

THE GENERIC
SAFETY EVALUATION REPORT RELATING TO
THE MARK III CONTAINMENT HYDROGEN CONTROL

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1 INTRODUCTION AND BACKGROUND

Following a loss-of-coolant accident in a light-water reactor (LWR) plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in both the reactor core and the containment sump; (3) the corrosion of certain materials within the containment by sprays; and (4) any synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on protective coatings, and electric cable insulation.

To provide protection against this possible hydrogen as a result of an accident, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," and GDC 41, "Containment system cleanup," Appendix A to 10 CFR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained. Conventional hydrogen control systems (e.g., hydrogen recombiners) historically have been installed to provide the capability to control the relatively low rate of hydrogen accumulation (or oxygen accumulation in inerted containments) resulting from radiolytic decomposition of water, corrosion of metals inside containment, and environmental effects on coatings and insulation. However, the net free volume inside containment (or inerting of the containment volume) is used to control the rapid hydrogen production resulting from a metal-water reaction of the fuel cladding. That is, the containment volume is large enough so that the hydrogen generated early would not reach the lower limit of flammability (or the inerting would prevent combustible mixtures). The rationale for this approach is that the rate of hydrogen release resulting from cladding reaction was assumed to be too rapid (on the order of minutes) following a postulated accident to allow for an active control system. Thus, hydrogen control systems (recombiners) would only be actuated later in the event to control the slow hydrogen/or oxygen release associated with radiolysis and reaction of materials inside containment.

To quantify the metal-water source for design-basis accidents, 10 CFR 50.44, codified in October 1978, requires that the combustible gas control system be capable of handling the hydrogen generated from five times the amount calculated in demonstrating compliance with 10 CFR 50.46 or the amount corresponding to the total reaction of the cladding to a depth of 0.23 mils, whichever amount was greater. Typically, this would translate to a 1 to 5% metal-water reaction of the active clad.

However, the accident at Three Mile Island Unit 2 (TM1-2) on March 28, 1979, resulted in a metal-water reaction that involved approximately 45% of the fuel cladding (i.e., about 990 lbs), which resulted in hydrogen generation well in excess of the amounts specified in 10 CFR 50.44. The combustion of this hydrogen produced a significant pressure spike inside containment. As a result, it became apparent that additional design measures were needed to handle larger hydrogen releases, particularly for smaller volume containments and those with

lower design pressures. The Nuclear Regulatory Commission (NRC) determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be considered in plant design. An advance notice of the rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980. In addition, a new unresolved safety issue was instituted (A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment") to evaluate this new concern.

To formalize its requirements for additional hydrogen control measures to deal with degraded core accidents affecting pressurized-water reactors (PWRs) with ice condenser containments and BWRs with Mark III containments, the NRC published an amendment to the hydrogen rule (10 CFR 50.44) on January 25, 1985 (50 FR 3498). The amended rule required that a hydrogen control system be provided and that the system be capable of accommodating, without loss of containment structural integrity, the amount of hydrogen generated from a metal-water reaction of up to 75% of the active fuel cladding. In addition, systems and components necessary to establish and maintain safe shutdown must be capable of performing their function regardless of hydrogen burning.

Pursuant to the provisions of the rule, each utility with a Mark III containment has installed a hydrogen ignition system (HIS) and submitted a preliminary analysis and a schedule for meeting the full requirements of the rule. The affected plants with Mark III containments are Grand Gulf Nuclear Station, River Bend Station, Perry Nuclear Power Plant, and Clinton Power Station. The staff's interim evaluations of these initial plant responses are documented in supplements to the safety evaluation reports for each of these plants (NUREG-0831, NUREG-0989, NUREG-0887, and NUREG-0853, respectively).

These responses were aided by the efforts of the Mark III Containment Hydrogen Control Owners Group (HCOG), formed in May 1981 by the utilities with Mark III containments to collectively perform testing and analyses to demonstrate the effectiveness and reliability of the hydrogen ignition systems.

In addition, each licensee with a Mark III containment is required to provide a final analysis [10 CFR 50.44(c)(3)(vii)(B)] to confirm the conclusions of the preliminary analysis and/or, if necessary, to institute modifications to ensure compliance with the rule. The scope of this analysis is specified in 10 CFR 50.44(c)(3)(vi)(B). The generic findings from the HCOG's program will be utilized for this final analysis, supplemented by plant-specific design considerations as addressed in the licensee's IPE program.

The following staff evaluation focuses on the assessment of the completed generic testing and analyses performed by the HCOG in support of the plant unique analysis. The HCOG activities have been summarized in a topical report transmitted by letter dated February 23, 1987, correspondence identification HGN-112-NP, "Generic Hydrogen Control Information for BWR-6 Mark III Containments." The topical report is a summary document of all of the individual generic submittals that have been sent to the staff by the HCOG. It should be noted that HCOG correspondence identification designators with a "P" suffix (HGN-XXX-P) are proprietary to HCOG. Whereas, those without a suffix or with an "NP" suffix are nonproprietary.

As part of the review of the HCOG program, the staff obtained technical assistance from the Sandia National Laboratory (SNL), the principal contractor for the NRC research program on hydrogen control and combustion phenomena. SNL provided the NRC with an independent assessment of technical issues contained in selected HCOG submittals pertaining to hydrogen behavior.

The staff evaluation of the generic considerations of the hydrogen control system for the Mark III containment can best be understood if categorized as follows:

- general description of the hydrogen ignition system
- combustion and igniter testing
- containment structural capacity
- degraded core events and hydrogen generation
- containment response-analytical modeling
- survivability of essential equipment
- overall conclusions

Therefore, the following discussion will follow this general outline.

2 GENERAL DESCRIPTION OF THE HYDROGEN IGNITION SYSTEM

The regulation, 10 CFR 50.44(c)(3)(iv)(A), states:

Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment..., shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program 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 reaction involving 75% of the fuel cladding surrounding the active fuel region [excluding the cladding surrounding the plenum volume].

The concept employed by the licensees with a Mark III containment, and similarly the ice condenser licensees, is to intentionally ignite hydrogen generated inside containment. This method precludes buildup of relatively high concentrations of hydrogen during degraded core accident scenarios.

To accomplish this early ignition a hydrogen ignition system (HIS) is installed in each of the four plants with Mark III containments. The HIS is a system which consists of approximately 100 igniter assemblies distributed throughout the drywell and containment regions.

The main element of the igniter assembly is the Model 7G thermal igniter glow plug (commonly used in diesel engines) manufactured by the General Motors AC Division. Each Mark III containment has an identical igniter assembly design. Each igniter is powered directly from a 120/12-V stepdown transformer and designed to provide a minimum surface temperature of 1700°F. The igniter assembly (see Figure 2.1) consists of a 1/8-inch-thick stainless steel box that contains the transformer and all electrical connections and is manufactured by the Power Systems Division of Morrison Knudsen. Igniter assemblies are Class 1E, seismic Category I, and meet the requirements of the Institute of Electrical and Electronics Engineers (IEEE) Std 323-1974 and NUREG-0588 for environmental qualification.

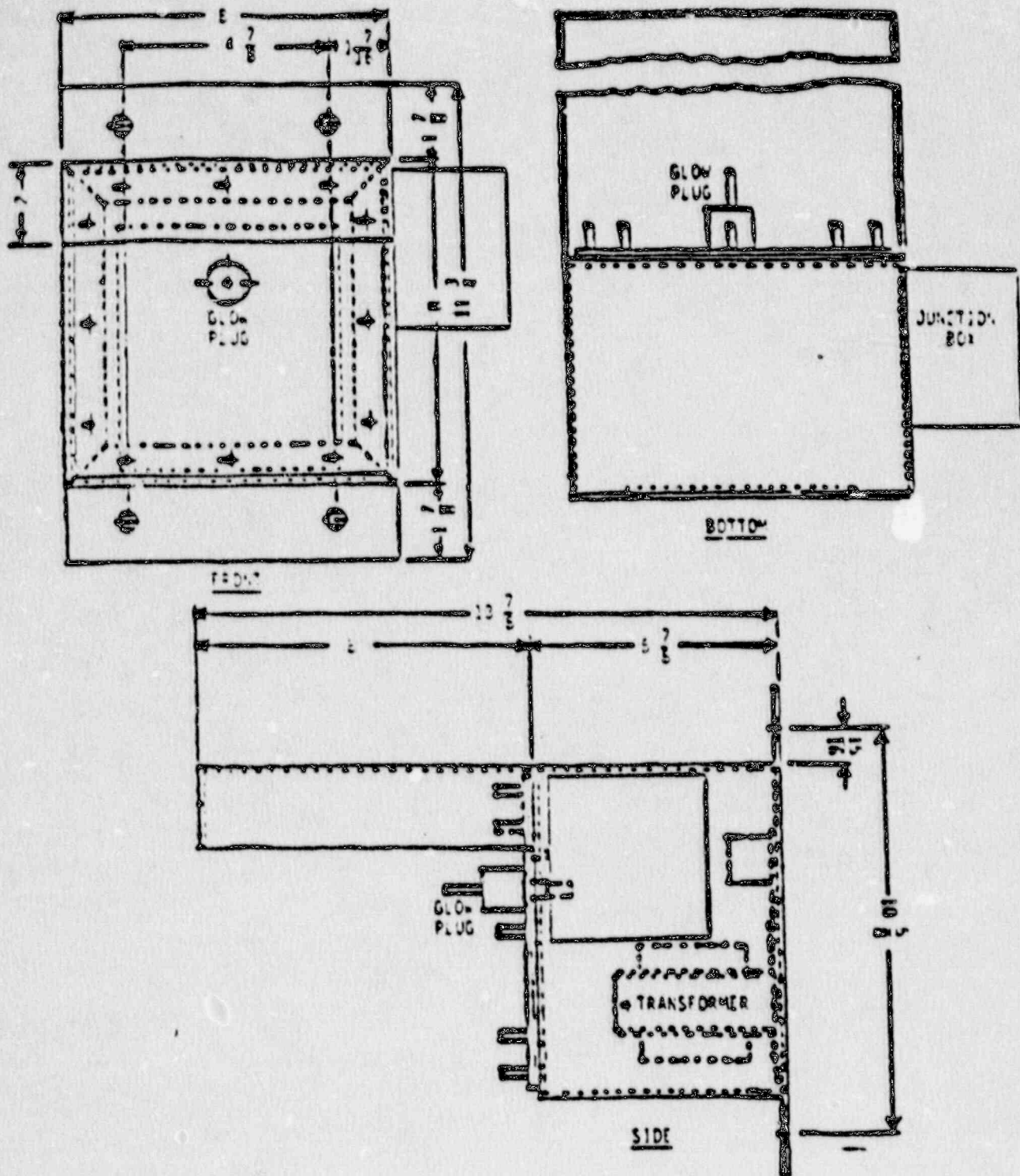


Figure 2.1 General hydrogen igniter assembly

The igniter assemblies are divided into two groups; each group being powered from a separate Class 1E division power supply. The intent is to have at least two igniters located in each enclosed volume or area within the containment that could be subject to possible hydrogen pocketing, and each igniter be powered from a separate power division. In open areas within the containment and drywell regions, igniter assemblies at the same elevation are designed to have alternating power division sources. Igniters are separated by about 30 feet when both engineered safety feature (ESF) power divisions are operable or by about 60 feet when one power division is inoperable. Igniter placement is designed to be more widely spaced in the large open regions, such as above the refueling floor, and in the lower regions of the drywell that are subject to flooding. Requirements of placement as well as other parameters of the system are contained in the technical specifications for the hydrogen ignition system, as proposed by the HCOG, and addressed in Appendix A of this report.

Each igniter power division has a corresponding onsite emergency diesel generator. Incorporation of the emergency diesels into the design addresses the question of igniter power for many sequences, but not for station blackout conditions. Under blackout conditions, the HIS would not be operable. On the basis of a separate evaluation of this possibility in the context of the NRC Containment Performance Improvement (CPI) program, the staff has recommended that the vulnerability to interruption of power to the hydrogen igniters be evaluated further on a plant-specific basis as part of the Individual Plant Examinations of the Mark III plants. (See Generic Letter 88-20, Supplement 3)

The HIS is designed for manual actuation from the main control room. Actuation by the operator is required by plant emergency procedures when the reactor pressure vessel (RPV) water level reaches the top of active fuel (TAF). The proposed combustible gas control emergency procedures for plants with Mark III containments are further discussed in Appendix B.

By letter dated March 5, 1986 (HGN-073), the HCOG provided the following justification to support manual actuation. Actuation is linked to indication of the RPV water level, which is a key safety parameter and is closely monitored by the operators. Also, HIS actuation requires only the positioning of two hand-switches. Furthermore, operators should not hesitate to energize the system during accident scenarios in which the hydrogen threat is uncertain or marginal because there would be no adverse effect on the plant as a result of unnecessary igniter actuation. Time available to actuate the HIS is the other significant parameter. Based on the hydrogen events considered, the HCOG estimated this time to be approximately no less than 25 minutes; that is, after the water level reaches TAF to the lower hydrogen flammability limit reached in the wetwell volume. The HCOG also noted that hydrogen would migrate to the upper portions of the containment before the wetwell reaches hydrogen combustion conditions. This effect was seen in the quarter-scale tests. Therefore, the time interval is expected to be somewhat greater than 25 minutes.

Because (1) manual actuation is a simple task, (2) the operator has sufficient time to perform the task, (3) and there are no negative effects if the system is inadvertently or unnecessarily actuated, the staff finds manual operator actuation acceptable.

HIS Design Assessment

The staff finds these hydrogen igniter systems currently installed in the plants with a Mark III containment to be acceptable, with the caveat that the vulnerability to interruption of power to the hydrogen igniters be further evaluated on a plant-specific basis as part of IPEs of the Mark III plants. (See Supplement No. 3 to Generic Letter 88-20)

3 COMBUSTION/IGNITER TESTING

Numerous research programs have been conducted since 1980 to better understand hydrogen combustion behavior and the performance of ignition devices. Because these programs were varied in scope, Sandia National Laboratory (SNL) summarized the findings of recent hydrogen combustion test programs in NUREG/CR-5079. This report also provides additional background information and insights related to hydrogen behavior.

The specific test programs considered necessary by HCOG to support the unique plant characteristics of a Mark III containment are discussed below.

3.1 Small-Scale Tests

Small-scale hydrogen combustion tests were performed at Whiteshell Laboratories and documented by the HCOG in a letter dated June 7, 1984 (HGN-017-NP). The program was intended to investigate ignition and combustion behavior of mixtures predominantly composed of hydrogen and steam, i.e., with limited available oxygen. This condition may exist for a postulated drywell break event in which air is initially swept from the drywell and then later reintroduced into the steam-hydrogen environment. These tests confirmed that such hydrogen-air-steam mixtures can be successfully ignited as long as the oxygen concentration exceeds approximately 5-6 volume percent.

A 1/20th-scale Mark III hydrogen combustion program was conducted by Acurex Corporation and documented by the HCOG in a letter dated February 9, 1984 (HGN-014-NP). The objective of this program was to provide a visual record of hydrogen combustion behavior in a 360-degree model of a Mark III containment. Modelling included the suppression pool and major blockages in the annular region between the drywell and outer containment walls. Hydrogen was admitted through simulated quenchers and/or vents into the suppression pool and ignited by prototypical ignition sources.

The most important result obtained from the 1/20-scale test was the confirmation of continuous hydrogen burning in the form of steady diffusion flames above the suppression pool. The significance of this mode of hydrogen burning is the observed severe thermal loads that occur near the diffusion flames and could threaten the integrity of the containment and equipment. Diffusion flames were observed when hydrogen flow rates of 0.4 lb/sec (full-scale equivalent) or

greater were used. Combustion was initiated by the igniters and rapidly propagated to the pool surface and formed steady diffusion flames that were anchored at the surface of the pool and located above the submerged spargers that released the hydrogen. Hydrogen burning was observed to intensify as the hydrogen injection flow rate was increased, as evidenced by taller flames and higher temperatures.

As part of the 1/20-scale program to determine the sensitivity of scaling, a 1/5-scale single-sparger mockup was constructed. A significant result of these tests was that approximately one-half the flame height of what would have been predicted based on results of 1/20-scale tests was observed. On the basis of these tests, it became necessary for the HCOG to pursue a larger-scale test program to obtain thermal environmental data more representative of a Mark III containment. Subsequently, the HCOG undertook an extensive program to better define the conditions that could exist during a degraded core accident. A major element of this effort was the quarter-scale Mark III containment combustion test program.

3.2: Quarter-Scale Test Facility

The quarter-scale test program became the major element of the HCOG's hydrogen research program. The primary objective of this program was the investigation and characterization of the environment that could result from diffusive burning on the suppression pool in a Mark III containment. Ultimately, the information gathered from the quarter-scale test facility (QSTF) would be used in determining the survivability of select equipment. Test facility description and the combustion test results may be found in the HCOG's letters HGN-098-P, July 18, 1986; HGN-115-NP, February 10, 1987; and HGN-121-P, July 22, 1987.

The test facility is a quarter-linear-scale model of a Mark III containment, designed and constructed by Factory Mutual Research Corporation (FMRC), and located in West Gloucester, Rhode Island. The test enclosure is designed to operate at pressures up to 40 psig and consists of an outside tank, 31.5 feet in diameter, 49.4 feet high, containing a smaller tank, about 21 feet in diameter by 23 feet high. The space between the two tanks is the test volume; which contains floors and other large blockages simulating the obstructions that exist in the actual containments. Because of the unique features of the four plants studied, modular construction of the annular floors is used to modify the vessel interior when needed. At the bottom of the two tanks, the suppression pool is simulated. Several views of the facility are shown in Figures 3.1 and 3.2.

Facility design features include:

- containment sprays
- simulated loss-of-coolant-accident (LOCA) vents (top row only, numbering 48)
- simulated spargers (uniformly spaced every 15 degrees azimuthally and totalling 24)
- unit coolers (for the River Bend configuration)

The facility is heavily instrumented to measure gas and surface temperatures, gas velocities, gas concentrations, heat fluxes, pressure, and five video cameras are used for a visual record.

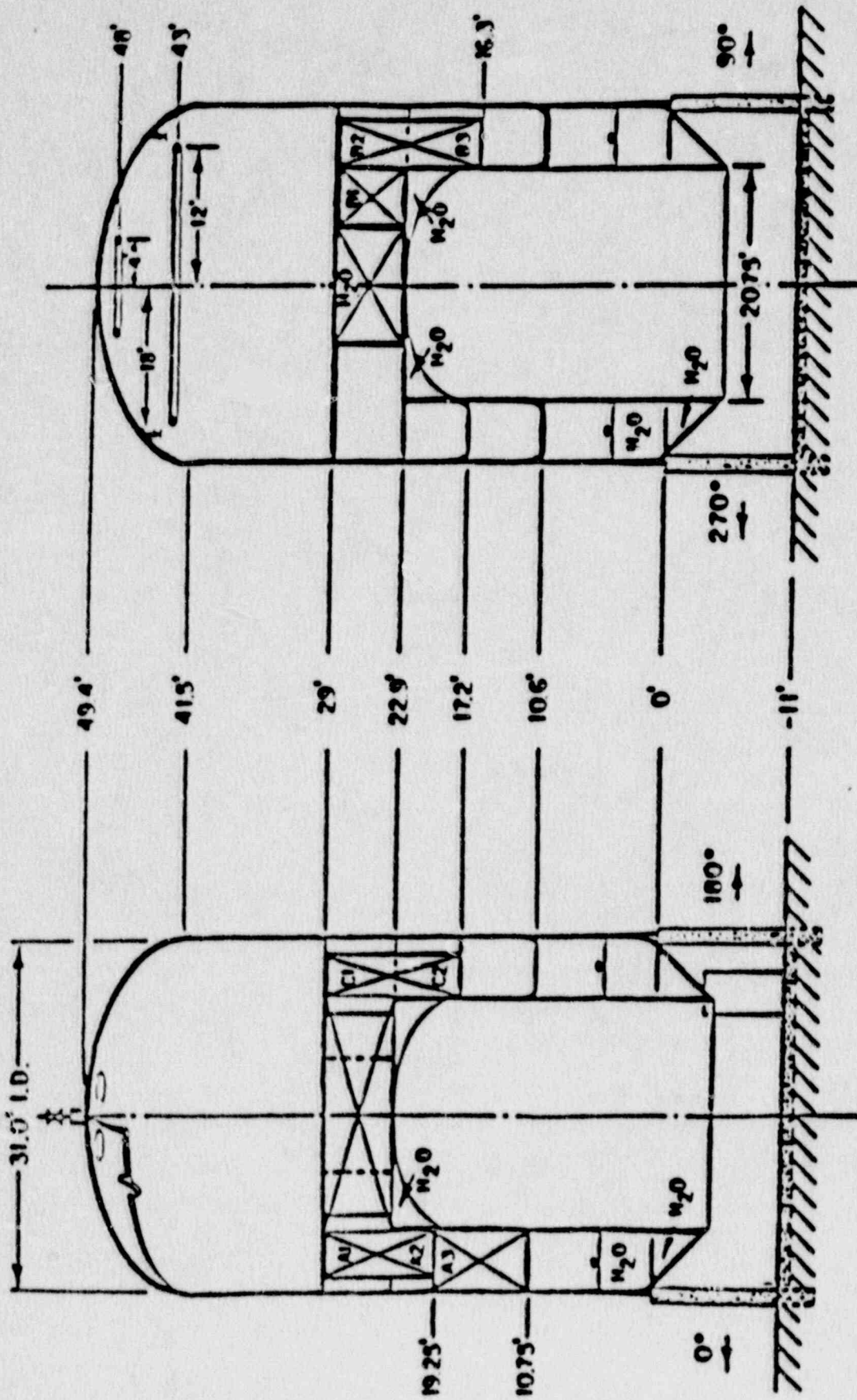


Figure 3.1 Elevation views of the quarter-scale test facility
 Source: HCN-098-P

CONTAINMENT BUILDING
1st FLOOR

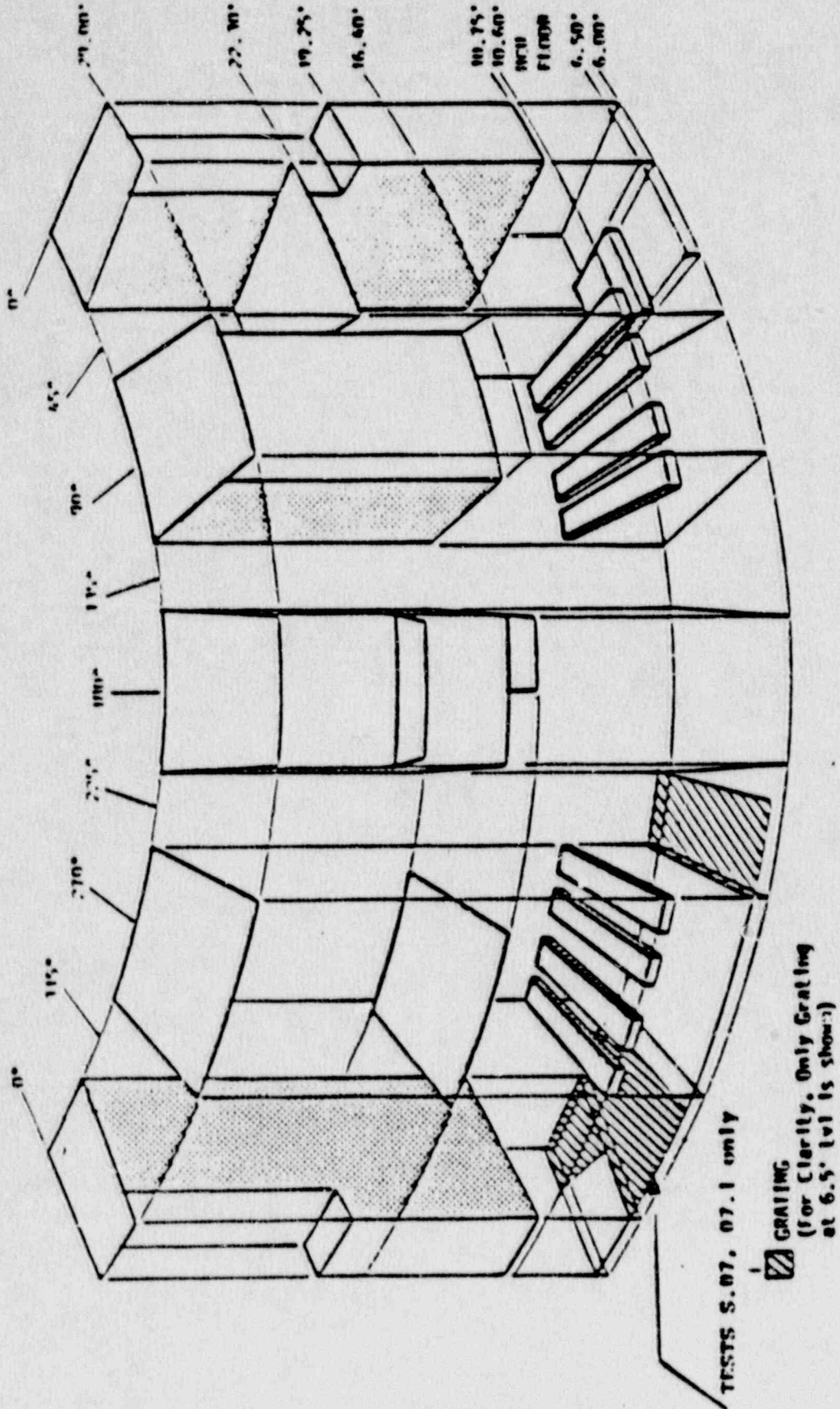


Figure 3.2 Facility configuration tests S.01 - S.11
Source: HGN-098-P

3.2.1 Scaling Methodology

The theoretical basis for modeling hydrogen flames in the test facility is based on Froude scaling. The modeling assumes fully developed, buoyancy dominated, turbulent flows are achieved to preserve the equivalent value for the Froude number in the model and in the full-scale plant. The technique of Froude scaling is supported by numerous experimental demonstrations in the field of fire research.

