYSIS

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C. Single copies of the analysis may be obtained from Morton R. Fleishman, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 443-7616.

PAPERWORK REDUCTION ACT

This final rule imposes information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501, et seq.) These requirements were approved by the Office of Management and Budget. Approval Number 3150-0011.

REGULATORY FLEXIBILITY ACT

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "Small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this rule does not fall within the purview of the act.

LIST OF SUBJECTS IN 10 CFR PART 50

Antitrust, Classified information, Fire prevention, Incorporation by Reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting requirements.

Accordingly, notice is hereby given that, pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of Title 5 of the United States Code, the following amendments to 10 CFR Part 50 are published as a document subject to codification.

PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.58, 50.91 and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.10(a), (b), and (c) 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and

(c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In § 50.44, paragraph (c)(3) is revised by adding new paragraphs (iv), (v), (vi) and (vii) to read as follows:

§ 50.44 Standards for combustible gas control system in light water cooled power reactors.

* * * * *

(c)(3) ***

(iv)(A) [~~Effective-one-year-after-the-effective-date-of-the-rule,-or-the-date-of-issuance-of-a-license-authorizing-operation-above-5-percent-of-full-power,-whichever-is-later,~~] Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment [~~for-which~~] issued a construction permit [~~was-issued-prior-to~~] before March 28, 1979, shall [~~be~~] provide[~~s~~] its nuclear power reactor with a[~~n-acceptable~~] hydrogen control system justified by a suitable program[~~s~~] of experiment and analysis. The hydrogen control system must be capable of handling without loss of containment structural integrity an amount of hydrogen equivalent to that generated from a metal-water [~~the~~] reaction [~~of~~] involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). [~~with-water~~]

(B) Containment structural integrity must be demonstrated by use of an analytical technique that is accepted by the NRC staff. This demonstration must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. This method could include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on. Another method could include a showing that the following specific criteria of the ASME Boiler and Pressure Vessel Code are met:

(1) That steel containments [~~must~~] meet the requirements of the ASME Boiler and Pressure Vessel Code (Edition and Addenda as incorporated by reference in paragraph 50.55a(b)(1) of this part), specifically in Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, [~~except that~~] considering pressure and dead load alone (evaluation of instability is not required); and

(2) That concrete containments [~~must~~] meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone.

(C) Subsubarticle NE-3220, Division 1, and subsubarticle CC-3720, Division 2, of Section III of the ASME Boiler and Pressure Vessel Code, referenced in paragraphs(c)(3)(iv)(B)(1) and (c)(3)(iv)(B)(2) of this section, [~~These subsubarticles~~] have been approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. It is also available for inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street N.W., Washington, D.C.

Enclosure "F"

(D) If the hydrogen control system relies on post-accident inerting, the containment structure must be capable of withstanding the increased pressure:

(1) During the accident, where it [~~must~~] is acceptable to show that it does not exceed Service Level C Limits or the Factored Load Category (as [~~previously specified~~] described in paragraph (c)(3)(iv)(B) of this section [~~paragraph~~]); and

(2) Following inadvertent full inerting [~~that may occur~~] during normal plant operations, where it [~~must~~] is acceptable to show that it does not exceed either the Service Level A Limits of Subsubarticle NE-3220 (for a steel containment) or the Service Load Category of Subsubarticle CC-3720 (for a concrete containment).

(3) Modest deviations from the criteria in paragraph (c)(3)(iv)(D) of this section will be considered by the Commission if good cause is shown.

(E) If the hydrogen control system relies on post-accident inerting, the systems and components [~~equipment~~] required to establish and maintain safe [~~cold~~] shutdown and containment integrity must be designed and qualified for the environment caused by such [~~post-accident~~] inerting. Furthermore, inadvertent full inerting during normal plant operations must not adversely affect systems and components needed for safe operation of the plant.

(v) (A) Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment and each licensee with a pressurized light-water nuclear power reactor with an ice condenser type of containment [~~for which~~] issued a construction permit [~~was issued prior to~~] before March 28, 1979, for a reactor that does not rely upon an inerted atmosphere to control hydrogen inside the containment shall [~~be~~] provide[~~d~~] its nuclear power reactor with systems and components necessary to establish and maintain safe [~~cold~~] shutdown and to maintain containment integrity. These systems and components must be [~~that are~~] capable of performing their functions during and after [~~being exposed~~] exposure to the environmental conditions created by the burning [~~(or local detonation)~~] of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur.