Using this type of modelling full scale was scaled to quarter scale, a 4-to-1 linear scaling resulting in:

- 32-to-1 reduction in mass and volume flow rates
- 64-to-1 reduction in total hydrogen released
- 2-to-1 reduction in the time scale
- 2-to-1 reduction in gas velocities
- 1-to-1 relationship for gas temperatures and gas concentrations

Flame heights and global flow patterns also were determined by Froude scaling. Generally, Froude scaling was used to reasonably and practically design the QSTF (e.g., spray flow and droplet sizes, heat sink thermal characteristics, blockages). However, the following discrepancies were noted:

- (1) The quarter-scale tests revealed that the insulation used in the facility became wet and its thermal properties departed from dry insulation.
- (2) The QSTF had only 30% of the mass as prescribed by Froude scaling.
- (3) The scaling method did not rigorously simulate convective and radiative heat losses to structural heat sinks.

To assess the impact of these discrepancies, the HCOG provided a comprehensive analysis as documented by HCOG letter HGN-085, dated May 5, 1986. The HCOG determined that compensating effects existed in the treatment of heat sinks. Thus, the data obtained from the QSTF does provide a reasonably accurate description of the full-scale thermal environment when extrapolated by Froude scaling.

To assist the staff in the review of this complex matter, SNL studied the subject analysis and submitted various comments. On October 7, 1987, a meeting between the HCOG, SNL, and the staff took place to resolve the SNL's comments. Subsequently, HCOG documented its responses in a letter (HGN-128) dated November 6, 1987. On the basis of the additional information, SNL concluded and the staff concurs that the application of quarter-scale experimental data directly to full-scale equipment survivability can be done conservatively in spite of the above discrepancies. SNL's assessment is documented in correspondence dated December 23, 1987.

3.2.2 Quarter-Scale Testing Approach

Tests were performed in the QSTF for the four different plants with a Mark III containment; i.e., the QSTF was customized to reasonably represent the plant-specific characteristics of each Mark III configuration. During the tests,

hydrogen was released through spargers used to simulate the automatic depressurization system (ADS), and a stuck-open relief valve (SORV) or through the simulated LOCA vents. Two hydrogen release profiles were used in the facility, a low-reflood case (150 gpm) and a high-reflood case (5000 gpm). A discussion of the development of these profiles is contained in Section 5.

However, before the plant unique or production tests were conducted, a series of scoping tests were performed to assess data repeatability and the significance of various parameters. Results from these tests formed the basis for developing the final test matrix that were used in the production test program. Also, in the early development of postulated degraded core events, spray availability was uncertain, therefore tests were performed with and without sprays activated. In the production tests, each plant had its own specific array of tests, focusing on the SORV locations, the combination of ADS spargers and LOCA vent releases, and the effect of sprays/coolers. The data obtained from these production tests formed a basis for determining the full-scale thermal environment and became a central element of each licensee's final analysis. This information was used as input to analytical evaluations of equipment thermal response for assessing survivability of critical equipment. Further discussions on the use of this data are contained in Section 7.

3.2.3 Quarter-Scale Test Results

The scoping test and partial production test results are summarized and presented in the HCOG's correspondence (HGN-098-P, and HGN-121-P). These results demonstrated that the distributed glow plug igniter system can provide an effective means for limiting accumulation of hydrogen in plants with Mark III containments. Hydrogen concentrations throughout the facility were maintained near or below 5 volume percent (dry basis) for all tests and steam concentrations were determined to be about 10-15 volume percent for selected tests. Although low hydrogen concentrations were maintained, different types of combustion behavior were observed during the tests, depending on the synergistic conditions. The various observed combustion modes are described below.

Diffusion Flames

When hydrogen was released into selected spargers, hydrogen combustion would initiate as a mild deflagration or lightoff (pressure rise about 1 psi) in the wetwell region between the hydraulic control unit (HCU) floor and the suppression pool surface and would persist in the form of standing diffusion flames anchored to the pool surface. This was the dominant mode of combustion and occurred for bulk oxygen concentrations of 8 volume percent (dry) and hydrogen injection rates greater than 0.15 lb/sec. It should be noted that the hydrogen flow rates are full-scale equivalent values (i.e., a 32:1 increase). In this regime of steady flames, combustion was essentially complete. For an injection rate of 1 lb/sec, a flame height of about 8 feet (full scale) was reached.

As the hydrogen injection rate was decreased to about 0.15 lb/sec, combustion became less complete and the flames less stable. As the rate was further decreased, diffusion flames on the pool surface could not be maintained. This point is known as the flame extinguishment limit. Moreover, it was observed that this limit was strongly influenced by background gas concentrations.

August 20, 1990

Docket No. 50-461

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Dear Mr. Spangenberg:

SUBJECT: CLINTON POWER STATION HYDROGEN CONTROL FINAL ANALYSIS REQUIRED BY
10 CFR 50.44 (TAC NO. 62988)

The NRC staff has completed its review of the Mark III Containment Hydrogen Control Owners Group (HCOG) Topical Report HGN-112-NP. Enclosed is a copy of the staff's Safety Evaluation Report (SER) transmitted to HCOG by letter dated August 6, 1990.

By letter dated August 24, 1987, you committed to provide, within 6 months of SER issuance, the final analysis of the Clinton Power Station combustible gas control system required by 10 CFR 50.44. Accordingly, you are requested to provide the final analysis by March 1, 1991, addressing each of the key elements identified in the staff's SER, Section 8.0.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

John B. Hickman, Project Manager
Project Directorate III-3
Division of Reactor Projects III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

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August 6, 1990

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Dear Mr. Langley:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT TITLED,
"GENERIC HYDROGEN CONTROL INFORMATION FOR BWR-6 MARK III
CONTAINMENTS", HGN-112-NP

We have completed our review of the subject topical report submitted by your letter dated February 23, 1987.

We find the report acceptable for referencing in licensee analyses of hydrogen control systems for BWR Mark III containments under the limitations delineated in the report and its references and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

Furthermore, each licensee should provide a plant-specific analysis and an assessment of the need for an independent power supply for the hydrogen ignition system. The plant-specific analysis may use test data described in the topical report to confirm that the equipment necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the hydrogen in all credible severe accident scenarios.

Recent risk studies reported in NUREG-1150 have shown that the overall core melt frequency for one Mark III plant (the Grand Gulf Nuclear Station) is very low, i.e., $1E-6$ /year. However, a potential vulnerability for Mark III plants involves station blackout (SBO), during which the igniters would be inoperable; and this condition appears to dominate the residual risk from severe accident in the Mark III plants. Under SBO conditions, a detonable mixture of hydrogen could develop which could be ignited upon restoration of power resulting in loss of containment integrity. On the basis of a separate evaluation of this possibility in the context of the NRC staff Containment Performance Improvement (CPI) program, the staff has recommended that the vulnerability to interruption of power to the hydrogen igniters be evaluated further on a plant-specific basis as part of the Individual Plant Examinations (IPEs) of the Mark III plants. The staff has requested that the licensees consider this issue as part of the IPE in Generic Letter 88-20, Supplement 3.

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July 27, 1990

Mr. J. R. Langley

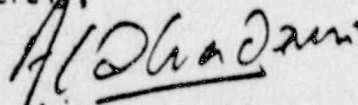
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We do not intend to repeat our review of the matters described in the report and found acceptable when the report is referenced in licensee requests for approval of final analyses of the hydrogen control system, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report and its references.

In accordance with procedures established in NUREG-0390, we request that HCOG submit to the NRC accepted versions of this report within three months of receipt of this letter. The accepted versions should incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions should also incorporate as appendices those references used as a basis for the staff's evaluation. The accepted version should include an -A (designating accepted) following the report identification number. Your submittal should include an application for withholding the proprietary information accompanied by an affidavit meeting the requirements of 10 CFR 2.790(b). This final report submittal should also include a non-proprietary version of the proprietary reports referenced and incorporated in the approved topical report and intended to be employed as a part of a licensee application.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, HCOG and/or the licensees referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,



Ashok C. Thadani, Director
Division of Systems Technology
Office of Nuclear Reactor Regulation

THE GENERIC
SAFETY EVALUATION REPORT RELATING TO
THE MARK III CONTAINMENT HYDROGEN CONTROL

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1 INTRODUCTION AND BACKGROUND

Following a loss-of-coolant accident in a light-water reactor (LWR) plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in both the reactor core and the containment sump; (3) the corrosion of certain materials within the containment by sprays; and (4) any synergistic chemical, thermal, and radiolytic effects of post-accident environmental conditions on protective coatings and electric cable insulation.

To provide protection against this possible hydrogen as a result of an accident, Title 10 of the Code of Federal Regulations (10 CFR) Section 50.44, "Standards for combustible gas control system in light-water-cooled power reactors," and GDC 41, "Containment atmosphere cleanup," Appendix A to 10 CFR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained. Conventional hydrogen control systems (e.g., hydrogen recombiners) historically have been installed to provide the capability to control the relatively low rate of hydrogen accumulation (or oxygen accumulation in inerted containments) resulting from radiolytic decomposition of water, corrosion of metals inside containment, and environmental effects on coatings and insulation. However, the net free volume inside containment (or inerting of the containment volume) is used to control the rapid hydrogen production resulting from a metal-water reaction of the fuel cladding. That is, the containment volume is large enough so that the hydrogen generated early would not reach the lower limit of flammability (or the inerting would prevent combustible mixtures). The rationale for this approach is that the rate of hydrogen release resulting from cladding reaction was assumed to be too rapid (on the order of minutes) following a postulated accident to allow for an active control system. Thus, hydrogen control systems (recombiners) would only be actuated later in the event to control the slow hydrogen/or oxygen release associated with radiolysis and reaction of materials inside containment.

To quantify the metal-water source for design-basis accidents, 10 CFR 50.44, codified in October 1978, requires that the combustible gas control system be capable of handling the hydrogen generated from five times the amount calculated in demonstrating compliance with 10 CFR 50.46 or the amount corresponding to the total reaction of the cladding to a depth of 0.23 mils, whichever amount was greater. Typically, this would translate to a 1 to 5% metal-water reaction of the active clad.

However, the accident at Three Mile Island Unit 2 (TM1-2) on March 28, 1979, resulted in a metal-water reaction that involved approximately 45% of the fuel cladding (i.e., about 990 lbs), which resulted in hydrogen generation well in excess of the amounts specified in 10 CFR 50.44. The combustion of this hydrogen produced a significant pressure spike inside containment. As a result, it became apparent that additional design measures were needed to handle larger hydrogen releases, particularly for smaller volume containments and those with

lower design pressures. The Nuclear Regulatory Commission (NRC) determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be considered in plant design. An advance notice of the rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980. In addition, a new unresolved safety issue was instituted (A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment") to evaluate this new concern.

To formalize its requirements for additional hydrogen control measures to deal with degraded core accidents affecting pressurized-water reactors (PWRs) with ice condenser containments and BWRs with Mark III containments, the NRC published an amendment to the hydrogen rule (10 CFR 50.44) on January 25, 1985 (50 FR 3498). The amended rule required that a hydrogen control system be provided and that the system be capable of accommodating, without loss of containment structural integrity, the amount of hydrogen generated from a metal-water reaction of up to 75% of the active fuel cladding. In addition, systems and components necessary to establish and maintain safe shutdown must be capable of performing their function regardless of hydrogen burning.

Pursuant to the provisions of the rule, each utility with a Mark III containment has installed a hydrogen ignition system (HIS) and submitted a preliminary analysis and a schedule for meeting the full requirements of the rule. The affected plants with Mark III containments are Grand Gulf Nuclear Station, River Bend Station, Perry Nuclear Power Plant, and Clinton Power Station. The staff's interim evaluations of these initial plant responses are documented in supplements to the safety evaluation reports for each of these plants (NUREG-0831, NUREG-0989, NUREG-0807, and NUREG-0853, respectively).

These responses were aided by the efforts of the Mark III Containment Hydrogen Control Owners Group (HCOG), formed in May 1981 by the utilities with Mark III containments to collectively perform testing and analyses to demonstrate the effectiveness and reliability of the hydrogen ignition systems.

In addition, each licensee with a Mark III containment is required to provide a final analysis [10 CFR 50.44(c)(3)(vii)(B)] to confirm the conclusions of the preliminary analysis and/or, if necessary, to institute modifications to ensure compliance with the rule. The scope of this analysis is specified in 10 CFR 50.44(c)(3)(vi)(B). The generic findings from the HCOG's program will be utilized for this final analysis, supplemented by plant specific design considerations as addressed in the licensee's IPE program.

The following staff evaluation focuses on the assessment of the completed generic testing and analyses performed by the HCOG in support of the plant unique analysis. The HCOG activities have been summarized in a topical report transmitted by letter dated February 23, 1987, correspondence identification HGN-112-NP, "Generic Hydrogen Control Information for BWR-6 Mark III Containments." The topical report is a summary document of all of the individual generic submittals that have been sent to the staff by the HCOG. It should be noted that HCOG correspondence identification designators with a "P" suffix (HGN-XXX-P) are proprietary to HCOG. Whereas, those without a suffix or with an "NP" suffix are nonproprietary.

As part of the review of the HCOG program, the staff obtained technical assistance from the Sandia National Laboratory (SNL), the principal contractor for the NRC research program on hydrogen control and combustion phenomena. SNL provided the NRC with an independent assessment of technical issues contained in selected HCOG submittals pertaining to hydrogen behavior.

The staff evaluation of the generic considerations of the hydrogen control system for the Mark III containment can best be understood if categorized as follows:

- general description of the hydrogen ignition system
- combustion and igniter testing
- containment structural capacity
- degraded core events and hydrogen generation
- containment response-analytical modeling
- survivability of essential equipment
- overall conclusions

Therefore, the following discussion will follow this general outline.

2 GENERAL DESCRIPTION OF THE HYDROGEN IGNITION SYSTEM

The regulation, 10 CFR 50.44(c)(3)(iv)(A), states:

Each licensee with a boiling light-water nuclear power reactor with a Mark III type of containment..., shall provide its nuclear power reactor with a hydrogen control system justified by a suitable program 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 reaction involving 75% of the fuel cladding surrounding the active fuel region [excluding the cladding surrounding the plenum volume].

The concept employed by the licensees with a Mark III containment, and similarly the ice condenser licensees, is to intentionally ignite hydrogen generated inside containment. This method precludes buildup of relatively high concentrations of hydrogen during degraded core accident scenarios.

To accomplish this early ignition a hydrogen ignition system (HIS) is installed in each of the four plants with Mark III containments. The HIS is a system which consists of approximately 100 igniter assemblies distributed throughout the drywell and containment regions.

The main element of the igniter assembly is the Model 7G thermal igniter glow plug (commonly used in diesel engines) manufactured by the General Motors AC Division. Each Mark III containment has an identical igniter assembly design. Each igniter is powered directly from a 120/12-V stepdown transformer and designed to provide a minimum surface temperature of 1700°F. The igniter assembly (see Figure 2.1) consists of a 1/8-inch-thick stainless steel box that contains the transformer and all electrical connections and is manufactured by the Power Systems Division of Morrison Knudsen. Igniter assemblies are Class 1E, seismic Category I, and meet the requirements of the Institute of Electrical and Electronics Engineers (IEEE) Std 323-1974 and NUREG-0588 for environmental qualification.

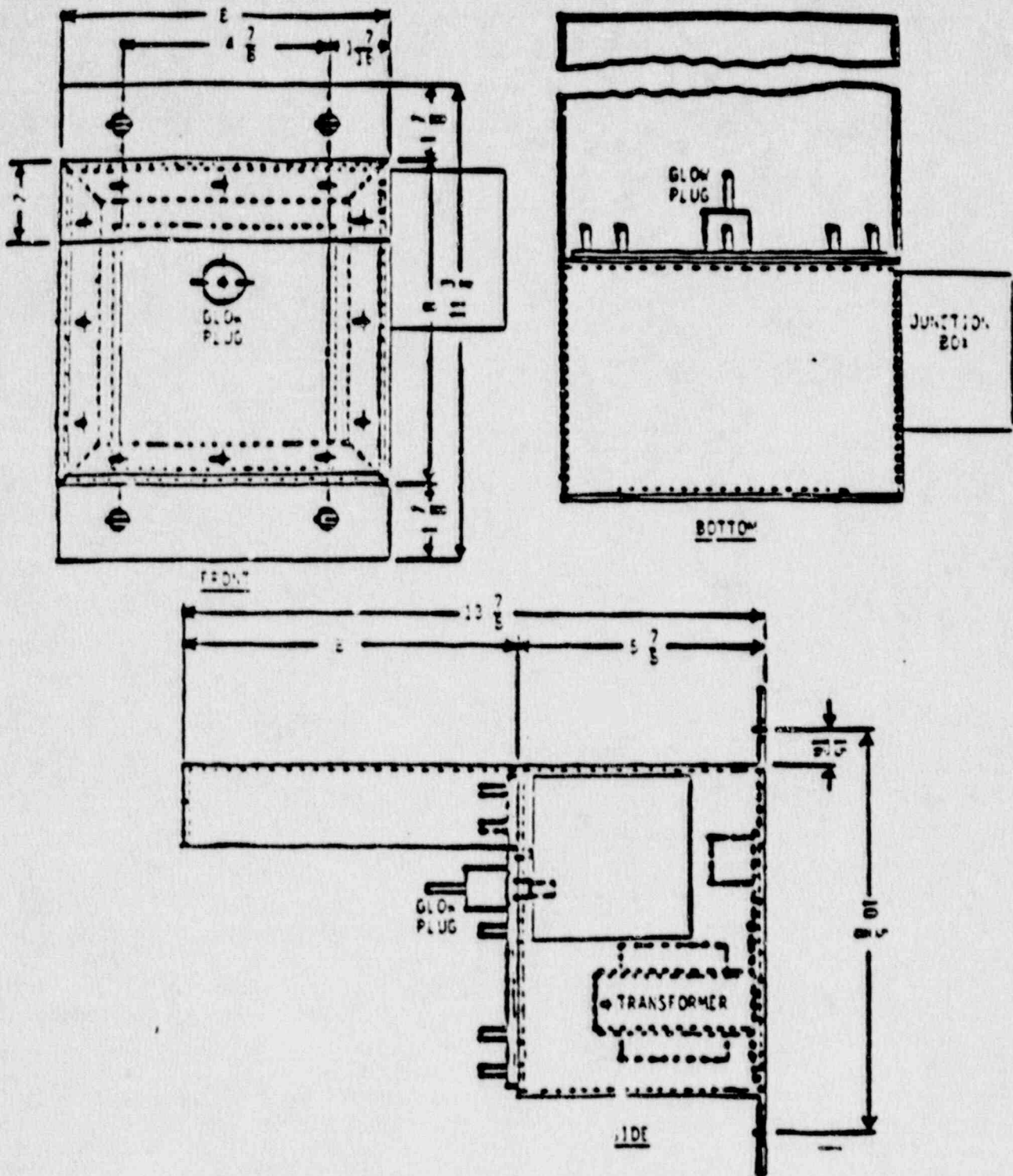


Figure 2.1 General hydrogen igniter assembly

The igniter assemblies are divided into two groups; each group being powered from a separate Class 1E division power supply. The intent is to have at least two igniters located in each enclosed volume or area within the containment that could be subject to possible hydrogen pocketing, and each igniter be powered from a separate power division. In open areas within the containment and drywell regions, igniter assemblies at the same elevation are designed to have alternating power division sources. Igniters are separated by about 30 feet when both engineered safety feature (ESF) power divisions are operable or by about 60 feet when one power division is inoperable. Igniter placement is designed to be more widely spaced in the large open regions, such as above the refueling floor, and in the lower regions of the drywell that are subject to flooding. Requirements of placement as well as other parameters of the system are contained in the technical specifications for the hydrogen ignition system, as proposed by the HCOG, and addressed in Appendix A of this report.

Each igniter power division has a corresponding onsite emergency diesel generator. Incorporation of the emergency diesels into the design addresses the question of igniter power for many sequences, but not for station blackout conditions. Under blackout conditions, the HIS would not be operable. On the basis of a separate evaluation of this possibility in the context of the NRC Containment Performance Improvement (CPI) program, the staff has recommended that the vulnerability to interruption of power to the hydrogen igniters be evaluated further on a plant-specific basis as part of the Individual Plant Examinations of the Mark III plants. (See Generic Letter 88-20, Supplement 3)

The HIS is designed for manual actuation from the main control room. Actuation by the operator is required by plant emergency procedures when the reactor pressure vessel (RPV) water level reaches the top of active fuel (TAF). The proposed combustible gas control emergency procedures for plants with Mark III containments are further discussed in Appendix B.

By letter dated March 5, 1986 (HGN-073), the HCOG provided the following justification to support manual actuation. Actuation is linked to indication of the RPV water level, which is a key safety parameter and is closely monitored by the operators. Also, HIS actuation requires only the positioning of two hand-switches. Furthermore, operators should not hesitate to energize the system during accident scenarios in which the hydrogen threat is uncertain or marginal because there would be no adverse effect on the plant as a result of unnecessary igniter actuation. Time available to actuate the HIS is the other significant parameter. Based on the hydrogen events considered, the HCOG estimated this time to be approximately no less than 25 minutes; that is, after the water level reaches TAF to the lower hydrogen flammability limit reached in the wetwell volume. The HCOG also noted that hydrogen would migrate to the upper portions of the containment before the wetwell reaches hydrogen combustion conditions. This effect was seen in the quarter-scale tests. Therefore, the time interval is expected to be somewhat greater than 25 minutes.

Because (1) manual actuation is a simple task, (2) the operator has sufficient time to perform the task, (3) and there are no negative effects if the system is inadvertently or unnecessarily actuated, the staff finds manual operator actuation acceptable.

HIS Design Assessment

The staff finds these hydrogen igniter systems currently installed in the plants with a Mark III containment to be acceptable, with the caveat that the vulnerability to interruption of power to the hydrogen igniters be further evaluated on a plant-specific basis as part of IPEs of the Mark III plants. (See Supplement No. 3 to Generic Letter 88-20)

3 COMBUSTION/IGNITER TESTING

Numerous research programs have been conducted since 1980 to better understand hydrogen combustion behavior and the performance of ignition devices. Because these programs were varied in scope, Sandia National Laboratory (SNL) summarized the findings of recent hydrogen combustion test programs in NUREG/CR-5079. This report also provides additional background information and insights related to hydrogen behavior.

The specific test programs considered necessary by HCOG to support the unique plant characteristics of a Mark III containment are discussed below.

3.1 Small-Scale Tests

Small-scale hydrogen combustion tests were performed at Whiteshell Laboratories and documented by the HCOG in a letter dated June 7, 1984 (HGN-017-NP). The program was intended to investigate ignition and combustion behavior of mixtures predominantly composed of hydrogen and steam, i.e., with limited available oxygen. This condition may exist for a postulated drywell break event in which air is initially swept from the drywell and then later reintroduced into the steam-hydrogen environment. These tests confirmed that such hydrogen-air-steam mixtures can be successfully ignited as long as the oxygen concentration exceeds approximately 5-6 volume percent.

A 1/20th-scale Mark III hydrogen combustion program was conducted by Acurex Corporation and documented by the HCOG in a letter dated February 9, 1984 (HGN-014-NP). The objective of this program was to provide a visual record of hydrogen combustion behavior in a 360-degree model of a Mark III containment. Modelling included the suppression pool and major blockages in the annular region between the drywell and outer containment walls. Hydrogen was admitted through simulated quenchers and/or vents into the suppression pool and ignited by prototypical ignition sources.

The most important result obtained from the 1/20-scale test was the confirmation of continuous hydrogen burning in the form of steady diffusion flames above the suppression pool. The significance of this mode of hydrogen burning is the observed severe thermal loads that occur near the diffusion flames and could threaten the integrity of the containment and equipment. Diffusion flames were observed when hydrogen flow rates of 0.4 lb/sec (full-scale equivalent) or

greater were used. Combustion was initiated by the igniters and rapidly propagated to the pool surface and formed steady diffusion flames that were anchored at the surface of the pool and located above the submerged spargers that released the hydrogen. Hydrogen burning was observed to intensify as the hydrogen injection flow rate was increased, as evidenced by taller flames and higher temperatures.

As part of the 1/20-scale program to determine the sensitivity of scaling, a 1/5-scale single-sparger mockup was constructed. A significant result of these tests was that approximately one-half the flame height of what would have been predicted based on results of 1/20-scale tests was observed. On the basis of these tests, it became necessary for the HCOG to pursue a larger-scale test program to obtain thermal environmental data more representative of a Mark III containment. Subsequently, the HCOG undertook an extensive program to better define the conditions that could exist during a degraded core accident. A major element of this effort was the quarter-scale Mark III containment combustion test program.

3.2 Quarter-Scale Test Facility

The quarter-scale test program became the major element of the HCOG's hydrogen research program. The primary objective of this program was the investigation and characterization of the environment that could result from diffusive burning on the suppression pool in a Mark III containment. Ultimately, the information gathered from the quarter-scale test facility (QSTF) would be used in determining the survivability of select equipment. Test facility description and the combustion test results may be found in the HCOG's letters HGN-098-P, July 18, 1986; HGN-115-NP, February 10, 1987; and HGN-121-P, July 22, 1987.

The test facility is a quarter-linear-scale model of a Mark III containment, designed and constructed by Factory Mutual Research Corporation (FMRC), and located in West Gloucester, Rhode Island. The test enclosure is designed to operate at pressures up to 40 psig and consists of an outside tank, 31.5 feet in diameter, 49.4 feet high, containing a smaller tank, about 21 feet in diameter by 23 feet high. The space between the two tanks is the test volume; which contains floors and other large blockages simulating the obstructions that exist in the actual containments. Because of the unique features of the four plants studied, modular construction of the annular floors is used to modify the vessel interior when needed. At the bottom of the two tanks, the suppression pool is simulated. Several views of the facility are shown in Figures 3.1 and 3.2.

Facility design features include:

- containment sprays
- simulated loss-of-coolant-accident (LOCA) vents (top row only, numbering 48)
- simulated spargers (uniformly spaced every 15 degree azimuthally and totalling 24)
- unit coolers (for the River Bend configuration)

The facility is heavily instrumented to measure gas and surface temperatures, gas velocities, gas concentrations, heat fluxes, pressure, and five video cameras are used for a visual record.

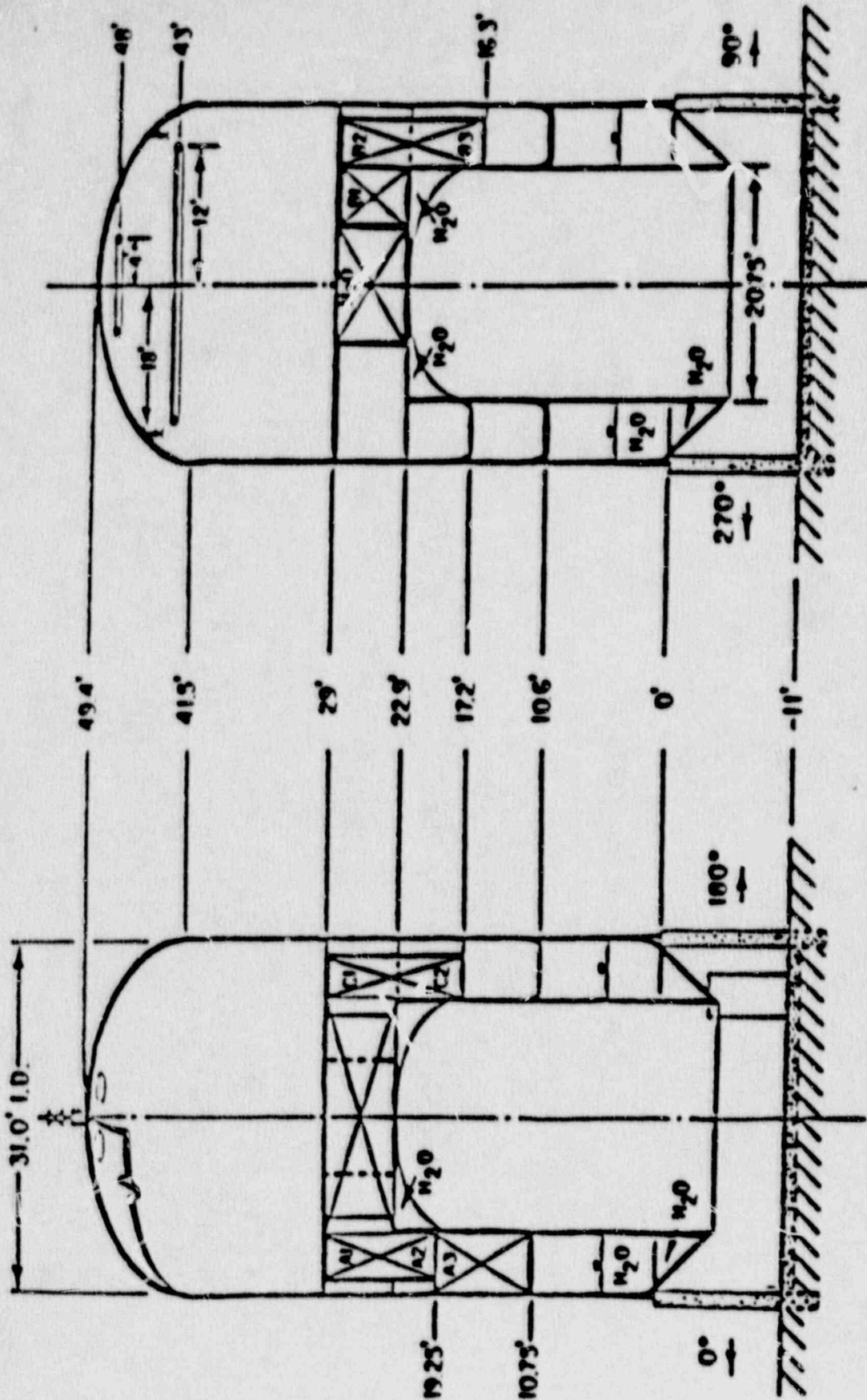


Figure 3.1 Elevation views of 1/4 quarter-scale test facility
 Source: HEN-098-P

CONTAINMENT BUILDING
1 TO VIEW

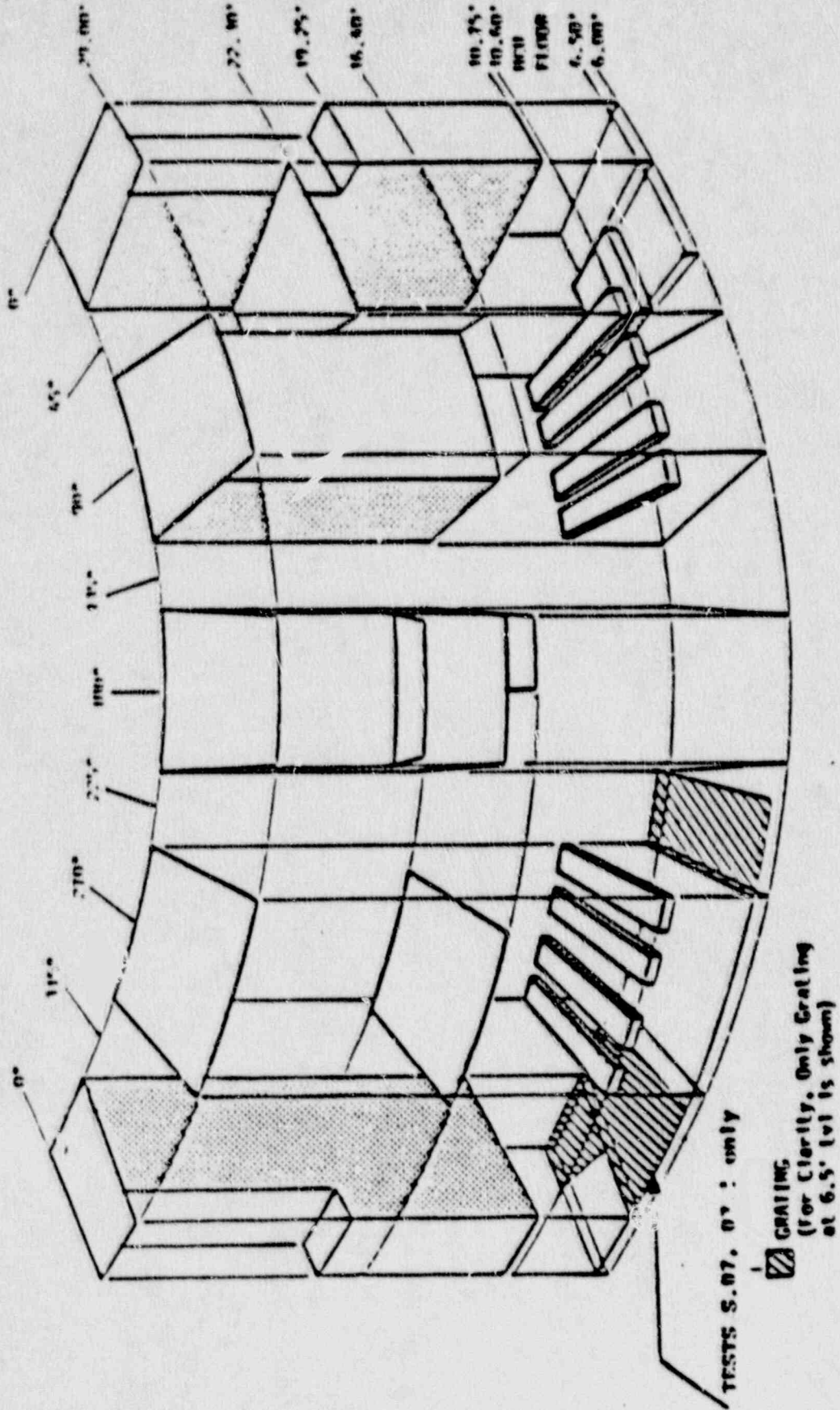


Figure 3.2 Facility configuration tests S.01 - S.11
Source: HGN-098-P

3.2.1 Scaling Methodology

The theoretical basis for modeling hydrogen flames in the test facility is based on Froude scaling. The modeling assumes fully developed, buoyancy dominated, turbulent flows are achieved to preserve the equivalent value for the Froude number in the model and in the full-scale plant. The technique of Froude scaling is supported by numerous experimental demonstrations in the field of fire research.

Using this type of modelling full scale was scaled to quarter scale, a 4-to-1 linear scaling resulting in:

- 32-to-1 reduction in mass and volume flow rates
- 64-to-1 reduction in total hydrogen released
- 2-to-1 reduction in the time scale
- 2-to-1 reduction in gas velocities
- 1-to-1 relationship for gas temperatures and gas concentrations

Flame heights and global flow patterns also were determined by Froude scaling. Generally, Froude scaling was used to reasonably and practically design the QSTF (e.g., spray flow and droplet sizes, heat sink thermal characteristics, blockages). However, the following discrepancies were noted:

- (1) The quarter-scale tests revealed that the insulation used in the facility became wet and its thermal properties departed from dry insulation.
- (2) The QSTF had only 30% of the mass as prescribed by Froude scaling.
- (3) The scaling method did not rigorously simulate convective and radiative heat losses to structural heat sinks.

To assess the impact of these discrepancies, the HCOG provided a comprehensive analysis as documented by HCOG letter HGN-085, dated May 5, 1986. The HCOG determined that compensating effects existed in the treatment of heat sinks. Thus, the data obtained from the QSTF does provide a reasonably accurate description of the full-scale thermal environment when extrapolated by Froude scaling.

To assist the staff in the review of this complex matter, SNL studied the subject analysis and submitted various comments. On October 7, 1987, a meeting between the HCOG, SNL, and the staff took place to resolve the SNL's comments. Subsequently, HCOG documented its responses in a letter (HGN-128) dated November 6, 1987. On the basis of the additional information, SNL concluded and the staff concurs that the application of quarter-scale experimental data directly to full-scale equipment survivability can be done conservatively in spite of the above discrepancies. SNL's assessment is documented in correspondence dated December 23, 1987.

3.2.2 Quarter-Scale Testing Approach

Tests were performed in the QSTF for the four different plants with a Mark III containment; i.e., the QSTF was customized to reasonably represent the plant-specific characteristics of each Mark III configuration. During the tests,

hydrogen was released through spargers used to simulate the automatic depressurization system (ADS) and a stuck-open relief valve (SORV) or through the simulated LOCA vents. Two hydrogen release profiles were used in the facility, a low-reflood case (150 gpm) and a high-reflood case (5000 gpm). A discussion of the development of these profiles is contained in Section 5.

However, before the plant unique or production tests were conducted, a series of scoping tests were performed to assess data repeatability and the significance of various parameters. Results from these tests formed the basis for developing the final test matrix that were used in the production test program. Also, in the early development of postulated degraded core events, spray availability was uncertain, therefore tests were performed with and without sprays activated. In the production tests, each plant had its own specific array of tests, focusing on the SORV locations, the combination of ADS spargers and LOCA vent releases, and the effect of sprays/coolers. The data obtained from these production tests formed a basis for determining the full-scale thermal environment and became a central element of each licensee's final analysis. This information was used as input to analytical evaluations of equipment thermal response for assessing survivability of critical equipment. Further discussions on the use of this data are contained in Section 7.

3.2.3 Quarter-Scale Test Results

The scoping test and partial production test results are summarized and presented in the HCOG's correspondence (HGN-098-P, and HGN-121-P). These results demonstrated that the distributed glow plug igniter system can provide an effective means for limiting accumulation of hydrogen in plants with Mark III containments. Hydrogen concentrations throughout the facility were maintained near or below 5 volume percent (dry basis) for all tests and steam concentrations were determined to be about 10-15 volume percent for selected tests. Although low hydrogen concentrations were maintained, different types of combustion behavior were observed during the tests, depending on the synergistic conditions. The various observed combustion modes are described below.

Diffusion Flames

When hydrogen was released into selected spargers, hydrogen combustion would initiate as a mild deflagration or lightoff (pressure rise about 1 psi) in the wetwell region between the hydraulic control unit (HCU) floor and the suppression pool surface and would persist in the form of standing diffusion flames anchored to the pool surface. This was the dominant mode of combustion and occurred for bulk oxygen concentrations of 8 volume percent (dry) and hydrogen injection rates greater than 0.15 lb/sec. It should be noted that the hydrogen flow rates are full-scale equivalent values (i.e., a 32:1 increase). In this regime of steady flames, combustion was essentially complete. For an injection rate of 1 lb/sec, a flame height of about 8 feet (full scale) was reached.

As the hydrogen injection rate was decreased to about 0.15 lb/sec, combustion became less complete and the flames less stable. As the rate was further decreased, diffusion flames on the pool surface could not be maintained. This point is known as the flame extinguishment limit. Moreover, it was observed that this limit was strongly influenced by background gas concentrations.

To illustrate the various relationships, the following was extracted from the QSTF scoping test report:

- (1) The flame extinguishment limit ranges from ~ 0.07 to ~ 0.15 lb/sec for ambient hydrogen concentrations below ~ 4.1 volume percent dry (high oxygen conditions).
- (2) The flame extinguishment limit decreases with increasing hydrogen concentration to a minimum between ~ 0.025 and ~ 0.03 lb/sec at a hydrogen concentration of ~ 4.5 volume percent (high oxygen conditions.)
- (3) For comparable ambient hydrogen concentrations, the flame extinguishment limit is slightly higher at low oxygen conditions.

Another effect that accompanied hydrogen burning was the formation of bulk air currents. Horizontal air flow was created above the pool surface allowing diffusion burning to continue by providing a source of oxygen. Another pattern of circulation was the creation of chimneys, which provide for flow to and from the region of burning and exchange flow with the upper containment, that is, hot (upward flow) and cold (downward flow) chimneys.

Localized Combustion

Below the flame extinguishment limit, flames on the pool were not observed. The prevalent burning mode at very low hydrogen release rates has been termed localized combustion. This phenomenon is characterized as weak flames or volume burning through a marginally combustible hydrogen/air/steam mixture. This type of combustion was detected only in regions at or above the HCU floor and concentrated mostly in chimney areas. This was evident by temperature measurements; localized combustion was not observed by video recordings. Localized combustion appeared to be relatively benign (i.e., less than 250°F at instrumented locations). Burning was more widespread and somewhat more intense at low oxygen conditions and was accompanied by slightly higher background hydrogen concentrations (i.e., near 5%).

The QSTF was oriented to investigate the burning phenomena in the area immediately above the suppression pool. As a result, the instrumentation layout above the HCU floor was not sufficient to provide a detailed mapping of the conditions in that region. Consequently, a rigorous investigation of localized combustion was not possible; however, the instrumentation that was present, along with HCOG's analytical effort (see Section 6), provided a reasonable characterization of the phenomena. Localized combustion is discussed further in Section 6 as it relates to the analytical methods used by the HCOG.

Secondary Burning

During the quarter-scale testing, an additional combustion phenomenon was observed late in one of the tests. When the bulk oxygen concentration dropped below 8 volume percent (dry), flames extinguished on the surface of the pool but formed at the HCU floor elevation. This type of burning has been termed secondary burning.

During a June 1986 meeting with the staff, HCOG revealed the presence of secondary burning in one of the Perry production tests. This phenomenon was not observed in previous production tests or in the scoping test phase. Until this particular test, only in a scoping test did the containment oxygen concentration drop below 8%. Oxygen concentrations were generally maintained above 8 percent due to a unique need associated with the video coverage. Each of the five video cameras used in the QSTF required a continuous air purge for the camera lenses to prevent a condensation on the lens. This resulted in a continuous inflow of oxygen in the facility, thus precluding the atmospheric oxygen concentration to fall below 8%. However, the camera air purges were not run continuously in the subject test; subsequently, late in this test the oxygen concentration fell below 8%. Additional information is provided by HCOG submittal HGN-106-P dated September 29, 1986, and also discussed in detail in the quarter-scale combustion test report.