(B) The amount of hydrogen to be considered is equivalent to that generated from [~~the~~] a metal-water reaction [~~of~~] involving 75% of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). [~~with water This requirement shall be effective as follows: for each boiling light-water nuclear power reactor with a Mark III type containment and each pressurized light-water nuclear power reactor with an ice condenser type containment, on [one year after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later; for every other light-water nuclear power reactor that must meet this requirement, on [two years after the effective date of the rule] or the date of issuance of a license authorizing operation above 5 percent of full power, whichever is later.~~]

(vi) (A) Each applicant for or holder of an operating license for a boiling light-water nuclear power reactor with a Mark III type of containment or for a pressurized light-water nuclear power reactor with an ice condenser type of containment issued a construction permit before March 28, 1979, shall [~~analyses-shall-be-performed-and~~] submit[~~ted~~] an analysis to the Director of the Office of Nuclear Reactor Regulation. [~~for-each-light-water nuclear-power-reactor,-for-which-a-construction-permit-was-issued-prior-to March-28,-1979,-to-evaluate~~]

(B) The analysis required by paragraph (c)(3)(vi)(A) of this section must:

(1) Provide an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75% [~~percent~~] of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume) and include [~~with-water-including~~] consideration of hydrogen control measures as appropriate; [~~Each-analysis-must~~]

(2) Include the period of recovery from the degraded condition;

(3) Use [~~the~~] accident scenarios [~~to-be-used-in-the-analyses-must-be~~] [~~acceptable-to~~] that are accepted by the NRC staff. These scenarios must be accompanied by sufficient supporting justification to show that they describe the behavior of the reactor system during and following an accident resulting in a degraded core. [~~The-scope-and-implementation requirements-for-The-analyses-for-the-various-types-of-light-water-nuclear power-reactors-are-as-follows:~~

~~(A)--For-each-boiling-light-water-nuclear-power-reactor-with-a-Mark-III type-containment-and-each-pressurized-light-water-nuclear-power-reactor-with an-ice-condenser-type-containment,-analyses-shall-be-performed-that-justify the-selection~~]

(4) Support the design of the hydrogen control system selected [~~required-by-§-50.44~~] under paragraph (c)(3)(iv) of this section; and, [~~These analyses-shall-be-completed-and-submitted-by-[one-year-after-the-effective-date~~

~~of-the-rule]-or-the-date-of-issuance-of-a-license-authorizing-operation-above~~ ~~5-percent-of-full-power,-whichever-is-later.]~~

(5) Show that, for those reactors described in paragraph (c)(3)(iv) of this section that do [~~For-each-light-water-nuclear-power-reactor-that-does~~] not rely upon an inerted atmosphere to control hydrogen inside the containment: [~~analyses-shall-be-performed-to-show-that~~]

(i) The containment structural integrity as [~~defined~~] described in [~~§-50.44~~] paragraph (c)(3)(iv) of this section will be maintained; and

(ii) Systems and components necessary to establish and maintain safe [~~cold~~] shutdown and to maintain containment integrity will be capable of performing their functions during and after [~~being-exposed~~] exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such detonations can be shown unlikely to occur. [~~These-analyses-shall-be-completed-and-submitted-as-follows:-for-each-boiling-light-water-nuclear-power-reactor-with-a-Mark-III-type-containment-and-each-pressurized-light-water-nuclear-power-reactor-with-an-ice-condenser-type-containment,-by-[one-year-after-the-effective-date-of-the-rule]-or-the-date-of-issuance-of-a-license-authorizing-operation-above-5-percent-of-full-power,-whichever-is-later;-for-every-other-light-water-nuclear-power-reactor-for-which-these-analyses-are-required,-by-[two-years-after-the-effective-date-of-the-rule]-or-the-date-of-issuance-of-a-license-authorizing-operation-above-5-percent-of-full-power,-whichever-is-later.~~]

Enclosure "F"

(vii) Implementation. By [insert a date 180 days after the effective date of the amendment], each applicant for or holder of an operating license subject to the requirements of paragraphs (c)(3)(iv), (v) and(vi) of this section shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a schedule for meeting those requirements. A final schedule for meeting the requirements of paragraphs (c)(3)(iv), (v) and (vi) of this section shall then be mutually agreed upon by the applicant for or holder of the operating license and the NRC staff.

Dated at Washington, D.C. this _________ day of _______________, 1983.