To present its overall assessment of the significance of secondary burning, HCOG began by addressing the limitations of the QSTF. The QSTF has various physical and practical limitations associated with the investigation of secondary burning. The instrumentation in the facility was geared to define the thermal environment produced by diffusion flames anchored to the surface of the suppression pool, which is the dominant combustion mode. Therefore, more instrumentation would be needed to investigate burning above the HCU floor. Since all plants with Mark III containments have different containment volumes, simulation of the expected oxygen depletion profile for each plant would be difficult. Therefore, HCOG evaluated the need to further consider the secondary burning phenomenon. The following identifies the various factors considered and their relationship to the four plants with Mark III containments:

- (1) Secondary burning is expected to occur over a narrow range of oxygen concentrations, approximately from 6 to 8%. Based on the hydrogen generated by a 75% metal-water reaction, a Mark III containment would experience this oxygen concentration interval late in the transient or not at all. Assuming the drywell air is not added to the containment inventory, a metal-water reaction of 55 to 67% would be reached before the oxygen concentration is expected to fall below 8%. This range applies to three of the four plants. Because of the larger containment volume-to-power ratio, Clinton is not expected to fall below 10% oxygen; thus, secondary burning is not anticipated. When the drywell air inventory is included in the containment region, a metal-water reaction of 67% would be reached for Perry before the oxygen concentration is expected to fall below 8%; River Bend and Grand Gulf would have already consumed the equivalent hydrogen required by the rule (i.e., 75% of the fuel cladding surrounding the active fuel region).
- (2) Considerable uncertainty is inherent in predicting the long-term hydrogen profile, especially in the latter phase of the profile (refer to Section 5). For example, an alternate accident sequence such as a drywell break sequence, different hydrogen release rates, the use of the drywell mixing system, or not even reaching a 75% metal-water-reaction value, could reduce or possibly eliminate secondary burning.

(3) Also, not all conditions expected to exist in an actual plant existed in the test when secondary burning was observed. For example, sprays were not activated and the burning on the HCU floor was in a sector where the hydrogen release was most concentrated. This sector was in the 45-degree chimney in which the SORV was located and the steam tunnel structure would reduce the upward cross-sectional flow area. Therefore, it is expected that these factors contributed to the locally high concentration of hydrogen that is required for secondary burning. It should also be noted that the overall shape of the flames occupied a relatively small area, forming a flame zone near the corner of the steam tunnel and drywell wall. On the basis of these differences, the following significant mitigating factors can be inferred to reduce the consequence of secondary burning:

- (a) Increased turbulence inside containment through spray operation or unit cooling could potentially delay or preclude secondary burning. This was evident to some degree in one of the scoping tests during which conditions were similar to those during the Perry test where secondary burning was present. This scoping test had sprays functioning and the oxygen concentration fell to approximately 7.8%. Secondary burning did not occur. Also, sprays/unit coolers would provide cooling to mitigate the consequences of secondary burning if it were to occur.
- (b) Secondary burning appears to be extremely localized. It occurred in the region above the location where three adjacent safety relief valves (SRVs) spargers released hydrogen. Further, secondary burning occupied only a small zone. Because of equipment redundancy and separation, secondary burning is expected to affect only one train of equipment.

On the basis of its findings, the HCOG determined further experimental investigation of secondary burning was not necessary.

The staff's review of the evidence indicated that secondary burning is not expected to present a significant additional threat and, if this combustion mode were to occur, it is expected that the thermal zone of influence would be limited. Therefore, the staff agrees that further detail study of secondary burning is unwarranted. However, since the redundancy of equipment (i.e., spatial separation of equipment performing the same function) is the most important element, the staff requests that the Perry, River Bend, and Grand Gulf licensees (excluding Clinton) confirm that sufficient separation (i.e., at least a 90° azimuthal displacement) exists between the redundant equipment expected to be affected by secondary burning.

4 CONTAINMENT STRUCTURAL CAPACITY

The burning of hydrogen inside containment has the potential to induce pressure excursions in excess of the containment/drywell design values. To determine the pressure capability of the containment structures, required by 10 CFR 50.44 (c)(3)(iv)(B), each licensee provided its plant-specific analysis for staff review. The details of the staff's evaluations regarding the containment and drywell ultimate capacities are documented in each of the plant's respective SER supplements. Rather than repeating these evaluations, a brief description of the Mark III containment will be provided.

In the Mark III containment design, the containment completely surrounds the drywell. At the bottom of the containment, a 360-degree annular suppression pool is located between the containment wall and drywell wall. Below the pool surface, horizontal vents are constructed in the drywell wall. The principal difference between the four plants is in the characteristics of the containment shell, as illustrated in Table 4.1. For Grand Gulf and Clinton, the primary containment is a steel-lined, reinforced concrete structure consisting of a vertical cylinder and a hemispherical dome top. For River Bend and Perry, the primary containment is a free-standing steel vessel consisting of a vertical cylinder and a torus-spherical dome surrounded by a concrete shield building. The internal containment design pressure of 15 psig is the same for each plant. The ultimate pressure capacity was determined to be about three times design (i.e., approximately 50-60 psig) for each plant. Since the drywell structure is designed to greater pressure values than the containment vessel, the drywell ultimate capacities also are greater and are not limiting in the forward or reverse direction. The containment pressure capacity, taking into consideration limiting containment penetrations, is used as the limiting parameter when evaluating the consequences of hydrogen deflagrations inside containment. Figure 4.1 is an illustration of a Mark III containment configuration.

5 DEGRADED CORE EVENTS AND HYDROGEN GENERATION

5.1 Introduction

To determine the consequences of hydrogen burning, the hydrogen generation release must be addressed to establish a representative hydrogen-generation event (HGE) and define representative hydrogen release profiles.

The regulation, 10 CFR 50.44(c)(3)(vi)(B), specifically requires that the following be considered in the analysis:

- (1) large amounts of hydrogen generated after the start of an accident (hydrogen resulting from the metal-water reaction of up to and including 75% of the fuel cladding surrounding the active fuel region, excluding the cladding surrounding the plenum volume);
- (2) the period of recovery from the degraded condition;
- (3) accident scenarios that are accepted by the NRC staff and that are 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 HCOG analyzed two degraded core accident sequences (HCOG transmittals HGN-003, -006, -018-P, -031, -052, -055, -072, -104-P, -112-NP, -129-P and -132). The base-case scenario begins with a loss of offsite power, followed by reactor scram, isolation of both the containment and MSIVs, and power conversion system unavailability. One diesel generator fails to start and the relief valves cycle on high reactor pressure as a result of MSIV isolation. Relief valve cycling results in one stuck-open relief valve (SORV). The second scenario models a small break in the drywell by using the same total hydrogen and steam release histories as the previous case but the predicted hydrogen and steam release is mechanistically split between the drywell and the containment.

Table 4.1 Comparison of BWR Mark III Containment Characteristics

Characteristic	Grand Gulf	Perry	River Bend	Clinton
Rated thermal output, Mwt	3,833	3,579	2,894	2,894
Number of fuel bundles	800	748	624	624
Drywell structure:				
Design pressure, psig	30	30	25	30
External design pressure, psid	21	21	20	17
Air volume, ft ³	270,000	277,685	236,196	246,500
Suppression pool volume (includes vents), ft ³	1.3E4	1.12E4	1.3E4	1.1E4
Suppression pool surface area, ft ²	553	482	522	455
Holdup volume, ft ³	50,000	40,564	20,353	33,804
Holdup surface area, ft ²	3,145	2,617	2,564	2,490
Containment vessel:				
Design pressure, psig	15	15	15	15
Ultimate pressure capacity, psig	56	50	53	63
External design pressure, psid	3	0.8	0.6	3
Total air volume, ft ³	1.4E6	1.141E6	1.192E6	1.551E6
Air volume below hydraulic control unit floor, ft ³	151,644	181,626	153,792	173,000
Suppression pool volume, ft ³	1.24E5	1.06E5	1.28E5	1.35E5
Suppression pool surface area, ft ²	6,667	5,900	6,408	7,175
Upper pool makeup volume, ft ³	36,380	32,830	0	14,655
Containment spray flow rate (1 train), gpm	5,650	5,250	0	3,800
Number of loss-of-coolant-accident vents	135	120	129	102

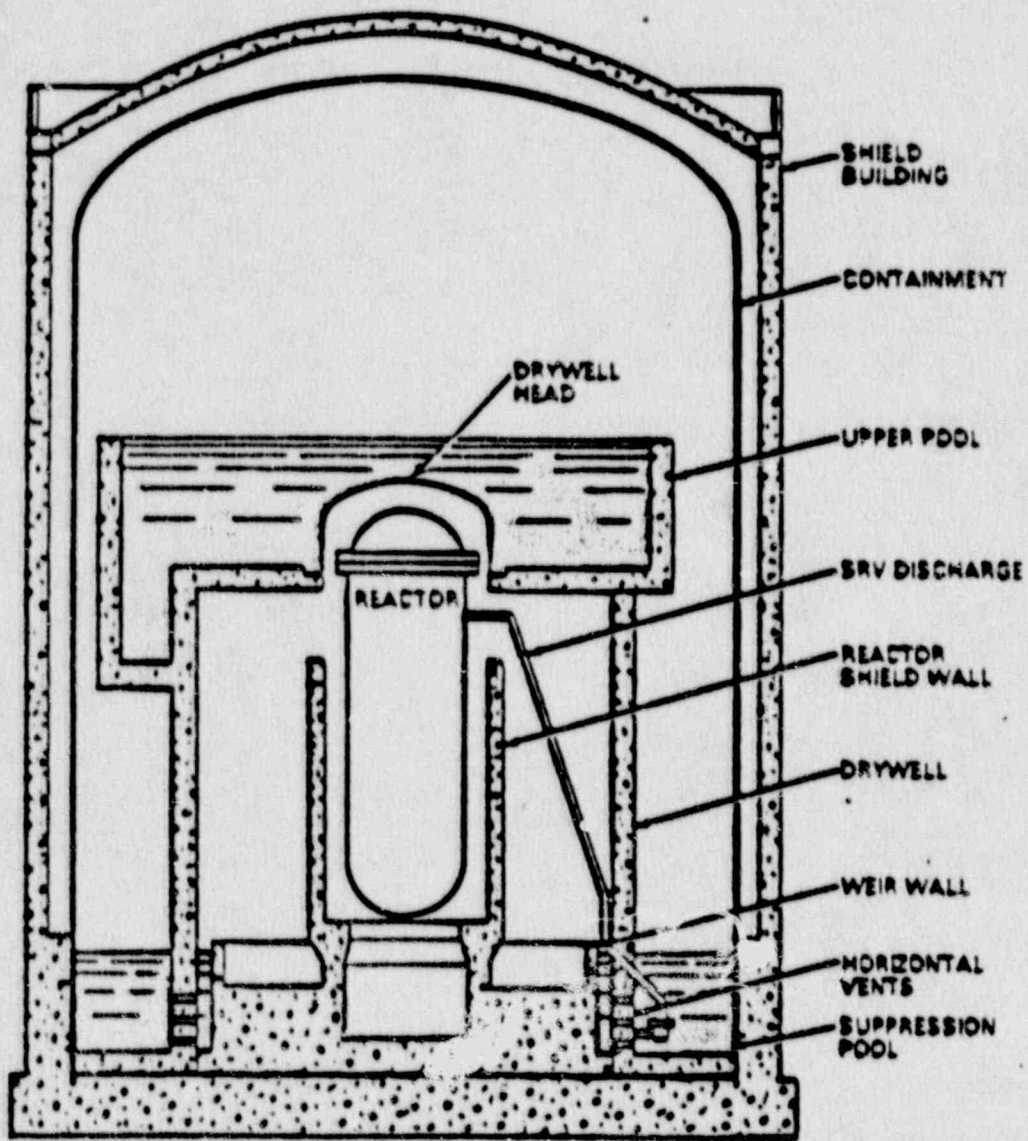


Figure 4.1 - Typical Mark III Containment Configuration

The transients resulting in an SORV were selected (1) to ensure a rapid loss of inventory and (2) to account for and create a limiting local thermal containment environment for analysis and testing. Small-break loss-of-coolant accidents (SBLOCAs) were selected as an alternate sequence to address the potential and consequences for hydrogen combustion in the drywell. Otherwise the SBLOCA sequence is identical to the SORV sequence.

For the base-case scenario, all ac-powered reactor makeup systems are assumed to initially fail. According to emergency procedures, the operator will depressurize the reactor when water level decreases to the top of the active fuel or when conditions requiring steam cooling are met. Following vessel depressurization, low-pressure system injection is assumed to fail. The scenario continues with the core becoming uncovered and core heatup beginning at about 35 minutes into the transient. Limited hydrogen is produced during core heatup. At about 65 minutes into the event, the core is reflooded before it becomes nonrecoverable (exceeding a 50% zirconium [Zr] melt fraction). During reflooding of the core a significant amount of hydrogen is generated. This hydrogen is transported to the suppression pool through the safety relief valve spargers and into containment where it is ignited and burned.

The selection of the SORV sequence was based on the reactor safety study methodology applications program (RSSMAP) study. In 1986, the staff questioned the absence of the station blackout (SBO) sequence (letter dated February 21, 1986). HCOG held the view that SBO is not a likely HGE contributor based on its relatively low core melt frequency (HGN-055). This conclusion is based on the results of the RSSMAP study (NUREG/CR-1659), which assumed Grand Gulf to be representative of the four plants with Mark III containments. The results of the GESSAR-II PRA (NUREG-0979) also found that SBO is a dominant contributor to the probability of core damage, although the core damage probability is quite low. The PRA results were reinforced by the staff findings reported in an NRC report (NUREG-1150). In view of these studies, the HCOG revised its submittal to account for SBO. These revised results are contained in two reports transmitted by letters dated January 8 and September 9, 1987. The staff's review focused on those revised results and also drew on information from previous submittals.

Additionally, the review effort focused on the hydrogen production profiles that were derived using the BWR core heatup code (BWRCHUC) (described in Science Application Inc. and International Technical Services (ITS) reports to the staff; HGN-020, -031, -032, -034, -089, -096, and -132; and HCOG/NRC meeting August 28, 1984). The objective of this review was to ascertain the capabilities and the acceptability of the BWRCHUC for use in generating the hydrogen generation profiles. Particular BWRCHUC concerns were (1) the Zircaloy oxidation model, (2) the transient simulation capabilities, (3) the ability to predict the maximum expected hydrogen production rate, and (4) the ability to predict the total amount of hydrogen produced in each transient.

5.2 Evaluation

The evaluation was divided into two parts: (1) the establishment of an acceptable HGE scenario and (2) the acceptability of the BWRCHUC to estimate hydrogen production histories.

5.2.1 Acceptable HGE Sequence

The analysis required by 50.44 is to be based upon an accident sequence which is "acceptable to the staff" and is at the same time limited to "recoverable" events. The rule, however, does not provide criteria for the determination of "acceptability" or "recoverability."

The staff position with regard to recoverability is that there should be a reasonable expectation that the original core geometry is generally maintained. However, a quantitative definition of a degraded core state that is recoverable is not required. The degraded core condition is a condition in which the reactor core has experienced or . . . at the onset of experiencing damage from excessive temperature (including permanent deformation or localized melting). Inherently in the degraded core condition is an extended loss of coolant injection without a chance of immediate recovery. The purpose here is not to associate core recoverability with detailed phenomena of cladding or fuel melting and relocation, but rather to provide a reasonable cut-off as far as the deterministic calculation of hydrogen production is concerned. The total amount of hydrogen production which must be considered is specified in the rule itself. It is in this limited sense that the term "recoverable" is used in this evaluation.

HCOG proposed a definition of recoverability in terms of the fraction of Zircaloy cladding which has reached or exceeded the Zircaloy melting temperature of 2170 degrees K. The staff accepted a 50% Zircaloy clad melt fraction as the cut-off point for "recoverability" based on HCOG's report that analyses indicate that at this point significant fuel melting is in progress. It is the staff's judgement that the maintenance of the original core geometry after damage to this extent is unlikely. Therefore, for the purposes of hydrogen rule considerations and hydrogen generation rate estimates, the 50% Zircaloy melt fraction criterion is acceptable.

With regard to the "acceptability" of sequences, the staff considered two criteria: (1) the likelihood of a given sequence and (2) the contribution to risk from a given sequence. Based upon NUREG-1150, the staff concluded that (1) the most likely HGEs would occur with the reactor vessel depressurized, (2) the potential for greater consequences is associated with HGEs at high pressure, and (3) the risk from all HGEs is estimated to be extremely low.

In assessing which sequences should be considered by HCOG, the staff also considered the uncertainties associated with low- and high-pressure events. For low-pressure events the requirements of the rule force conditions which are physically unrealistic (e.g. that the core be recoverable yet 75% of the Zircaloy is oxidized). This results in sequences which are somewhat artificial and therefore considerably uncertain. For high-pressure events these uncertainties are further complicated by a further lack of experimental data.

The staff therefore judged that it is sufficient to consider only low-pressure sequences because (1) the overall risk from HGEs is believed to be low, (2) from a risk perspective the reduced likelihood of a high-pressure HGE is likely to offset the potentially higher consequences of such an event, and (3) the additional uncertainties associated with high-pressure event progression.

5.2.1.1 The HCOG's Base-Case Scenario

The base case scenario proposed by HCOG results from a transient caused by loss Mark III SER

of offsite power, subsequent reactor scram, MSIV and containment isolation, and one SORV. All ac-powered reactor makeup systems are assumed to fail initially. However, the reactor core isolation cooling (RCIC) system which is dc powered and/or the fire truck diesel supply are available at Grand Gulf. Emergency operating procedures require depressurization when the water level reaches the top of the active fuel region and some low pressure injection system is available. It is assumed that following depressurization, the low-pressure systems fail to inject. The core becomes uncovered and core temperature begins to rise at about 35 minutes into the transient. As the core temperature continues to rise some hydrogen is produced. At about 65 minutes into the transient, the core is assumed to be reflooded at a high flow rate. During the reflooding of the core, large amounts of hydrogen are produced and transported to the containment through the safety relief valves. At this point in time, the core has reached the recoverability criterion (i.e., the Zircaloy melt fraction is at about 50%).

A variation to the SORV sequence is the SBLOCA scenario resulting from a hypothetical drywell break. The drywell break is essentially the same as the SORV sequence except that the hydrogen is discharged into the drywell as well as through the safety relief valves (SRVs). The staff considered this conservative sequence in order to evaluate the effect of hydrogen burning in the drywell where essential control equipment cabling is found.

5.2.1.2 Station Blackout and NUREG-1150

The results of the GESSAR-II PRA (NUREG-0979) also found that SBO is a dominant contributor to the probability of core damage, although the core damage probability is quite low. These results were reinforced by Grand Gulf findings documented in NUREG-1150. Subsequently, the HCOG submitted information to account for SBO to the staff by letters dated January 8 and September 9, 1987.

The results of analyses of Grand Gulf documented in NUREG-1150 indicate that the most probable HGEs result from SBO. The most likely of these sequences (designated as TBU sequences) consists of loss of offsite power followed by the failure of onsite ac power in divisions 1 and 2, the failure of high-pressure core spray (HPCS) and reactor core isolation cooling (RCIC), and depressurization of the reactor vessel. TBU represents more than 90% of all HGE sequences and more than 93% of all low-pressure HGE sequences. The development of the loss-of-offsite-power (depressurized vessel) sequence begins by boiling off the entire reactor vessel coolant inventory. With the core dry but at pressure, the operator depressurizes the reactor to increase the length of time available to support core recovery before the initiation of core damage. Following vessel depressurization the core begins to heatup causing oxidation of incore Zircaloy and core damage. Before core damage progresses to a point where a nonrecoverable core geometry could develop, a reactor vessel reflood system is assumed to be recovered. This reflood system then covers the fuel region with water terminating the event with a degraded, but recoverable, core geometry.

5.2.1.3 TBU the "Acceptable Sequence"

The HCOG had considered the applicability of the various significant sequences identified in the draft version of NUREG-1150 to the HCOG program. The scenarios were divided into three categories; the short-term (about 1 hr) damage states TBU, TBUX, TCUX, the intermediate (4-6 hrs), and the long-term (8-10 hrs) sequences TB, TBU1, and TQUX (HGN-123). Differences, however became apparent

when the results of the revised NUREG were reviewed. The revised version of NUREG-1150 estimates TBU to be the dominant HGE sequence, which accounts for 93% of core damage frequency. The phenomenology of the TBU sequence is similar to that of the HCOG base case with regard to the SORV. However, the HCOG experimental testing and analyses, which encompasses the TBU sequence, assumed the igniters were continuously powered, including during the portion of the transient when ac power was not available.

In addition, the rule requires that the containment structural integrity and a safe shutdown be established and maintained. The ability to satisfy these requirements depend on both the total amount and the rate of hydrogen production. To estimate the maximum hydrogen production rate and the total amount of hydrogen produced, the rate of water supply in the recovery phase of the HGE is critical. For purposes of the hydrogen control rule, the TBU sequence as described in NUREG-1150 (which encompasses the SORV as described by the HCOG) is an acceptable sequence leading up to core recovery. In summary, the TBU is acceptable for the time sequence of events and for the hydrogen production rate and total amount.

(In probabilistic risk assessment notation, the terms TB, TBU, TBUX, TBU1, TCUX and TQUX denote the following: TB - Station Blackout. TBU - Loss of offsite power (LOSP) with failure of all high pressure functions. The SRVs are operational and the vessel is depressurized. TBUX - LOSP with loss of all AC divisions and failure to depressurize. TBU1 - LOSP with failure of AC divisions 1 and 2 and of the high pressure core spray (HPCS). The reactor core isolation cooling (RCIC) operates for 6-8 hours before failing due to high pressure in the turbine exhaust. TCUX - ATWS with LOSP loss of AC and HPCS and RCIC failures. TQUX - Failure of all ECCS functions except power.)

5.2.1.4 Hydrogen Generation Profiles

Recovery of cooling water flow is effectively bounded between 150 gpm from a single control rod drive cooling pump to 5000 gpm from the emergency core cooling system (ECCS) low-pressure high flow-rate core recovery system. The hydrogen generation profiles for these extremes are qualitatively and quantitatively different. The probability of a high-flow-rate recovery is expected to be higher than that of a low-flow-rate system, because there are more high-flow systems (or combinations of systems) to inject water into a depressurized vessel; hence, it is reasonable to assume that the operator will attempt more often to recover one of the high-flow-rate systems. A high-flow reflood rate is associated with a high, narrow spike of hydrogen release, while the low-flow reflood rate will yield lower hydrogen production rates but for longer times (see Figures 5.1 and 5.2) (HGN-132). These profiles have been estimated by the HCOG using the BWRCHUC code, which is discussed in Section 5.2.2. The total mechanistic estimated amount of hydrogen released in the low-rate reflood case is higher than that released in the high-rate reflood case. The hydrogen peak release of the high-flow reflood case is about 35 seconds wide at half maximum, while for the low-reflood case significant hydrogen release lasts about 8.5 minutes. Table 5.1 shows a summary of the main features of both cases.

Table 5.1 Hydrogen Release Profiles

Reflood Rate (gpm)	Length of Transient (min.)	Total Hydrogen (lbs)	Peak Rate (lbs/sec)	Width at Half Max (sec)
150	80.0	903.4	0.95	510
5,000	25.8	604.8	8.00	35

5.2.1.5 Non-Mechanistic Hydrogen Release Profile

The hydrogen rule requires consideration of metal-water reaction (MWR) for 75% of the Zircaloy cladding surrounding the active fuel region. However, the estimated amount of metal-to-water reaction in either reflood rate scenario is far less than the required 75%. For Grand Gulf, the active core region cladding is 79,100 lbs. For the oxidation to proceed as: $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$, the amounts of Zr that correspond to the hydrogen released in Figures 5.1 and 5.2 are 20,450 lbs and 13,700 lbs which represent about 26.0% MWR and 17.0% MWR respectively.* (This MWR incorporates channel box and stainless steel oxidation.) A mechanism was needed to increase the release profiles to 75% MWR of the active core region cladding, as required by the rule. The 75% MWR of the Zr in Grand Gulf is 59,300 lbs, which when oxidized will create about 2,600 lbs of hydrogen. It must be pointed out that mechanistic models that account for 75% MWR of cladding oxidation result in a severely damaged core exceeding the recoverability criterion. There are many possible scenarios that can be hypothesized to yield 75% MWR cladding oxidation; however, no attempt is made to estimate the phenomena associated with such an oxidation level because it would require an unreasonable recovery criterion.

As discussed previously, after core quenching in the low-rate reflood case, the calculated maximum amount of metal-water reaction is limited to about 26%. To meet the rule requirement of 75% MWR, the HCOG submitted a nonmechanistic model used to predict hydrogen production based on an energy balance in a severely damaged core. It assumed that such a core has energy losses at least adequate to remove decay energy in the core, the energy produced by continued oxidation of Zircaloy, and excess stored energy in the core. It also assumed that termination of oxidation at 75% MWR takes place by quenching of the core and removal of all excess energy (HGN-034). Considering the above, the oxidation rate will support a constant hydrogen release of about 0.10 lbs/sec. The staff finds that this release rate is acceptable for hydrogen release to 75% MWR, as required by the rule.

Therefore, for the scenarios shown in Figures 5.1 and 5.2 the "tails" correspond to 1700 lbs and 2000 lbs of hydrogen, i.e., an extension of about 17,000 seconds (4.7 hours) and 20,000 seconds (5.6 hours), respectively.

The staff concludes that (1) mechanistic models can not predict the required 75% MWR of cladding oxidation in the active fuel region without core damage

*Zircaloy is assumed to consist of 100% Zr. The actual composition of Zircaloy-2 in weight percent includes, Sn: 1.2-1.7, Fe: 0.02-0.07, and Ni: 0.05-0.15

HYDROGEN GENERATION RATE (lbm/sec)

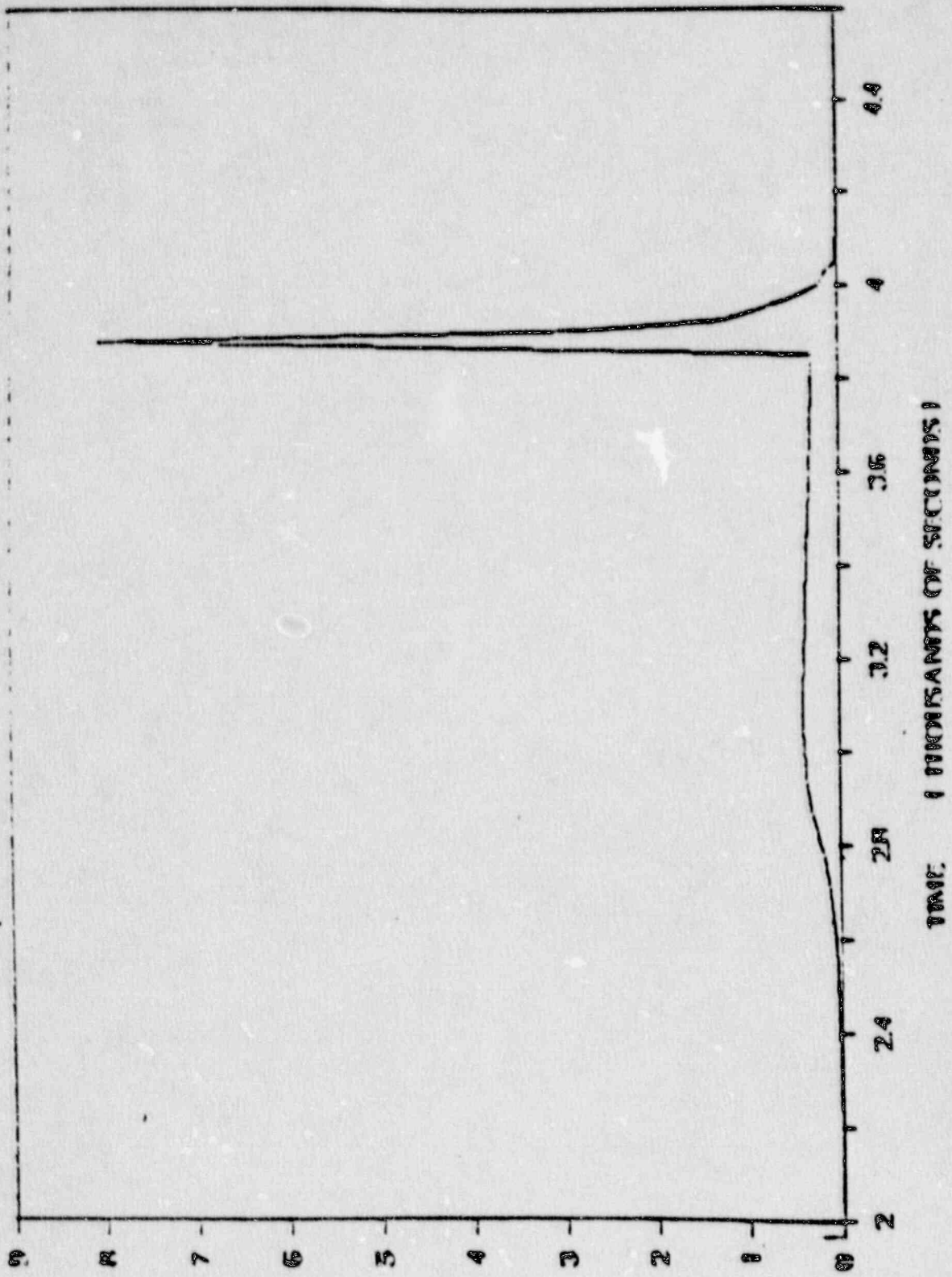


Figure 5.1 Hydrogen generation rate (5000 rpm reflood)

HYDROGEN GENERATION RATE

(RELEASE HISTORY A - FBI GPM REFLOOD)

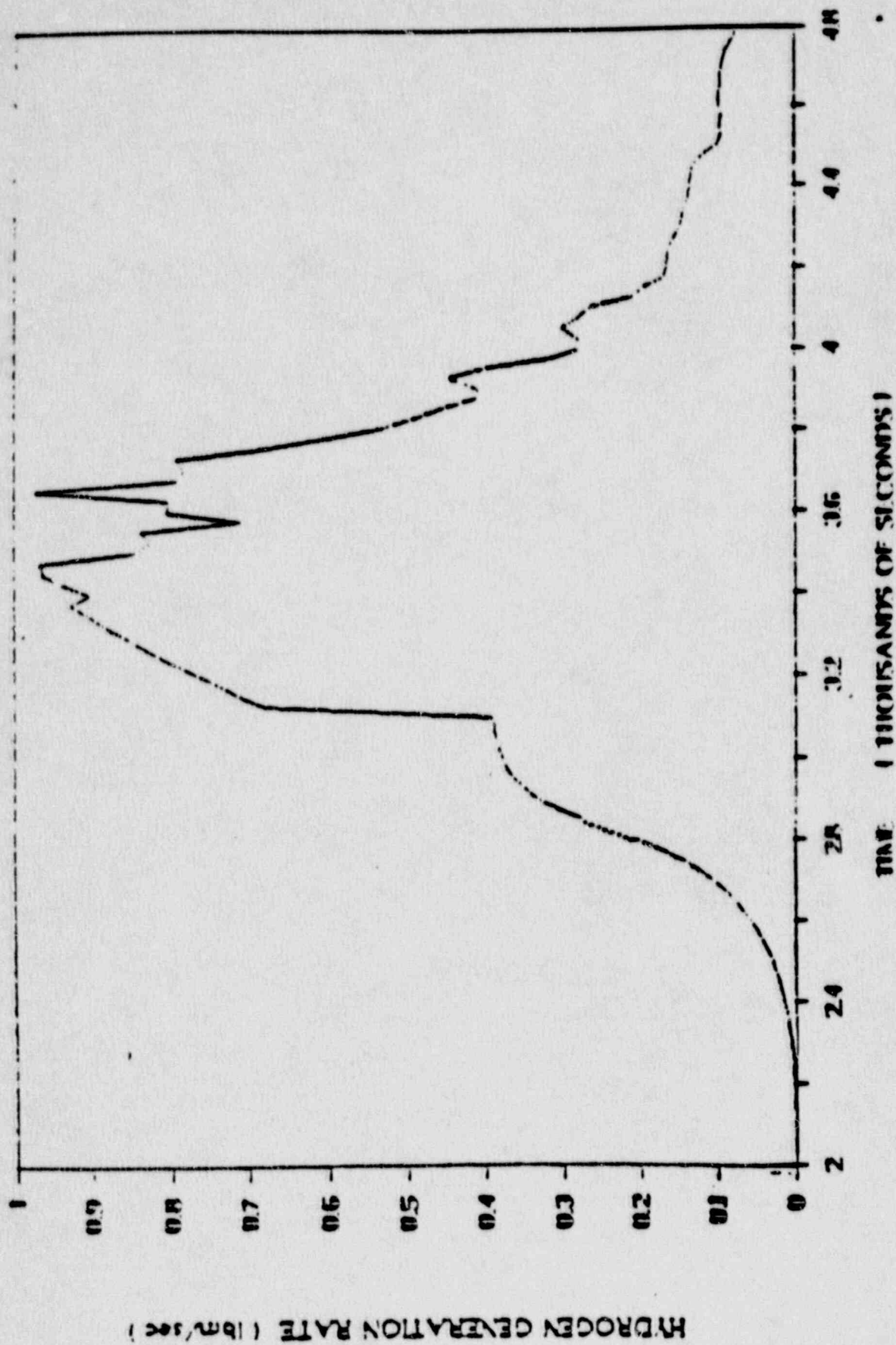


Figure 5.2 Hydrogen generation rate (150 gpm reflood)

beyond the recoverability criterion and (2) the use of a non-mechanistic release model based on heat balance is reasonable and acceptable. This leads to an oxidation rate producing 0.10 lbs of hydrogen per second, requiring an extension of about 4.7 and 5.6 hours for the scenarios of Figures 5.1 and 5.2, respectively.

5.2.2 BWR Core Heatup Code (BWRCHUC)

5.2.2.1 Introduction

The BWRCHUC has been used by the HCOG to calculate hydrogen rate release profiles for the hydrogen generation sequences described previously. The areas the staff considered in the review of the BWRCHUC is discussed below.

The BWRCHUC was not validated or benchmarked to global core experimental data, rather it relies on collective engineering judgment and understanding of the phenomena taking place in a core disruptive accident. The lack of benchmarking or validation is due to the absence of suitable experimental data. This lack of benchmarking prevents the results of the code from being used directly without appropriate consideration of selected input parameters. The results of the BWRCHUC should be seen as an engineering estimate of the anticipated phenomena. Accordingly, the code review was aimed at the reasonableness of the modeling, the physical significance of the assumptions, and possible conservatisms in the estimate. Reasonableness was assessed in terms of models and hypotheses that have been advanced by other researchers in this field and any other evidence that could be gleaned from whatever limited and partial experimental information was available. For code modeling, the TBU sequence for an HGE was considered equivalent to the SORV sequence (paragraph 5.2.1.4) with respect to the depressurization and core uncover time, thus similar as far as hydrogen generation is concerned. This sequence is the simplest and most straightforward, thus having the highest probability of being modeled correctly.

The BWRCHUC is a well-written computer code in that (1) it faithfully represents the BWR geometric core design and (2) the models included in the code are adequate to cover the specific HGEs selected for analysis by the HCOG and represented by the TBU sequence. Modular architecture has been used extensively, where each module (subroutine) in the code treats a different phenomenon or aspect of the problem. The code is built by connecting the various modules with executive routines. The numerical solution technique applied in the BWRCHUC is apparently as good as any employed in severe accident codes. Numerical stability, as reported by HCOG, is evidenced by the graphs of code output and the fact that reflooding calculations can be run.

The BWR core geometry is very complex. Some subtleties of the geometry have the potential to affect the prediction of hydrogen generation. Therefore, it is appropriate that a best-estimate code contain a representation of the geometry that is as complete as is reasonably achievable. This has been done in the BWRCHUC. Considerable attention has been drawn to the fact that the BWRCHUC allows for a different two-phase water level to be predicted in each fuel assembly represented. A separate level for the core bypass level varies according to the bundle power since the water in an assembly is assumed to be saturated at the system pressure. Water levels may also vary because the void fraction of

the water in an assembly is a function of assembly power. The bypass level calculation further assumes that water in the bypass is subcooled and thus corresponds to the collapsed water level in the core. There is a hydraulic connection between the assemblies and the bypass so that the water level in the bypass is reduced as the core water inventory is boiled away by the decay heat generated within the assemblies. This representation closely corresponds to a partially covered BWR/6 core at low pressure before any structures in the core reach temperatures significantly greater than saturation.

5.2.2.2 Phenomenological Assumptions

A model for channel blockage was included in the code, but has not been employed in the calculations since experimental results did not support total flow blockage. The blockage model assumed that the fuel rod cladding melts while the channel box remains intact. Molten cladding is then assumed to slump and refreeze within the channel forming a complete blockage, which prevents steam from reaching the Zircaloy surfaces within the assembly. In addition, steam generation below the blockage pressurizes that portion of the assembly forcing the two-phase level in that assembly below the core plate. Since no steam enters the channel, all oxidation would stop. Experimental results from the PBF tests (HCOG presentation to NRC January 14, 1985) indicated that a reduction in the flow area as a result of Zircaloy slumping did occur, but that complete blockage did not form. Without the channel blockage model hydrogen production is maximized all other conditions being the same. The lack of clad motion or channel blockage is a very conservative assumption with regard to hydrogen production.

It is assumed that the control rods will remain intact since it is consistent with the recovery criterion. Under certain conditions, experimental evidence (R. O. Gaunt) suggests that BWR control rod blades could melt early in the core heatup phase of a transient. This would lead to the possibility of local loss of control; thus, when the core is reflooded, local criticality could result in intense heat production and core damage beyond the limits of recoverability. Therefore, control rod melt would be beyond the scope of this program.

5.2.2.3 Steam Production

The modeling of steam generation within the reactor pressure vessel (RPV) can have a significant impact on the quantity of hydrogen generated. While some aspects of steam generation are accurately modeled in the BWRCHUC, other sources of steam are not modeled at all. The steam generation modeling generally is incomplete; however, for the most likely HGE considered, the steam sources not represented do not significantly impact production of hydrogen.

Within the BWRCHUC, the following five sources of steam generation are modeled:

- deposition of the decay power from that portion of the fuel assembly below the two-phase level into the saturated water within an assembly
- heat transfer (by nucleate boiling) from portions of fuel rods, channels, control blades, and the core shroud that are at temperatures greater than saturation when they are covered by the two-phase level

- radiative heat transfer from portions of control blades that have temperatures higher than saturation to surrounding channel walls when the two-phase level within the channel is at or above the portion (at elevated temperatures) of the control blade
- flashing of water in the downcomer and lower plenum as a result of reductions in the RPV pressure (Pressure-time history is provided by user input.)
- evaporation of core spray droplets entering the top of a fuel assembly during reflooding of the core

Steam generation resulting from flashing of the water inventory within the fuel assemblies and in the bypass region is not modeled. If system pressure would decrease, flashing would occur. However, the selected IGE sequence does not involve changes in pressure vessel pressure after hydrogen generation has begun. It is assumed that the RPV pressure is constant for at least 10 minutes, which is the time required to remove the bulk of the heat in the lower plenum structures. Therefore, the lack of a flashing model is not a factor.

Downward relocation of molten Zircaloy can have a large effect on steam generation. If the two-phase level is above the core support plate, molten Zircaloy can run into water. Quenching or relocating Zircaloy in water would enhance steam and hydrogen generation. This phenomenon is not modeled in the BWRCHUC. However, there is no water above the core support plate when Zircaloy melting occurs. Modeling of the melt relocation into the water would not increase the quantity of hydrogen produced compared to that which will be produced in the core reflood because of the more favorable surface-to-volume ratio.

An oxidation cutoff temperature is used in the BWRCHUC as a surrogate for the effect of cladding and channel box relocation and subsequent quenching thereby removing the Zircaloy from the oxidizing environment (R.O. Gaunt and HCOG presentation to NRC January 14, 1985). The HCOG estimated Zircaloy oxidation vs. Zircaloy temperature and concluded that 2400°K is a conservative representation to account for this effect (HGN-032, item 4). Based on the evaluation performed by ITS, the staff has accepted the 2400°K as the irreversible oxidation cut-off temperature (letter to HCOG June 4, 1985).

In the reflood stage, quenching of Zircaloy that is at temperatures higher than the saturation temperature is nonmechanistically estimated. This can lead to overprediction of the steam generation rate during the reflood phase. For nodes that are more than 100°K above saturation, quenching is assumed to take place in a single time step thus accelerating the heat transfer process and steam production.

Steam flow in the bypass region is underpredicted. However, the effect of this underprediction of the bypass steam flow rate on the overall prediction of hydrogen release is small. Overall steam generation rates in an overheated core could be underpredicted for transients in which the two-phase level is above the core support plate. In the staff's judgment, the extent of this underprediction is small compared with the uncertainties associated with predictions of this nature.

In summary, steam generation before reflood is reasonably well predicted provided the RPV pressure has been constant for approximately 10 minutes. In the sequence considered, the RPV is depressurized and steam and hydrogen production take place under conditions of constant pressure.

5.2.2.4 Hydrogen Generation

As with the modeling of steam generation, the approach to modeling hydrogen generation is reasonable considering the difficulty of representing the phenomena to modelling techniques. The lack of models for a few relevant phenomena combined with some of the assumptions made for phenomena that are modeled, leads to some uncertainty with regard to the predictions of the hydrogen generation rate during the dominant HGE. This uncertainty is expected to be negligible (i.e., possess compensating effects) in the present context. However, considering the conservative assumption of no clad motion, we concluded that the overall hydrogen generation estimate is conservative.

The considerations/phenomena that are related to hydrogen generation and are not modeled or are underpredicted are listed below.

- Oxidation below the location at which melting occurs is not modeled.
- Because of the underprediction of steam in the bypass channel, oxidation of stainless steel and the outside of the channel is probably underpredicted.
- Ballooning of the cladding and localized failure resulting in simultaneous interior and exterior oxidation is not modeled, thus limited hydrogen underprediction may result.
- Film boiling in a quenching mode is not modeled. This leads to higher rate of hydrogen production for shorter time periods. It is not clear that an overall underprediction will result.
- In the reflooding stage, vaporization of droplets that enter the top of fuel assemblies by radiant heat transfer does not remove heat from fuel rods. This results in a conservative hydrogen production if the maximum temperature is below the cut-off and possibly not conservative if it is above the cut-off. It is not clear if the overall effect is nonconservative.

Reaction rates of Zircaloy and stainless steel with steam are calculated using the Arrhenius relationship. The reaction rate constants used in these expressions were derived by others from experimental results. This modeling of reaction rates and the associated heat generation is appropriate and consistent with what is used in other severe accident modeling codes. A hydrogen blanketing factor is included in the formulation of the Arrhenius reaction rate expression. Hydrogen blanketing refers to the possible limitation of the oxidation rate from the diffusion rate of steam through the hydrogen emitted from the oxidizing surface. While the process represented by the hydrogen blanketing factor is real, a reduction of the oxidation rate is almost certainly not realized under the conditions expected during core damage in BWRs. Diffusion of steam through the oxide layer is the rate-limiting process. Therefore, the hydrogen blanketing effect was not considered in the HCOG calculations, which represents a slight conservatism.

Because the oxidation rate varies exponentially with temperature, the representation of intact Zircaloy nodes reaching temperatures significantly higher than the melting temperature leads to higher oxidation than would be predicted if melting were explicitly treated. Therefore, this is a conservative assumption. However, because the oxide layer is generally thick at these times, the actual quantity of additional oxidation is considered to be small. One could view this enhanced oxidation as a nonmechanistic approach to representing the initial enhancement in oxidation that probably accompanies slumping molten Zircaloy.

Heating of the cladding reduces the tensile strength and increases ductility. Simultaneous heating of the fuel and gases within the cladding leads to pressurization of the rod from within. Ballooning of the cladding and localized failure may occur before melting. Failure of the cladding would allow the interior surface to be exposed to steam. It is therefore entirely possible that both the interior and exterior surfaces of the cladding will undergo oxidation. Since this possibility is not modeled in the BWRCHUC, hydrogen generation rates and total hydrogen generation could be underpredicted. However, in the staff's judgment, the conservatism in the assumption that there is no clad slumping will adequately compensate for this potential underprediction.

Overall, it is the staff's judgment that the modeling of the hydrogen generation rate in BWRCHUC is reasonable and the total and peak hydrogen production estimates are expected to be conservative.

5.3 Summary and Conclusions

There was a twofold objective in this portion of the evaluation corresponding to two requirements of 10 CFR 50.44:

- (1) "Use accident scenarios that are accepted by the NRC staff." Paragraph (c)(3)(vi)(B)(3).
- (2) "Provide an evaluation of the consequences of large amounts of hydrogen generated after the start of an accident...up to and including 75% of the fuel cladding surrounding the active fuel region..." Paragraph (c)(3)(vi)(B)(1).

The first requirement corresponds to the dominant accident sequence that leads to an HGE. The information for such a sequence was derived from HCOG submissions and conforms to the revised (final) version of NUREG-1150. The TBU sequence was found to represent 93% of all HGE sequences and consists of loss of offsite power followed by failure of onsite ac power in divisions 1 and 2 and failure of the HPCS and RCIC. The TBU sequence was found to be similar to the SORV, which was initially proposed by HCOG. Thus, for hydrogen generation purposes the TBU sequence satisfies the requirements of 10 CFR 50.44.

The second requirement is limited to the acceptability of the BWRCHUC to estimate hydrogen generation profiles. The essential characteristics of such profiles are the peak rate, its duration, and the total amount of hydrogen produced.

The staff reviewed the BWRCHUC code on the basis of these requirements and information submitted by the HCOG. The staff finds the BWRCHUC code acceptable for use in calculating hydrogen production profiles. Therefore, the staff finds that the profiles estimated by HCOG using the BWRCHUC are acceptable for use in demonstrating compliance with 10 CFR 50.44.