For the Nuclear Regulatory Commission,

____

Samuel J. Chilk
Secretary of the Commission

Enclosure "F"

EXHIBIT 2

PROPOSED STAFF POSITION

The generic issue of degraded core accident H_2 control for BWR Mark III containments is a matter which is the subject of ongoing research and review by the NRC and the industry. The requirements for enhanced hydrogen control to deal with degraded core accident H_2 releases which have been imposed on the owners of ice condenser and Mark III plants have been developed to provide assurance that these small volume, low design pressure facilities could successfully accommodate those accidents whose chief threat to safety is derived from large hydrogen generation and release.

The NRC believes that the mission of reducing the risks associated with large H_2 releases may best be served by continuing to require utilities to provide protection for accidents involving the release of H_2 corresponding to a fuel cladding reaction of up to 75%. There is no current binding requirement upon the rate at which H_2 shall be assumed to be released. Therefore, utilities may utilize conservative hydrogen release rates which are representative of physical processes including those which may limit the release rates.

Based on our understanding of the preliminary assessment of the thermal environment as determined by the BWR HCOG we believe it prudent that positive action be taken to improve the capability of essential equipment to survive the effects of hydrogen burning. Essential equipment located in the vicinity of the suppression pool or other regions subjected to severe environments should be relocated wherever feasible. As an alternative for equipment which may not be moved additional thermal protection should be provided.

Additionally, it is our conclusion that the BWR HCOG should continue the investigation of hydrogen combustion via testing in a larger scale facility, such as a ¼ scale test. It is important that uncertainties in the characterization of the containment atmosphere response be held to an acceptable minimum level.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 16 1982

EXHIBIT 3

Docket Nos.: 50-440/441

Mr. Dalwyn R. Davidson
Vice President
System Engineering and Construction
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Dear Mr. Davidson:

Subject: Request for Additional Information Regarding Degraded Core Hydrogen Control for the Perry Nuclear Power Plant (Units 1 and 2)

The NRC staff has identified a number of areas pertaining to the Perry hydrogen ignition system where additional information is required. The information required is addressed in Enclosure (1). Please advise the project manager, John J. Stefano, when we may expect to receive your responses within five (5) days after receipt of this letter.

Your prompt attention to this request will be most appreciated.

Sincerely,

A. Schwencer /for

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page

Perry

Mr. Dalwyn R. Davidson
Vice President, Engineering
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

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Ashtabula County Courthouse
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REQUEST FOR ADDITIONAL INFORMATION FOR USE IN THE INTERIM EVALUATION OF THE HYDROGEN IGNITION SYSTEM FOR PERRY MARK III CONTAINMENT.

480.49 Provide a detailed description of the Hydrogen Ignition System (HIS) and its power supplies; include the total number of igniters, the number of circuit breakers, and a simplified electrical system schematic showing all the above stated items and any other major component.

480.50 Provide the following igniter information:

a) Vendor;

b) Model;

c) Qualification Program; and

d) Design Criteria.

480.51 Provide a detailed description of the preoperational surveillance and periodic testing programs of the HIS.

a) How will the system be tested? Specifically, what indicates that a particular igniter is or is not functioning properly?

b) Specify the frequency of testing.

c) Are hydrogen detectors to be used as part of the HIS? If so, please specify the types of detectors, number, location of sampling ports, system response time, and testing format and frequency.

480.52 Describe the glow plug igniter selection program; i.e., how will actual igniters be selected for installation in the assemblies.

480.53 Please provide construction drawings for several "typical" igniter mounts in the wetwell and containment regions. Also, provide a complete list of the approximate elevation,azimuthal and radial coordinates for each igniter in containment, and the corresponding elevation coordinate of the nearest ceiling (include the make-up of the nearest ceiling, i.e., open, solid, grated). Indicate whether all enclosed regions of the containment are served by redundant igniters.

480.54 For each floor within the containment annular region and the drywell, please provide information on the cross-sectional flow area and identify the various areas as gratings, solid regions, or equipment blockage.

480.55 Discuss the design adequacy of the igniter assembly to withstand pool swell events and the drywell negative pressure response.

480.56 Please provide full size sectional drawings of the containment and identify the location of each igniter, it's electrical division, and location of vacuum breaker lines and purge compressor lines.

480.57 Discuss the consideration of local impingement of break spray on the igniter assembly.

480.57 Evaluate whether the sheet-flow into the wetwell impinges on any igniter.

480.58 Discuss the effect of submergence on igniter performance. For those igniters which will continue to be necessary, describe the testing which will be performed to assure igniter performance before, during and after being subjected to submergence conditions.