6 CONTAINMENT RESPONSE - ANALYTICAL MODELING

In view of the quarter-scale test program, the emphasis on analytical methods for predicting containment response to hydrogen burning has significantly diminished. This conclusion is based on the broad range of hydrogen release rates in which diffusive combustion is expected to occur. The staff believes the evaluation of the survivability of essential equipment should be based on the QSTF data. Therefore, there is limited value in pursuing such analytical methods as the CLASIX-3 code; thus, the following evaluation addresses HCOG's effort to resolve the CLASIX-3 code analysis generically. As such, this effort is only relevant at low hydrogen flow rates that are near the flame extinguishment limit.

As documented by various staff evaluations performed before the completion of the quarter-scale test program, the CLASIX code has been the principal analytical tool in predicting the containment response as a consequence of burning hydrogen for plants with an ice condenser or Mark III containment. The CLASIX code or the CLASIX-3 code (which is the latest modified version that includes Mark III containment features) deals with deflagration (discrete-type) hydrogen burning. The code is a multivolume containment code that is used to calculate the containment pressure and temperature response in separate compartments. Moreover, the code has the capability to model characteristics that are unique to Mark III containments while tracking the distribution of the atmospheric constituents (i.e., oxygen, nitrogen, hydrogen, steam).

The staff's desire to demonstrate verification/validation of the CLASIX code has been an extensive effort. CLASIX results have compared well with results of other NRC-accepted analytical codes and hydrogen burning experiments. Furthermore, the HCOG has performed additional code validation by comparing the more recent Nevada Test Site large-scale hydrogen experiments to CLASIX-3 code predictions. This is documented in HCOG's letter (HGN-113) dated January 8, 1987. With regard to hydrogen burning, the focus on code validation has been on pressure predictions because temperature comparisons are more difficult to predict due to their time and spatially dependent fluctuations.

As discussed earlier, the major element of the HCOG's program is the quarter-scale test program. The data obtained from tests were used to perform equipment survivability analysis (see Section 7). These tests revealed that diffusion flames on the suppression pool surface can exist at a hydrogen injection rate as low as 0.02 lb/sec under certain background conditions. As such, it is expected for a significant portion of postulated degraded core hydrogen profiles that diffusion flames will be the dominant combustion mode. Since CLASIX-3 does not model diffusion flames, these results have a significant bearing on the extent to which the CLASIX-3 code can be relied upon to predict containment temperature environments, which further emphasizes the importance of the test program.

6.1 Localized Combustion/CLASIX-3

As noted in Section 3.1, testing performed in the QSTF revealed that combustion which occurred below the flame extinguishment limit was not global deflagration-type combustion; but "localized combustion." Localized combustion is characterized as weak flames or weak volume burning through a marginally combustible hydrogen-air-steam mixture. By letters dated June 10, 1986 (HGN-092-P), and December 15, 1987 (HGN-111-P), the HCOG provided various analyses to demonstrate that the CLASIX-3 model provided a boundary calculation for the combustion occurring below the diffusion flame threshold. This model employed a combustion mechanism that produces a more severe global thermal environment than has been measured locally in the QSTF for localized combustion.

The staff requested Sandia to review this approach. In its HGN-092-P submittal, the HCOG compared CLASIX-3 predictions for a quarter-scale model to experimental results of the corresponding test. The CLASIX-3 predictions of the wetwell volume showed that the temperature profile exceeded the volume-weighted average of experimental data. On the basis of this analysis, the HCOG concluded that CLASIX-3 yields conservative predictions of thermal environments inside containment for very low hydrogen flow rates. The staff questioned the applicability of using experimental data pertaining to local phenomena to demonstrate the capability of a lumped volume code. SCL shared the staff's concerns and recommended that the local combustion phenomenon observed in the QSTF warranted further evaluation.

In its HGN-111-P submittal, the HCOG provided a comprehensive assessment of localized combustion seen in several quarter-scale tests. In these tests, combustion activity, as evident by thermocouple responses, is widespread during periods of low hydrogen flow. Test data do not indicate that concentrated flame energy deposition occurs at fixed locations. Energy deposition appears to be rapid and diffuse and is dominated by convective mixing, combustion-induced turbulence, plume influence, and background gas flows. Typically, peak temperatures recorded during localized combustion are relatively low and persist for short durations. The temperature responses are cyclic and return to relatively low background levels.

The HCOG analyzed five quarter-scale tests conducted during the scoping test phase of the program in an effort to better understand localized combustion. At the low hydrogen flow rates, these tests demonstrated certain repeatable trends and the thermocouple activity observed recurrent and generally predictable. Some of the findings resulting from the evaluation of localized combustion are briefly described below.

- (1) In tests without sprays, combustion was generally widespread. Whereas, when sprays were activated, combustion appeared to be suppressed in open chimneys (i.e., annular quadrants) as a result of cooling effects and shifts in global flow patterns. Also, enhanced mixing resulting from sprays induced slightly higher temperatures in some areas, but not appreciably higher than those recorded when sprays were off.
- (2) Comparing the scoping test results, it appeared that the location of the SORVs did not have as significant an effect as other parameters, such as variation in hydrogen flow rates.

- (3) Probably the most important finding pertained to combustion activity in the vicinity of the hydrogen igniters. The closest thermocouple located to a nearby igniter was about 15 inches laterally or about 6 inches laterally and 15 inches above the igniter. Data indicated that temperatures close to igniters were generally no more severe than recordings several feet away. Also, when comparing the effects of blockages above igniters to open flow regions, a significant increase in temperature was not observed. Combustion energy dispersion was prevalent.

To further support its findings of turbulent mixing induced by combustion, the HCOG included a discussion of a test in which pool flames were observed. A thermocouple was placed about 1 foot directly above the pool surface over an active sparger. Readings indicated that at low hydrogen flows the flames at this location appeared to be intermittent and unstable. The temperature response did not exceed 425°F as a result of these unsteady pool flames. The HCOG contends that, because of the efficient mixing, one should expect local hydrogen concentrations elsewhere in the facility to be less than at the suppression pool surface. Moreover, this fact coupled with temperature readings (discussed above) and the absence of visual indications regarding flame formations above the HCU floor, is not strongly supportive of a hypothesis that sustained high-temperature localized combustion zones will be established at very low hydrogen flow rates. In addition, HCOG indicates that the temperatures generated from pool burning at low hydrogen flow rates (i.e., about 0.15 lb/sec) from resultant hot plumes represent a more severe thermal environment than localized combustion.

As part of this assessment, HCOG provided additional information with regard to the role of the CLASIX-3 code in its analyses and the conservatism used for containment modeling. While global or large volume deflagration, as modeled by CLASIX-3, did not occur in the QSTF, the HCOG contends that the CLASIX-3 modeling would conservatively bound the observed localized combustion environment. To assess the severity of the environment from an equipment survivability perspective, the HCOG compared the thermal loads created by the most severe localized combustion measurements at the QSTF to the corresponding CLASIX-3 temperature profile. The results of this comparison show the CLASIX-3 profile generates a significantly more severe environment than that produced by localized combustion.

The staff requested that SNL review this issue along with the consideration of scaling aspects. SNL determined that the HCOG adequately addressed the likely locations for localized combustion and identified reasonable bounds for the most threatening thermal environment for equipment located near regions of localized combustion or in the resultant hot plumes. Moreover, the thermal local comparison in combination with the HCOG's discussion of localized combustion provide adequate justification that the CLASIX-3 thermal load would be more severe than that experienced in the QSTF for low hydrogen injection rates. In conjunction with discussions contained in the QSTF test report (HGN-121-P), SNL asserts that there is reasonable assurance that the thermal response at full scale will be no more threatening than that experienced in the QSIF. The staff has also evaluated this issue and concurs with SNL's assessment. Based on the modeling methodology used in the referenced submittals (e.g., low hydrogen flow rates and the focus on the wetwell profile), the staff finds that the CLASIX-3 prediction would be acceptable in determining the containment environmental conditions as a consequence of localized hydrogen burning. Accordingly, these profiles could be used to evaluate the survivability of equipment.

6.2 Containment Pressure and Temperature Calculations

Letters HGN-092-P and HGN-109-P documented the HCOG's calculations of the containment pressure and temperature response based on postulated degraded core scenarios that are discussed in Section 5 using the CLASIX-3 code. To determine the adequacy of the hydrogen ignition system (HIS), the HCOG considered two types of accidents in its generic analysis: a stuck-open relief valve (SORV) transient and a small-break loss-of-coolant accident (SBLOCA) in the drywell. The component of the hydrogen release history that is of interest in this analysis is referred to as the "tail" portion and represents a nonmechanistically defined constant hydrogen production rate. As discussed above, the CLASIX-3 results bound the thermal environment that may be produced for low hydrogen release rates that are below the diffusion flame extinguishment limit.

HCOG provided a generic sensitivity study using the Perry Nuclear Power Plant containment characteristics for the CLASIX-3 model. In this sensitivity study, different parameters were varied to assess the effects on the calculated results. The staff focused on the most important parameter considered, which was the assumed availability of the containment sprays. HCOG had chosen to use the CLASIX-3 code predictions without sprays (i.e., for the SORV case) in its generic survivability study (discussed in Section 7). For the equipment survivability analysis to be generic, it became necessary to consider the no-spray case because fan coolers rather than sprays are part of the River Bend containment design.

For the SORV case, all mass and energy releases were directed into the suppression pool. The CLASIX-3 model used in the generic analysis simulated four compartments of the Perry containment: the drywell volume, the wetwell volume (bounded by the HCU floor and the surface of the suppression pool), the intermediate volume (bounded by the HCU floor and the refueling floor), and the containment volume (above the refueling floor). Figure 6.1 presents a schematic representation of the model. Ignition of hydrogen combustion was assumed to occur at a 6% hydrogen concentration with 65% combustion completeness. The CLASIX-3 SORV base-case model produce a transient in which the hydrogen was ignited in a series of burns in the wetwell volume. Figures 6.2 and 6.3 show the computed wetwell temperature and pressure profiles. The early portion of the transient resulted in the highest wetwell temperature. This is symptomatic of the hydrogen spiked release in the early phase of the release profile. Diffusion flames would be prevalent in this interval and would be beyond the range of use for the CLASIX-3 methodology. For the major portion of the temperature profile, the wetwell burns produce a peak wetwell temperature of above 800°F.

At the end of the hydrogen release period, the calculated hydrogen concentration in the containment volume did not reach the ignition criterion of 6%. In the CLASIX-3 calculation, the HCOG assumed a containment burn to occur at this lower concentration which resulted in the most severe pressure excursion, to approximately 23 psig.

6.3 Drywell Analysis

For the base-case analysis of a small pipe break in the drywell (DWB) the CLASIX-3 containment model was similar to the SORV case except the hydrogen/steam source

terms are directed to both the drywell volume and the suppression pool. The DWB scenario was chosen because of the potential and consequences for hydrogen combustion in the drywell. Of the cases studied, only the 2-inch DWB case had conditions where a hydrogen burn was predicted to occur. Ignition in the drywell was limited by lack of oxygen (i.e., below 5%) because air is forced from the drywell by vessel blowdown. The only burn predicted by CLASIX-3 for the 2-inch DWB case resulted in a peak drywell temperature of about 1050°F and a peak drywell pressure of about 13 psig. For the events considered, where high steam flows are directed into the drywell along with the diversion of most of the hydrogen to the suppression pool, the hydrogen threat to the drywell appears to be relatively small. As indicated in Section 7, HCOG performed thermal response analysis of selected drywell equipment. The results demonstrated that the equipment would survive the drywell burn.

6.4 Existence of Drywell Diffusion Flames

In the DWB case, air would be reintroduced in the drywell through vacuum breaker actuation or operation of the drywell mixing system. The drywell environment is predicted to be a hydrogen-rich/oxygen-lean mixture. When oxygen is reintroduced in the presence of an ignition source, a diffusion flame may result in the vicinity of the oxygen source. This possible combustion phenomenon is referred to as an inverted diffusion flame. This is a concern since the potential to establish a continuous inverted diffusion flame at the oxygen source may result in locally severe thermal loads.

By letters dated June 25, 1986 (HGN-091), and June 10, 1987 (HGN-119), HCOG evaluated the potential impact of inverted diffusion flames in plants with a Mark III containment. In the HGN-091 submittal, the HCOG discussed the criteria for establishing the existence of inverted diffusion flames. The HCOG indicated that flames will not occur in the drywell when conditions are outside the flammability curve. In the HGN-119 submittal, the HCOG further discusses the low likelihood of achieving the necessary combustible conditions in the drywell based on the CLASIX-3 predictions.

SNL reviewed the initial submittal and determined that the HCOG did not provide sufficient justification to preclude drywell burning. Specifically, SNL commented that the flammability limit merely establishes the limits that will allow flame propagation; burns that do not propagate into the mixture are not precluded by being outside the flammability limits. Furthermore, it was not obvious that the burning mixture should be expected to follow the path predicted by the HCOG.

Generally, there is a lack of experimental data to support the HCOG's position. However, recent risk studies do not support the DWB case as a dominant core-melt/degraded-core event for plants with a Mark III containment; therefore, further phenomenological investigation may not be warranted. Drywell break events are further discussed in Section 5. In addition, the expected redundancy (i.e., spatial separation of equipment performing the same function) of the critical equipment should compensate for possible locally severe thermal loads.

The staff believes there is a reasonable level of assurance that the consequences of a drywell break event would not pose a significant threat to containment integrity and would not preclude safe shutdown of the plant. However, the staff believes, as part of IPE process that each licensee of a plant with a Mark III

PERRY CLASIX-3 MODEL

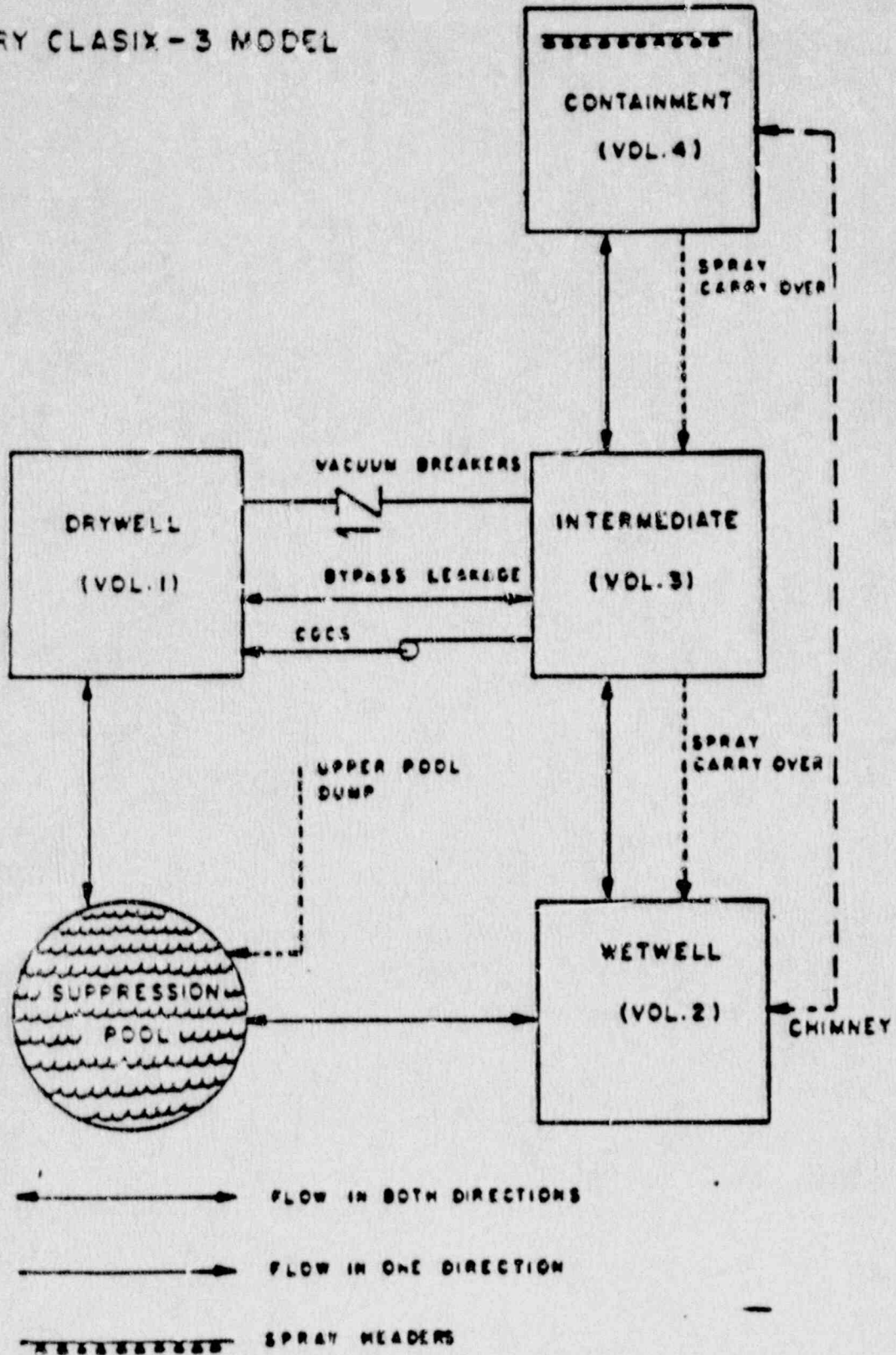


Figure 6.1 Perry CLASIX-3 Model
Source: HGN-092-P

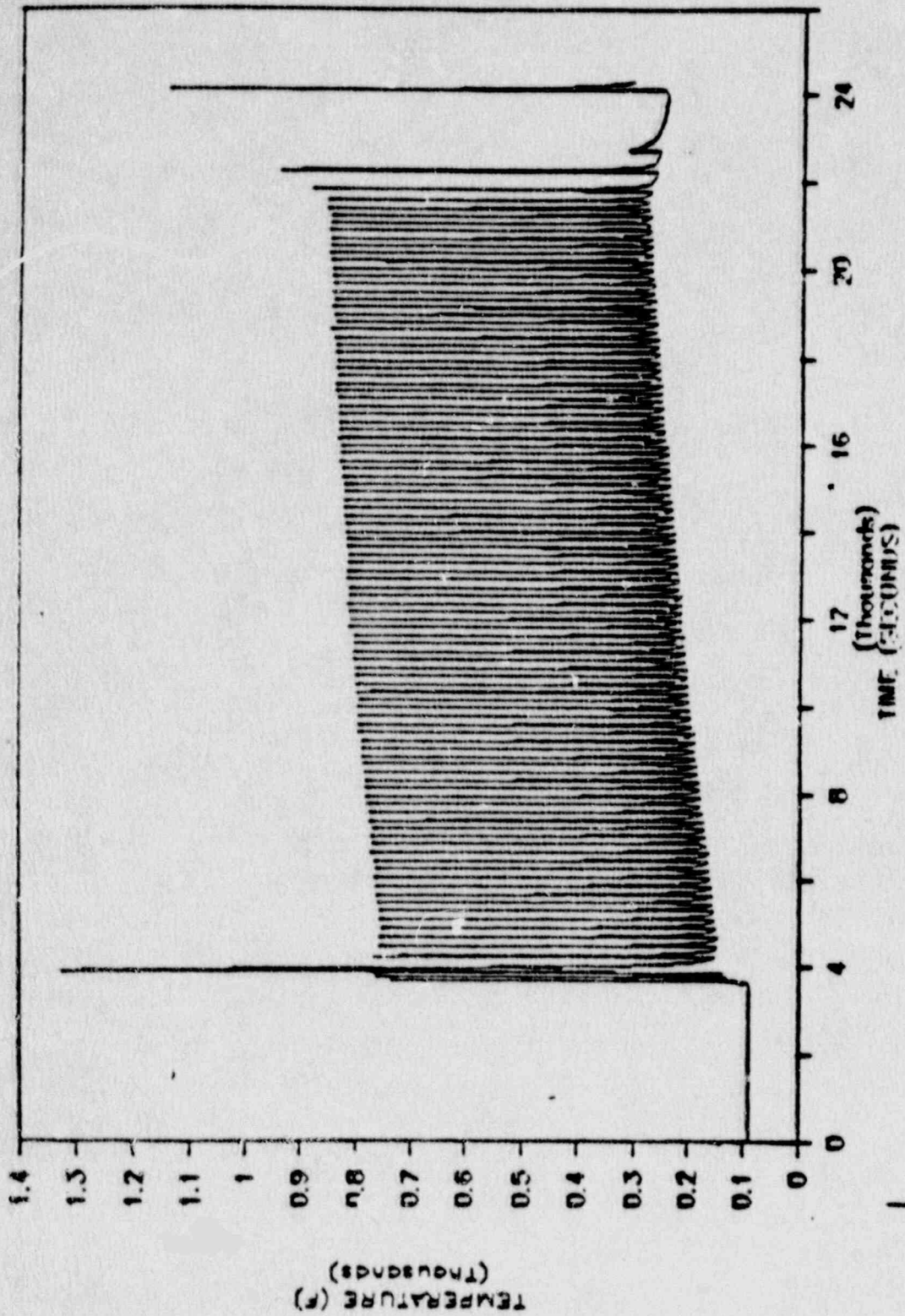


Figure 6.2 SORV with no spray-wetwell temperature
 Source: HGN-109-p

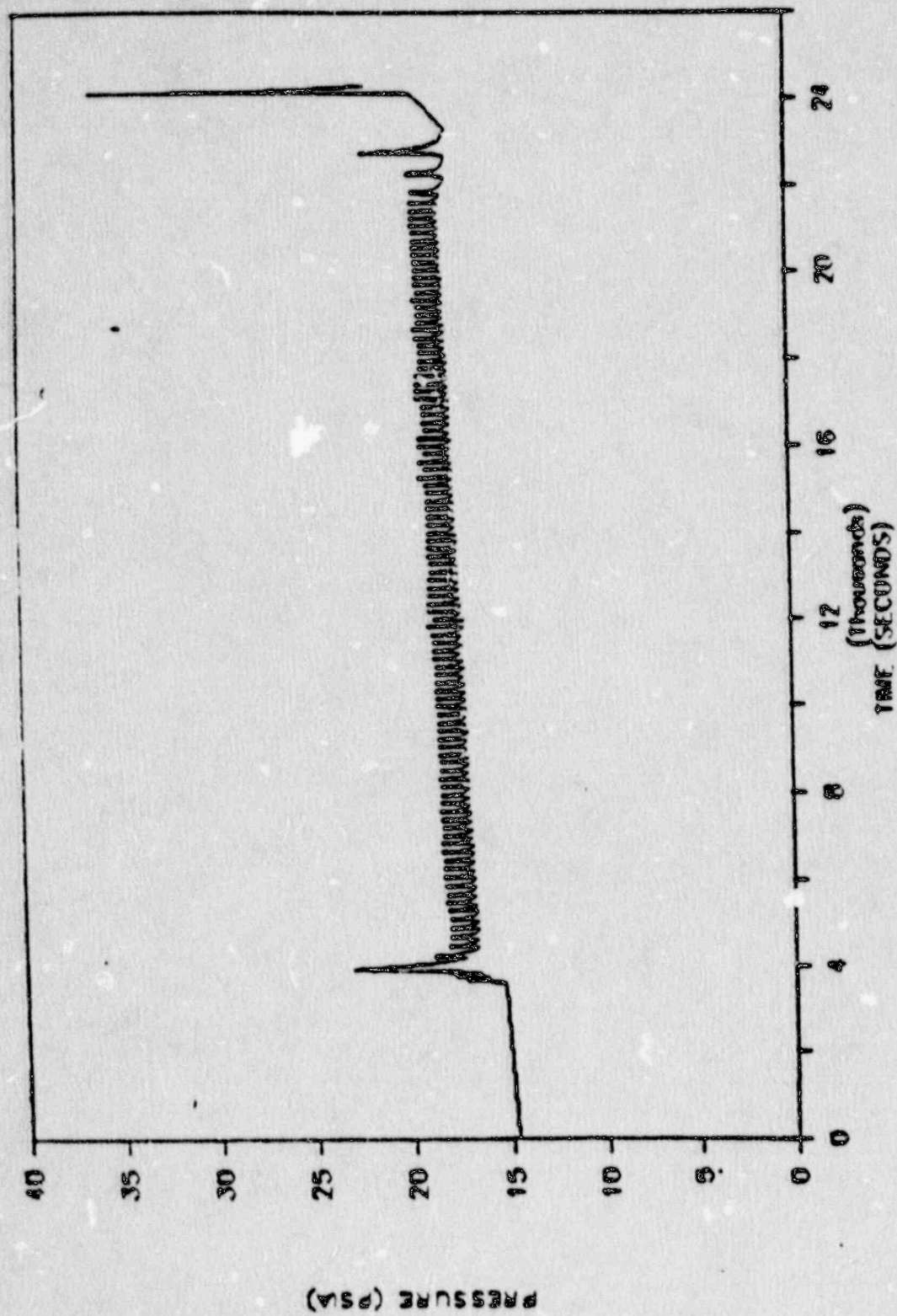


Figure 6.3 SORV with no spray-well pressure
Source: NGM-109-P

containment should confirm the location of critical equipment with respect to potential oxygen sources through the drywell vacuum breakers and a drywell mixing system to support the above conclusion for each plant.

7 SURVIVABILITY OF ESSENTIAL EQUIPMENT

As part of the analysis, 10 CFR 50.44(c)(3)(vi)(B)(5)(ii) states:

Systems and components necessary to establish and maintain safe shut-down and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen, including the effect of local detonations, unless such detonations can be shown unlikely to occur.

Accordingly, each licensee with a Mark III containment is required to demonstrate that the essential equipment located inside the containment will survive the hydrogen burn environment. To support this objective, the HCOG conducted two programs to define the environment that would result from hydrogen combustion. As discussed earlier in the evaluation, the quarter-scale test data will be used to define the environment that would be produced by diffusive combustion on the suppression pool surface. In addition, the CLASIX-3 code analysis will be used to define a bounding environment for localized combustion below the diffusion flame extinguishment limit.

7.1 Identification of Essential Equipment

The equipment that has to survive hydrogen burning was selected on the basis of function during and after a postulated degraded core accident. Generally, all the equipment located in the containment that was considered to be in one of the five categories listed below was considered to be essential for the safe shutdown of the plant.

- (1) systems and components that mitigate the consequences of the accident
- (2) systems and components needed for maintaining the integrity of the containment boundary
- (3) systems and components needed for maintaining the core in a coolable geometry.
- (4) systems and components needed for monitoring the course of the accident and providing guidance to the operator for initiating action in accordance with emergency procedure guidelines
- (5) components whose failure could preclude the ability of the above systems to fulfill their intended function

Using these criteria, the HCOG identified the equipment that would be needed to be evaluated for survivability. In its letter (HGN-084) dated May 16, 1986, the HCOG transmitted to the staff the list identifying the following system/components:

- (1) containment penetrations: air lock, hatch seals, electrical penetrations, vent valves, and vacuum breakers
- (2) drywell components: air locks, hatch seals, and post-accident vacuum breakers
- (3) hydrogen igniter system
- (4) combustible gas control system: hydrogen recombiners, drywell mixing system, and post-accident atmosphere sampling valves
- (5) containment cooling: spray isolation valves, LPCI injection valves, and unit coolers
- (6) automatic depressurization system
- (7) containment and reactor monitoring:
 - containment and drywell temperature instruments
 - reactor pressure vessel wide-range pressure instruments
 - reactor pressure vessel wide-range and fuel-zone level instruments
- (8) associated instruments, controls, cables, interlocks, terminal and junction blocks.

The staff finds that HCOG's generic equipment survivability list contains the equipment essential for the mitigation of postulated degraded core accident conditions. As part of the final analysis, each licensee with a Mark III containment should provide plant-specific information corresponding to the generic list in conjunction with unique design features that are relevant to the selection criteria.

7.2 Generic Equipment Survivability Analysis (Localized Combustion)

As discussed in Section 6, pressure and temperature predictions were obtained by using the CLASIX-3 code. This calculational methodology was used to provide the containment environment or boundary conditions necessary to perform equipment response analyses for hydrogen release rates below the diffusion flame extinguishment limit where burning is limited to localized combustion. The HCOG believes that equipment survivability can be established generically. The supporting analyses was presented in its letter dated August 7, 1987 (HGN-118-P). It was stated that a generic approach is sufficient because of the conservative nature of the combustion phenomena modeled by the CLASIX-3 code and the boundary conditions used in the generic equipment survivability analysis.

The HCOG identified a number of conservatisms in its generic analysis; the staff has listed some of the more important items below.

- (1) The constant hydrogen release rate of 0.1 lb/sec, which is the nonmechanistic "tail" portion of the release profile, is unlikely to occur after core recovery. Also, on the basis of QSTF tests, diffusion flames on the surface of the pool may exist as low as 0.02 lb/sec indicating localized

combustion would not occur. It is expected that the presence of diffusion flames would probably be the dominant combustion mode, possibly in combination with localized combustion phenomena when the hydrogen flow rate is below the flame extinguishment limit. The significance of these two different combustion modes is the spatial shifting of thermal loads; as such, a single piece of equipment would not continually be exposed to hydrogen burning resulting in a lower temperature profile.

- (2) The CLASIX-3 wetwell temperature profile was used as the boundary condition for the equipment response analysis although the most sensitive equipment is located outside of the wetwell volume. The wetwell has the severest environment of the three containment volumes. The staff finds that based on limitations of the CLASIX-3 methodology used in the generic analysis, there is no choice but to use the wetwell volume. However, the staff does recognize the selected profile is limiting.
- (3) In the selected CLASIX-3 case, there are no active containment cooling mechanisms (i.e., the lack of availability of sprays or unit coolers). Because of the type of event considered, a recoverable degraded core, the HCOG expects that sometime during these relatively long transient events, spray/unit coolers would become available.

A set of equipment common to each plant with a Mark III containment was compiled from the list of generic equipment. Subsequently, the most thermally sensitive equipment, such as cables, pressure transmitter, hydrogen igniter assembly, and ADS solenoid valve, were included in the generic survivability analysis. Based on the results of the generic equipment survivability analysis, the drywell break equipment response analysis showed favorable results; whereas, in the SORV case, the thermal analysis response for the pressure transmitter indicated a 27°F exceedance above its qualification temperature. The significance of this result is assessed below.

The equipment response analysis for the SORV case used the wetwell CLASIX-3 temperature profiles, presented in Section 6, with some modifications. These modified profiles exclude the few initial burns in which diffusion flames would exist and the last induced global burn. As a result, the modified wetwell profile contains about 90 serial hydrogen burns. The calculated critical component of the pressure transmitters exceeded its qualification after the seventy-first hydrogen burn. However, the HCOG indicated that the pressure transmitter is expected to survive the hydrogen event because of various conservatisms in the analysis.

The staff acknowledges conservatisms, as discussed earlier, are contained in these analyses which could compensate for the temperature exceedance over the qualification of the pressure transmitter. Nonetheless, the staff requested the HCOG to provide additional data on the qualification of the pressure transmitter. By letter dated April 5, 1988 (HGN-131-P), the HCOG indicated that during qualification testing the transmitter had operated without failure at surface temperatures approaching 380°F for several minutes (as compared to the qualification of 320°F). As part of the HCOG's response, an equipment response analysis of the pressure transmitter was re-evaluated assuming containment sprays to be available. This response analysis indicated that the equipment surface temperature was about 70°F less than qualification temperature of 320°F. These results

demonstrated the impact of sprays to cool the containment environment, thus maintaining the function of essential equipment. River Bend Station is the only plant without containment sprays, but unit coolers are part of its design. While specific analyses have not been performed to support quantification of the cooling effect provided by unit coolers versus sprays, the HCOG concludes that a reduction in background temperature would be adequate to reduce the thermal loads on the pressure transmitter.

In summary, HCOG contends that further analysis is unwarranted because, with the potential for active containment cooling and the conservatism inherent in the analyses, the pressure transmitter will function as designed during recoverable degraded core events that progress to 75% metal-water reaction. With regard to these analyses at low hydrogen flowrates, the staff agrees with HCOG's position that further effort in this area is not warranted. Moreover, the staff finds that the determination of equipment survivability based on the data obtained from the QSTF for diffusion flames is more appropriate than an assessment based on localized combustion conditions.

7.3 Diffusion Flame Thermal Environment Methodology

In its letter dated July 30, 1986 (HGN-103), the HCOG outlined the methodology to be used by each member licensee to determine the full-scale plant-specific containment thermal environments from the QSTF data. The full-scale environmental conditions would be used as boundary conditions in the HEATING-6 computer code to analyze the response of containment equipment during postulated diffusive combustion events. As a result of these analyses, the survivability of essential equipment would be determined.

From the production test series conducted for each Mark III containment, the test that produces the most limiting environment at the corresponding (to full scale) equipment location is used. Thermal profiles are constructed by spatial mapping of the test facility data. Specific plant profiles are developed from average temperatures for time intervals of maximum hydrogen flow and low constant hydrogen flow from the production tests. This allows determination of the plume locations and the effects of blockages and spargers.

Full-scale velocities are computed from the quarter-scale measured velocities using Froude scaling; test temperatures are used directly (scaling is 1:1). The convective heat transfer and radiative heat fluxes are computed using the scaled velocities and temperatures. Since this approach establishes an environmental map, the heat transfer modes that should be considered are dependent on the location of the affected equipment.

To validate and assess the heat transfer methodology, a complex (three-dimensional geometry) calorimeter assembly was used in several quarter-scale tests to subject the calorimeter to different locations and different thermal environments. A HEATING-6 model of the complex calorimeter was constructed and the calculated response was compared to the measured response to validate the methodology. This effort is presented in the HCOG's letter (HGN-105-P) dated August 29, 1986.

SNL assessed the submittals and determined that most of the computed results were conservative from the standpoint of the equipment survivability. Therefore, the correlations and results presented are reasonable. However, SNL recommended

that when the generic methodology is used for plant-specific equipment evaluations, a review should be conducted to assure conservative specification of the boundary conditions.

In summary, the staff finds that the heat transfer methodology dealing with diffusive combustion as presented in HGN-103, provides an acceptable foundation to perform plant-specific equipment response analyses. Accordingly, each licensee with a Mark III containment intends to use this methodology as part of its final analysis, required by the hydrogen rule. The staff agrees with SNL's recommendation, that sufficient detail of input data should be provided by each licensee to ensure that its analysis is conducted in an appropriate manner.

7.4 Spray Availability

In the preliminary evaluations of hydrogen igniter systems (e.g., see Grand Gulf SSER No. 3, NUREG-0831, July 1982) the staff allowed credit for operation of containment sprays in the analyses of the consequences of hydrogen combustion during degraded core accidents. The validity of the assumption of containment spray operability was premised on several considerations. First, in the preliminary evaluations of igniter systems the staff and HCOG focused on the SORV transient and drywell pipe break accident sequences. These accident sequences do not necessarily imply loss of the containment spray function of the RHR pumps. RHR pumps may be operable but the LPCI injection path may be interrupted or lost. Further, at the time of the preliminary evaluations the overall tone of the BWR emergency procedure guidelines (EPG) was to focus on containment integrity rather than adequacy of core cooling at an earlier point in a degraded core accident sequence.

Since the preliminary evaluations were conducted, additional information has been developed which raises questions regarding the validity of assumptions concerning availability of the RHR pumps in the containment spray mode. In contrast to the earlier focus on the SORV transient and drywell pipe break recent risk analysis indicates that SBO is a significant contributor to hydrogen generation events. For the SBO, the loss of reactor makeup is tied to the loss of pumps, including RHR pumps, in either the LPCI or containment spray mode. Thus for the SBO sequence, the RHR spray function cannot be reasonably assumed to be available until ac power is restored. Finally, the earlier emphasis in the EPG's on containment integrity vs core cooling for containment spray operation has been reversed. In Rev. 4 to the BWR EPG's, (March 1987) the sequence of steps has been modified. Use of RHR pumps in the containment spray mode, irrespective of adequate core cooling, is now directed as the last step, to control pressure rather than before the decision to vent.

For the above reasons the staff concludes the BWR Mark III owners should evaluate the containment and essential equipment response to hydrogen generation events assuming containment sprays are unavailable, consistent with SBO assumptions and the EPGs. Spray operability can be modelled but should be treated in the context of establishing margins for a variety of possible plant conditions. Similarly, assumptions regarding availability of containment coolers should be consistent with the basic premise of the SBO accident sequence.

7.5 Pressure Effects

In HGN-118-P, the HCOG indicated equipment located inside containment is qualified to a pressure loading of at least 30 psig applied externally. The CLASIX-3 predictions produced the most severe pressure rise of about 23 psig in the Mark III containment. The staff concludes that pressure is not a concern pending confirmation by each licensee of the 30 psig capability. When the hydrogen ignition system is functioning, various containment subvolumes will be randomly affected by hydrogen burning, however, a large pressure spike is not expected to occur.

7.6 Detonations

The HCOG believes that a detonation is not a credible phenomenon in the Mark III containment because (1) no rich hydrogen concentrations will accumulate inside containment since the distributed igniters will initiate combustion as the mixture reaches the lower flammability limit and effective mixing will occur and (2) there are no regions of the containment with sufficient geometrical confinement to allow for the flame acceleration necessary to yield a transition to detonation.

The staff agrees with the HCOG position. As confirmed by the quarter-scale test results, the atmospheric conditions inside the test facility was well mixed and burning at low hydrogen concentrations was prevalent. Thus, the potential for localized accumulation of significant concentrations of hydrogen is concluded to be unlikely.

8 CONCLUSION

On the basis of the above evaluation, the staff finds the HCOG topical report, "Generic Hydrogen Control Information for BWR-6 Mark III Containments," (HGN-112-NP) dated February 23, 1987, provides an acceptable basis for technical resolution of the Mark III containment degraded core hydrogen issue. Each licensee should provide a plant-specific final analysis, as required by 10 CFR 50.44(c)(3)(vii)(B), which will address the elements specified in 10 CFR 50.44(c)(3)(vi)(B). The HCOG topical report, or portions thereof, may be referenced where appropriate, taking into consideration the staff recommendation as stated in this report. The plant-specific analysis will use test data described in the topical report to confirm that the equipment necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by the hydrogen in all credible severe accident scenarios.

An element of the staff's assessment for determining the adequacy of the HIS was the determination of whether or not an alternate power supply was appropriate. An important factor in this decision process is the level of risk associated with SBO events leading to core damage. Recent risk studies reported in NUREG-1150 have shown that the overall core melt frequency for one Mark III plant (Grand Gulf Nuclear Station) is very low, i.e., $1E-6$ /year. However, a potential vulnerability for Mark III plants involves station blackout (SBO), during which the igniters would be inoperable; and this

condition appears to dominate the residual risk from severe accident in the Mark III plants. Under SBO conditions, a detonable mixture of hydrogen could develop which could be ignited upon restoration of power resulting in loss of containment integrity. On the basis of a separate evaluation of this possibility in the context of the NRC staff Containment Performance Improvement (CPI) program, the staff has recommended that the vulnerability to interruption of power to the hydrogen igniters be evaluated further on a plant-specific basis as part of the Individual Plant Examination (IPEs) of the Mark III plants. The staff has requested that the licensees consider this issue as part of the IPE in Generic Letter 88-20, Supplement 3.

With the caveat that the vulnerability to interruption of power to the hydrogen igniters should be further evaluated on a plant-specific basis as part of the IPEs of the Mark III plants, the staff finds that there is reasonable assurance that the HIS installed in the plants with Mark III containments will act to control the burning of hydrogen so that there is adequate protection against containment failure.

In summary, the staff concludes that the following key elements should be addressed in each licensee's plant-specific final analysis to resolve the degraded core hydrogen control issue:

- (1) hydrogen ignition system design, (the vulnerability to interruption of power to the hydrogen igniters should be further evaluated on a plant-specific basis as part of the Mark III plants IPEs)
- (2) confirmation of applicability to the generic effort
- (3) quarter-scale plant specific production testing results
- (4) primary containment structural survivability
 - quarter-scale testing
 - pressure capacity analyses for drywell and containment, for example, confirm previous plant-specific analyses
- (5) survivability of essential equipment
 - identify plant-specific essential equipment
 - define thermal environment from quarter-scale testing
 - perform equipment response analysis
 - confirm that redundancy exists for that equipment affected by secondary burning and drywell inverted diffusion flames
 - confirm pressure capability of equipment

- (6) licensee's position regarding the proposed HCOG emergency procedures for combustible gas control
- (7) overall conclusions relating to conformance of the hydrogen rule

APPENDIX A

GENERIC HYDROGEN IGNITION SYSTEM TECHNICAL SPECIFICATIONS

Generally, technical specifications (TS) of a particular system consist of two distinct sections: Surveillance Requirements to ensure system operability and Limiting Condition for Operation (LCO) to define the allowable operability range in conjunction with various plant actions when needed. Each of the four plants with Mark III containments have similar TS for the hydrogen ignition system (HIS). The following discussion on this subject is focused on the proposed generic HIS TS and its deviations from the current TS.

Currently, TS on igniter systems in plants with Mark III containments prescribe two types of surveillance practice. At 184-day intervals, all the igniter assemblies are energized and current/voltage measurements are performed and compared with similar measurements taken previously. If more than three igniter assemblies on either subsystem are determined to be inoperable, there is an increase of the surveillance frequency to a 92-day interval. A second part of this first surveillance requirement is the verification that inoperable igniters are not adjacent to each other, if more than one igniter on each subsystem is determined to be inoperable. This requirement is based on the staff's view regarding potential hydrogen pocketing in enclosed areas. The second type of surveillance is conducted at 18-month intervals to verify a surface temperature of at least 1700°F for each accessible igniter and verify by measurement sufficient current/voltage to develop 1700°F surface temperature for those igniter assemblies in inaccessible areas. Accordingly, the bases section of the TS indicates that inaccessible areas are defined as areas that have high-radiation levels during the entire refueling outage; such enclosures include the heat exchanger, filter demineralizer, and the pump room for the reactor water cleanup (RWCU) system.

The current LCO allows no more than 10% of the igniter assemblies inoperable per subsystem. And if one subsystem is inoperable, the action statement requires restoration to operable status, or be in the required operational condition, within 30 days (similar to the hydrogen recombiner TS).

By letter dated April 16, 1986 (HGN-070), the HCOG proposed to revise selected portions of the existing plant specific TS as outlined above. Principally, there are two significant proposed changes: (1) an increase of the allowable inoperable igniters per subsystem to about 40%, as compared to the current value of 10%, and (2) removal of the surveillance requirement in determining the location of the inoperable igniters after the requisite number of failed igniters have been attained. Also, as discussed above, the current action statement requires an allowable period of 30 days to restore the igniter subsystem to operable status. HCOG proposed to change this interval to 60 days because these events in which the HIS is required to be operable are less probable than design-basis accidents. The staff finds the proposed inoperable period increase is not based on sound engineering judgment since it relies

on being beyond the DBA but does not provide a rationale for 60 days. Thus the 30-day interval is appropriate and should be maintained.

In its letter of April 16, 1985, the HCOG provided a justification for the other significant proposed changes. Essentially, the HCOG cited the conclusions observed regarding the QSTF testing (see Section 3.1) of a particular scoping test in which about 40% of an igniter subsystem was inoperable in conjunction with the other subsystem not functioning.

The staff has concluded relative to the hydrogen aspects of the TS justifications that the HCOG had not provided sufficient justification to relax the TS to such a degree. The staff made its determination because of the inherent uncertainties such as extrapolation of the quarter-scale results to full scale, various injection rates, different safety relief valves actuating, and different combinations as to where the 40% of the inoperable igniters could be located. Therefore, the allowable value for inoperable igniters should be as low as practical; a 10% value appears to be a reasonable limit.

The second proposed change is to remove the surveillance that ensures that inoperable igniters are not adjacent. Principally, this surveillance ensures at least one operable igniter in each enclosed area and coverage of the azimuthal-positioned igniters in the open regions. Essentially, the MIS TS are intended to prevent buildup of hydrogen in subvolumes of the Mark III containment, and thereby preclude the occurrence of large volume burns.

The following considerations are highlighted as part of the HCOG's justification for proposing to remove the TS provision to determine that inoperable igniters are not adjacent:

- The HCOG evaluated the potential flow paths that could transport hydrogen in or near enclosed regions of the containment and determined that no potential hydrogen source exists. It is expected that igniters in open areas will function to preclude local hydrogen pocketing.
- Observations of the quarter-scale tests indicated that the released hydrogen will tend to mix with the surrounding atmosphere and thus reduce the potential of locally high hydrogen concentrations.
- The likelihood is low for inoperable igniters being located in such a fashion as to create a large containment subvolume that would be without igniter coverage. Igniters would tend to fail in a random manner.
- Currently, whenever at least one igniter is inoperable in each subsystem, containment entry is normally required to find the location of the failed igniter. This would subject plant personnel to various occupational safety hazards such as radiation exposures and the risks associated with the construction of scaffolding.

On the basis of these considerations provided by HCOG, the staff has determined that there is reasonable assurance that the proposed TS without the adjacent igniter provision would not have an adverse effect on the effectiveness of the igniter system.

The staff finds the generic HIS TS as documented in the HCOG letter dated April 16, 1986, to be acceptable contingent on the following changes: the 40% value of allowable inoperable igniters should be 10% in the appropriate locations of the text and the 60-day interval to restore a subsystem in the action statement should be 30 days. Each Mark III owner that intends to adopt the generic HIS technical specification must confirm that the HCOG assumptions used in the development of the TS are valid for their plant-specific configuration.