480.59 Considering the actuation criteria of safety systems including operator action:

a) Under what conditions are the sprays activated?

b) How long after the sprays are actuated does the spray system attain full flow rate?

c) When during an emergency situation would the HIS be activated?

d) What role, if any, would the hydrogen recombiner play with respect to the HIS?

e) What are the emergency procedure criteria for post accident containment purge/vent?

480.60 Regarding the containment atmosphere mixing mechanisms:

a) Describe the flow rate of the ventilation system in the containment/wetwell regions.

b) What are the elevations and radial positions of the spray rings?

c) Which spray ring operates when a single RHR loop is operating and what is the flowrate under such conditions? Does the spray water contain chemical additives?

d) Describe any sprays, fans or other systems that could move air in the annular wetwell region and estimate the air velocities in the region due to these systems.

480.61 Briefly explain the workings of the "drywell purge system" including purge compressors and vacuum breakers. Estimate flowrates from the system during an accident. Describe the operation of the Combustible Gas Control System (CGCS) during hydrogen burns (including a discussion of the logic for the purge compressors and vacuum breakers).

480.62 In Mark III containments, the sprays are not made up of dedicated components but share pumps with other subsystems intended to deliver water cool to the core. A basic postulate of degraded core accidents is that cooling water to the core is unavailable (e.g., cooling pumps unavailable). It appears inconsistent to assume that components of a core cooling system would be available to provide containment spray flow. Therefore, provide justification for the assumption that sprays are available.

480.63 Provide the following plant specific CLASIX-3 containment transient analysis*:

(1) SORV Base Case Transient;

(2) Small Break LOCA Base Case;

(3) Small Break LOCA with a burn criterion of 10% hydrogen concentration and 100% complete combustion in the containment assuming a minimum oxygen concentration of 6.5% in the drywell; and

(4) Small Break LOCA with a burn criterion of 10% hydrogen concentration, 100% completeness and a flame speed of 12 fps.

*Note: If spray availability is questionable, do not consider them in the containment analysis. [Even though the HCOG sensitivity study (HGN-001, Jan., 1982) presents a "no spray" SORV case in which the compartment pressures are relatively low with respect to the SORV base case. This is so, since the containment oxygen concentration is slightly below the five percent molar concentration criterion, which results in the absence of a containment burn. However, if the transient is extended in time, the oxygen concentration would exceed five percent and trigger a containment burn. Hence, the "no-spray" SORV case may be more severe than the SORV base case with respect to peak temperatures and pressures.] If credit is taken for spray availability provide and justify the following inputs to the CLASIX-3 analysis:

(1) flowrates per spray train;
(2) number of spray trains to be used;
(3) containment to wetwell carry-over fraction;
(4) percentage of carry-over which is in droplet form and sheet-flow;
(5) droplet mass mean diameter;
(6) drop efficiency; and
(7) sheet-flow efficiency.

480.64 Identify the most severe pool dynamic load conditions in the wetwell when considering the effect of hydrogen combustion in the drywell. Discuss the effects of the pool dynamic loads on the containment structures and the essential equipment within the zone of influence. Also, evaluate in a similar manner the most severe drywell negative differential pressure transient and the pool dynamic loads created within the drywell.

480.65 Are there any accident sequences that might lead to the introduction of hydrogen and steam directly into the containment without having passed through the suppression pool?

480.66 Provide an evaluation of the potential and consequences of flame acceleration in the various containment regions including consideration of circumstances leading to transition to detonation.

480.67 Provide an analysis of the concomitant effects of the largest conceivable containment detonation which could occur. Demonstrate that the effects of such an event could be safely accommodated by structures and essential equipment. Also, provide an estimate of the limiting size of a cloud of detonable gas with regard to the structural capability of the containment shell and the drywell.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 2 2 1983

Docket No.: 50-416

EXHIBIT 4

Mr. James P. McGaughy, Jr.
Vice President - Nuclear Production
Mississippi Power & Light Company
P. O. Box 1640
Jackson, Mississippi 39205

Dear Mr. McGaughy:

Subject: Mark III BWR Hydrogen Control

On June 29, 1983, the Mark III BWR Hydrogen Control Owner's Group (HCOG) presented to the staff, results from the hydrogen control R&D program. The information presented at this meeting has raised concerns, described below, that must be appropriately addressed and resolved by MP&L before we will be in a position to recommend operation above 5% of full power.

Based on a preliminary analysis of the HCOG data presented, we believe that, for unmitigated accidents, your analysis and results may be correct. That is, it is very unlikely that large amounts of metal water reaction would occur before core melt in a loss of coolant accident that had little or no water provided to the reactor other than the initial inventory. However, our position on this matter has been that hydrogen control systems are to be based on a variety of degraded core accidents, including mitigated types of accidents (e.g. TMI-2) and must be capable of handling a 75% M-W reaction (active cladding), with options on H_2 /steam releases. In other words, you are required to consider accidents that move more slowly toward core melt but stop short of core melt because of some intermittent availability of water, sufficient to keep the core from melting and sufficient to fuel the metal water reaction over an extended period of time, perhaps several hours. Events of this sort could be arrested short of core melt and still yield large amounts of metal water reaction (up to 75%). Accordingly, we request that you specifically address the consequences of sequences of events that may occur in your plant with varying rates of H_2 production for a 75% M-W reaction.

The available scaled test data show that for certain of the presented rates of H_2 production, very high local ambient containment temperatures would be produced from the H_2 burns in which survivability of some vital equipment may be questionable. We question the adequacy of 1/20 scale tests performed to date for providing reliable information on the effects of H_2 burns in a Mark III containment. Further testing may be required in order to properly evaluate equipment survivability and other aspects relative to this issue.

Furthermore, the staff is concerned with the calculated high drywell atmospheric temperature. This matter also requires prompt resolution. A copy of our request on this concern is enclosed and was telecopied to you on July 21, 1983.

Based on the review performed to date on these matters, the staff concludes that MP&L must provide further technical justification for interim operations with the distributed ignition system to support operation above 5% power. The staff will restate this conclusion generically at the meeting on July 28, 1983, with the BWR Hydrogen Control Owners Group.

If you have any questions concerning this issue, please contact me.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosure:
As stated

cc: See next page

REQUEST FOR ADDITIONAL INFORMATION RELATED TO DEGRADED CORE HYDROGEN CONTROL

As part of our review of the adequacy of the CLASIX-3 code, we have performed certain confirmatory analyses with the Contempt codes. Based on a preliminary evaluation of the results from our analysis and the CLASIX-3 results, there appears to be some degree of non-conformance to the provisions of NUREG-0588. The non-compliance seems to be related to the passive heat-sink, heat-transfer model assumed in CLASIX-3.

As a result of this apparent non-conformance to the provisions of NUREG-0588 the temperature profiles presented in the CLASIX-3 containment response sensitivity studies (correspondence Nos. HGN-001, and AECM-83/0212) are believed to underestimate the compartment temperature atmospheric conditions.

Since the methodology described in NUREG-0588 is generally recognized as an acceptable approach for the above concern, describe and justify deviations from NUREG-0588 for the passive heat-sink, heat-transfer assumptions that have been used in the following areas:

1) the temperature difference used with the heat-transfer film coefficients for both saturated and super-heated atmospheres; 2) the analytical model and assumptions used to account for condensate removal from the heat sink surface; and 3) the energy removal associated with condensed mass.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 30 1984

Docket Nos.: 50-440
and 50-441

EXHIBIT 5

Mr. Murray R. Edelman
Vice President - Nuclear Group
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Dear Mr. Edelman:

Subject: Request for Additional Information Relative to the Mark III Containment Design Ultimate Pressure Capability for the Perry Nuclear Power Plant (Units 1 and 2)

The NRC staff is evaluating the ultimate capability of the positive and negative (reverse) pressure differentials of the Mark III containment design, including the structural capability of the drywell and steel head for positive and negative pressures. Based on its review of the Perry FSAR, the staff is in need of the following information in the performance of their evaluations:

1. The ultimate capacity in terms of psid of the containment shell for negative (reverse) presssure.

2. The ultimate capacity in terms of psid of the drywell pressure retaining boundary for positive and negative pressure. If the refueling pool is filled with water during operation, the effect of the water on the drywell including the steel head should be considered. In all cases, the structural region or items which limit the pressure retaining capability should be identified as well as the particular failure mechanism.

3. The maximum calculated negative containment pressure which would result from complete combustion of an amount of hydrogen corresponding to a 75% metal-water reaction (oxygen depletion), and the subsequent cooling of the containment atmosphere. Include a description of the analytical model and justify the assumptions used to determine the internal containment pressure response; e.g., by addressing the conservatism with respect to plant-specific applications. It is anticipated that, in most cases, the calculated containment negative pressure differential would exceed the design value. Therefore, in providing your response, you may elect to demonstrate that:

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a. The calculated external containment pressure capability value bounds the above transient which is determined to be the most limiting pressure differential. Thus, the containment has the capability to withstand the most severe external pressure that might result following a hydrogen combustion event.

b. Alternatively provide a description of the design provisions regarding automatic and manual means for relieving reverse pressure differentials; e.g. by use of vacuum breakers. The discussion should include the operating procedures concerning monitoring of containment pressure, and operator actions to relieve reverse pressure differentials following onset of an accident. In addition, (1) the system that is relied upon to relieve reverse pressure differentials must be shown to survive the consequences of burning the hydrogen generated from a 75% metal-water reaction, and (2) an analysis should be included to show the effectiveness of this system when considering the above stated assumptions.

In providng your responses to the above information items, please identify items 1 and 2 as responding to Q220.30 and Q220.31, and item 3 (and subparts) as responding to Q480.52, for eventual documentation in a future FSAR amendment.

If there are any questions or clarifications required, please address them to the project manager, John J. Stefano. Also advise the project manager when we may expect to receive a response to these items within 7 days after receipt of this letter.

Sincerely,

B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

cc: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 30 1984

Docket Nos.: 50-440
and 50-441

EXHIBIT 6

Mr. Murray R. Edelman
Vice President - Nuclear Group
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Dear Mr. Edelman:

Subject: Request for Additional Information Regarding Hydrogen Control for the Perry Nuclear Power Plant (Units 1 and 2)

As a part of its continuing review of hydrogen control for Mark III containment design plants during postulated degraded core accidents, the NRC staff has identified the need for additional information on several matters. The information being requested by the enclosed questions pertain to the CLASSIX-3 Code which has been used by the Hydrogen Control Owners Group to support the licensing activities associated with Mark III plants; e.g., determining the environmental conditions against which equipment survivability is to be evaluated.

Your response to the enclosed questions should be identified as answering Q.480.55 through Q 480.57 for eventual documentation in the Perry FSAR. Please advise the Project Manager when we may expect to receive your responses to the enclosed questions within 7 days after receipt of this letter.

Sincerely,

B. J. Youngblood

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
As stated

cc: See next page

REQUEST FOR ADDITIONAL INFORMATION RELATED
TO DEGRADED CORE HYDROGEN CONTROL
FOR PERRY

480.55 It is the intent of the Mark III owners to use the HCOG quarter-scale tests (which focuses on diffusion-type burning within the wetwell region) and plant specific/HCOG CLASIX-3 analyses (which focuses on discrete-type burning within the containment), to determine the most severe thermal environment within the containment and drywell for purposes of demonstrating equipment survivability. Since the present passive heat sink modeling in CLASIX-3 tends to underestimate the compartment atmosphere temperatures and since CLASIX-3 appears to be in non-conformance with the provisions of NUREG-0588, the CLASIX-3 containment response sensitivity studies (correspondence No. HGN-001) should not be used as the basis for determining the most severe compartment temperature conditions. In view of this concern, the present version of CLASIX-3 is inappropriate.

Since the methodology described in NUREG-0588 is generally recognized as an acceptable approach for addressing equipment qualification, describe and justify if there are deviations from the provisions of NUREG-0588 with regard to the passive heat-sink and heat-transfer assumptions that will be used for plant specific analyses in the following areas:

1) the temperature difference used with the heat-transfer film coefficients for both saturated and super-heated atmospheres;

2) the analytical model and assumptions used to account for condensate removal from the heat sink surface; and

3) the energy removal associated with condensed mass.

480.56 For each postulated degraded core sequence, (i.e., SORV and drywell break initiated events), provide an evaluation of the impact on the drywell atmosphere environment when considering heat losses from the reactor vessel and its associated piping (e.g., SRV lines). Provide and justify assumptions used in your evaluation, e.g., convective and radiative heat transfer parameters.

480.57 According to the BWR/6 Standard Technical Specifications, periodic low pressure leak testing of the drywell is required. The acceptance criterion is that the leakage shall be less than or equal to 10% of the maximum allowable $A/\sqrt{K}$ (i.e., approximately 1 ft^2). 1.6
Thus, the maximum allowable leak rate is equivalent to roughly 4000 SCFM at 3 psi pressure differential. Provide an evaluation of the consequences within the drywell and the containment by the combustion of hydrogen when considering the drywell bypass leakage (include mechanistically the effects of upper pool dump and pool drawdown).

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing were served by deposit in the U.S. Mail, first class, postage prepaid, this 3rd day of October, 1984 to those on the service list below.

Susan F. Hiatt
Susan L. Hiatt

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