APPENDIX B

MARK III COMBUSTIBLE GAS CONTROL EMERGENCY PROCEDURE GUIDELINE

As part of the generic program, the HCOG has developed Combustible Gas Control Emergency Procedure Guideline (EPG) for plants with Mark III containments. The latest version of the guidelines with supporting appendices were sent to the NRC by letter (HGN-122-P) dated July 8, 1988. This procedure includes operator actions for the hydrogen igniter system as well as other combustible gas control systems designed in the Mark III containment, such as hydrogen recombiners and the drywell mixing system. In addition, the proposed procedure provides guidance for spray actuation and containment venting. This effort is to supplement the overall BWR Owner Group's EPG program.

Figure B.1 highlights the operator actions in dealing with hydrogen in an emergency situation. These actions for controlling hydrogen depend on a determination of hydrogen concentration in the containment and drywell as indicated by hydrogen monitors and/or analyzers that obtain gas samples from the containment and drywell. The significant trigger limit used in the EPG is when the drywell or containment hydrogen level reaches a concentration at which a global deflagration could threaten containment or drywell integrity from overpressurization, referred to as the hydrogen deflagration overpressure limit (HDOL). At this limit or when the containment hydrogen concentration cannot be determined to be below the HDOL and it cannot be determined that the igniters have been continuously operating, the HIS should not be used. The containment HDOL is a curve of hydrogen concentration versus containment pressure, whereas the drywell HDOL is a single value representing a peak hydrogen concentration. The containment HDOL is more limiting than the drywell HDOL.

The hydrogen ignition system, the hydrogen recombiners and the drywell mixing system are the key hydrogen mitigating systems. As indicated in Figure B.1, these systems are activated at appropriate trigger points to deal with a progressing hydrogen build-up. With the addition of an independent power source to the HIS, it is anticipated that for most severe accident/degraded core situations the resulting large amounts of hydrogen can be accommodated.

As part of the subject letter, HCOG had addressed staff comments which were discussed in a October 22, 1986 HCOG/NRC meeting. In this latest version of the Mark III Combustible Gas Control EPGs (Revision 3), HCOG has addressed staff concerns or provided sufficient justification for their position as discussed below.

As one of the initial steps in the EPGs, the operator is instructed to vent the suppression chamber or drywell, whenever either of the respective regions reaches the minimum detectable hydrogen concentration (0.5%), provided the offsite radioactivity release rate is expected to remain below the offsite release rate limiting condition for operation (LCO). It should be noted that this step is similar to the BWR EPGs for Mark I and Mark II combustible gas

control. The staff previously commented that venting may not be necessary solely upon hydrogen concentration above the minimum detectable level and below flammability levels; the use of recombiners is valuable and should be utilized where appropriate.

In response, since dissolved hydrogen is present in the reactor coolant system during normal operation and the EPGs are based on a symptomatic approach, it is the intent of this step to remedy a hydrogen problem during normal operation, and within the constraints of technical specification limits. HCOG believes there is sufficient guidance to preclude this action from being implemented during a genuine emergency situation. Also, HCOG had committed to modify the generic Mark III procedure at later date, if necessary, to be consistent with the BWR0G's approved combustible gas control procedure. The staff finds that the subject procedure and the HCOG approach to be acceptable.

As one of the last steps to control hydrogen accumulation during a progressively worsening situation, containment venting is directed. Venting the containment irrespective of the offsite radioactivity release rate would only be considered to restore and maintain the containment hydrogen concentration below excessive limits. Containment failure may follow if a large deflagration were to occur. Venting the containment may be the only mechanism which remains to prevent an uncontrolled and unpredictable breach of the containment. The controlled release of radioactivity to the environment is preferable to containment failure whereby, adequate core cooling might also be lost and radioactivity released with no control. This concept of venting is similar to the emergency procedures for pressure control.

Regarding the second issue, HCOG provided additional information responding to NRC comment dealing with the limited use of the drywell mixing systems. The staff views containment venting as a last resort to deal with extraordinary conditions. The use of the drywell hydrogen mixing system may delay containment venting by diluting the containment volume (at a higher concentration of hydrogen) with the drywell volume (at a lower concentration of hydrogen). HCOG cited various factors to demonstrate the drywell mixing system is not beneficial for hydrogen control inside the containment volume, which includes: dilution effects are marginal, since the containment is significantly larger than the drywell; the mixing system would re-initiate a LOCA signal and potentially interfere with event recovery; and implementing a modified procedure may induce conflicting direction. In addition, the design intent of the mixing system is to deal with hydrogen in the drywell.

The staff believes some of HCOG concerns are valid. In addition, HCOG had modified its procedures to assure that the HIS would remain operational above the HDOL if it can be determined that the igniters have been continuously operating. The addition of an independent power supply to the MIS would further enhance the reliability of the system. Consequently, the added reliability would reduce the potential for containment venting to control hydrogen inside containment. Therefore, the staff agrees with HCOG that the inclusion of the drywell mixing system would not provide significant benefits (in delaying venting) as compared to its disadvantages.

Overall, the staff finds the proposed (Revision 3) Mark III containment EPGs are based on sound technical judgment, and are acceptable. Accordingly, each Mark III licensee should address its combustible gas control emergency procedure in the plant specific final analysis.

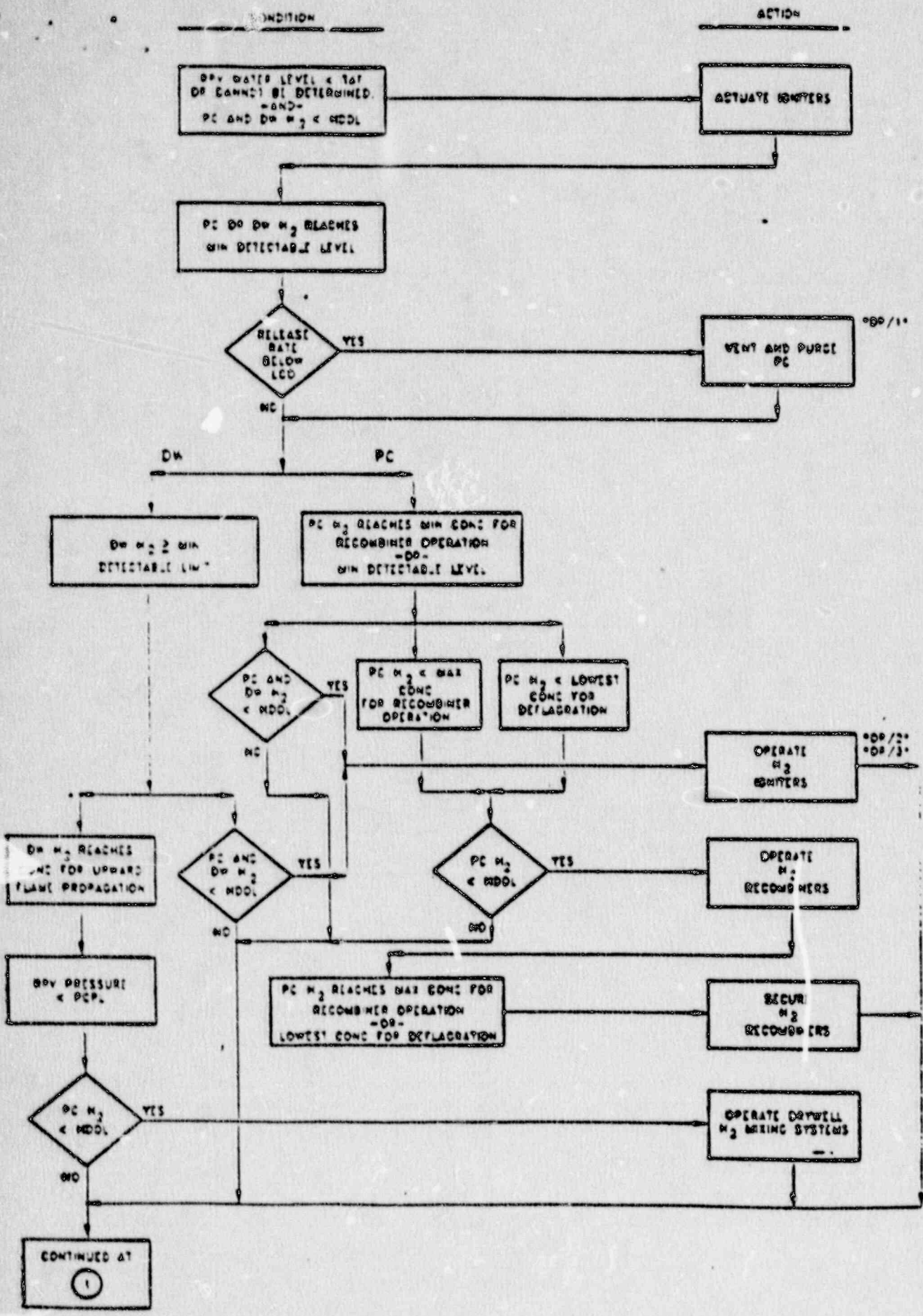


Figure B.1 Operator Actions to Control Hydrogen

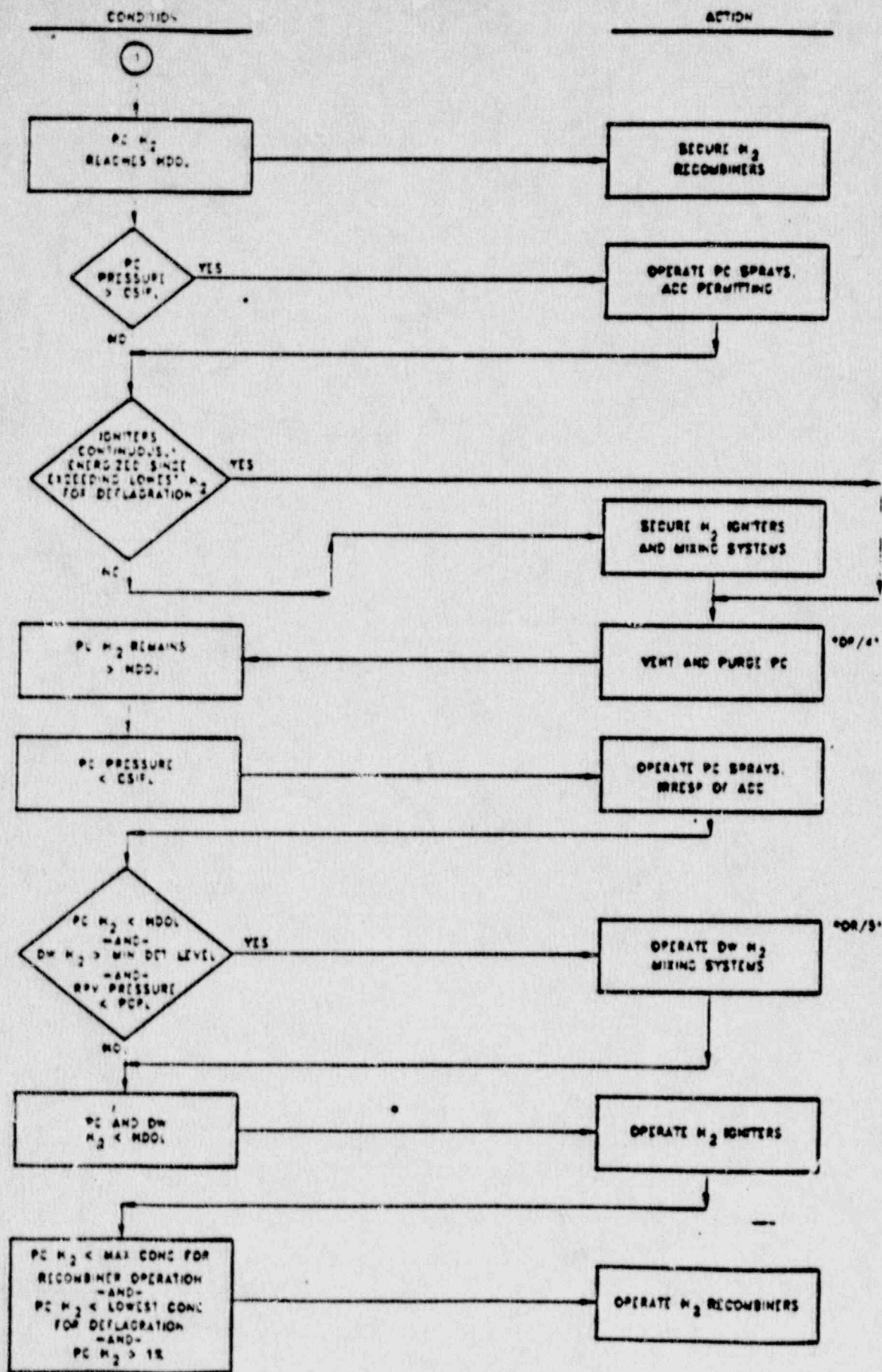


FIGURE B.1 (Continued) Operator Actions to Control Hydrogen

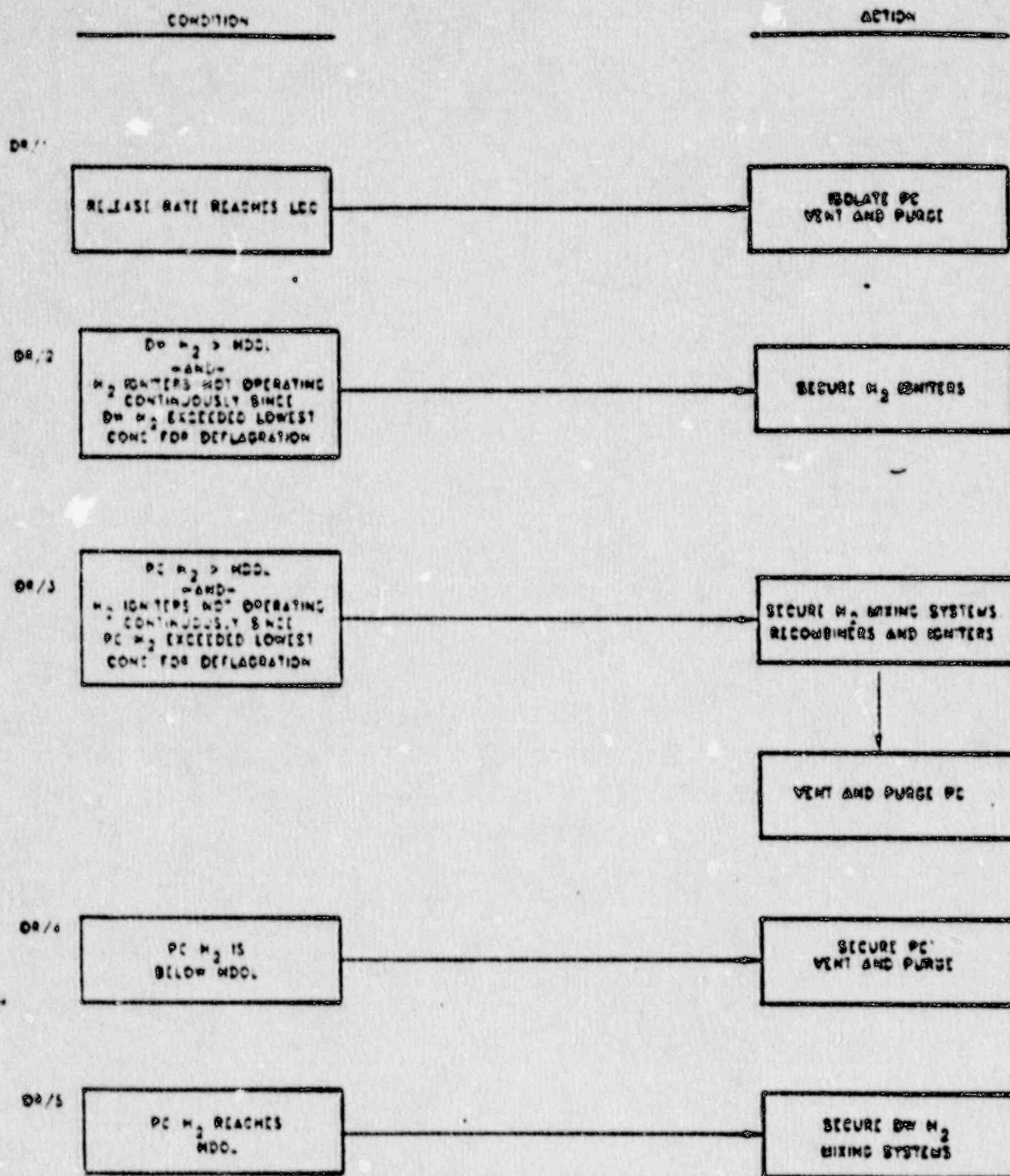


Figure B.1 (Continued) Operator Actions to Control Hydrogen

APPENDIX C
BIBLIOGRAPHY

- Gaunt, R. O. et al., "The DF-4 BWR Control Blade/Channel Box Fuel Damage Experiment," Sandia National Laboratories, to be published.
- Hydrogen Control Owners Group (HCOG), HGN-003, letter from J. D. Richardson (HCOG) to H. R. Denton (NRC), "Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements," April 8, 1982.
- , HGN-006, letter from J. D. Richardson (HCOG) to H. R. Denton (NRC), "Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements," Rev. 1, September 9, 1982.
- , HGN-009-P, letter from J. D. Richardson (HCOG) to H. Denton (NRC), "CLASIX-3 Report," March 18, 1983.
- , 011-NP, letter from J. D. Richardson (HCOG) to R. W. Houston (NRC), "Responses to NRC Requests for Additional Information," May 11, 1983.
- , 012-NP, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "1/4 Scale Hydrogen Test Program," August 12, 1983.
- , HGN-014-NP, letter from S. H. Hobbs (HCOG) to H. Denton (NRC), "Final 1/20th Scale Test Report," February 9, 1984.
- , HGN-017-NP, letter from S. H. Hobbs (HCOG) to H. Denton (NRC), "Mark III Hydrogen Control Owners Group Final Whiteshell Ignition Test Report," June 7, 1984.
- , HGN-020, letter from S. H. Hobbs (HCOG) to H. R. Denton (NRC), "Transmittal of BWR Core Heatup Code Manual," September 5, 1984.
- , HGN-027-NP, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "1/4 Scale Test Facility 3D-Complex Calorimeter," February 13, 1985.
- , HGN-031, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Hydrogen Release Histories and Test Matrix for 1/4 Scale Test Program," March 13, 1985.
- , HGN-032, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Submittal of Information on BWR Core Heatup Code," April 16, 1985.
- , HGN-034, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Model for Hydrogen Production Equivalent to 75% MWR," May 17, 1985.
- , HGN-051, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Availability of Containment Spray System," July 26, 1985.

- , HGN-052, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Hydrogen Release Time Histories," August 1, 1985.
- , HGN-055, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Evaluation of SBO and ATWS Contributions to Hydrogen Generation Events," September 27, 1985.
- , HGN-070, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Mark III Hydrogen Ignition System Technical Specifications," April 16, 1986.
- , HGN-072, letter from S. H. Hobbs (HCOG) to R. Bernero (NRC), "Event Scenarios Considered for Evaluation of Drywell Response to Degraded Core Accidents," March 5, 1986.
- , HGN-073, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Justification for Manual Actuation of Mark III Hydrogen Ignition Systems," March 5, 1986.
- , HGN-084, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Generic Equipment Survivability List," May 16, 1986.
- , HGN-085, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Final Report Assessing Adequacy of Heat Sink Modeling in the 1/4 Scale Test Facility," May 5, 1986.
- , HGN-089, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "BWR Core Heatup Code Responses," June 9, 1986.
- , HGN-091, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Criteria for Existence of Inverted Diffusion Flames in the Drywell," June 25, 1986.
- , HGN-092-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Report of CLASIX-3 Generic Analyses and Validation of CLASIX-3 Against 1/4 Scale Test Facility Data," June 10, 1986.
- , HGN-096-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "BWR Core Heatup Code Report," July 30, 1986.
- , HGN-098-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Scoping Test Report," July 18, 1986.
- , HGN-099-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Transmittal of Handouts from June 19, 1986 HCOG-NRC Meeting," July 11, 1986.
- , HGN-100, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Nevada Test Site Data Evaluation," July 31, 1986.
- , HGN-101, letter from J. R. Langley (HCOG) to H. Denton (NRC), "River Bend Station Unit Coolers," July 30, 1986.
- , HGN-103, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Diffusive Combustion Thermal Environment Methodology Definition Report," July 30, 1986.

- , HGN-104-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Evaluation of Emergency Procedure Guidelines Operator Actions Against HCOG Assumptions for Analysis of a Hydrogen Generation Event," August 18, 1986.
- , HGN-105-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Diffusive Combustion Heat Transfer Methodology Validation for Equipment Survivability in Mark III Containments," August 29, 1986.
- , HGN-106-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Supplemental Information on Secondary Burning," September 29, 1986.
- , HGN-109-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "CLASIX-3 Generic Sensitivity Analyses," December 9, 1986.
- , HGN-110-P, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Combustible Gas Control Emergency Procedure Guideline and Supporting Appendices," December 1, 1986.
- , HGN-111-P, letter from J. R. Langley (HCOG) to NRC, "CLASIX-3 Summary Report," December 15, 1987.
- , HGN-112-NP, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Generic Hydrogen Control Information for BWR-6 Mark III Containments," February 23, 1987.
- , HGN-113, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Comparison of CLASIX-3 Predictions to Nevada Test Site Data," January 8, 1987.
- , HGN-114, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "Response to NRC Concerns Regarding Potential Impact of SBO Events on Operation and Performance of the Hydrogen Ignition System," January 8, 1987.
- , HGN-115-NP, letter from J. R. Langley (HCOG) to R. Bernero (NRC), "1/4 Scale Test Facility Final Design Report," February 10, 1987.
- , HGN-118-P, letter from J. R. Langley (HCOG) to NRC, "Generic Equipment Survivability Analysis," August 7, 1987.
- , HGN-119, letter from J. R. Langley (HCOG) to NRC, "Final Inverted Diffusion Flame Report," June 10, 1988.
- , HGN-121-P, letter from Langley (HCOG) to NRC, "Report of Hydrogen Combustion Experiments in a 1/4 Scale Model of a Mark III Nuclear Reactor Containment," July 22, 1987.
- , HGN-122-P, letter from J. R. Langley (HCOG) to USHRC, "Revision 3 to Mark III Combustible Gas Control Emergency Procedure Guideline," July 8, 1988.
- , HGN-123, letter from J. R. Langley (HCOG) to USNRC, "Response to NRC Questions on Station Blackout and ATWS Sequences," September 9, 1987.
- , HGN-128, letter from J. R. Langley (HCOG) to NRC, "Responses to NRC/SNL Comments on Heat Loss Report," November 6, 1987.
- , HGN-129-P, letter, J. R. Langley (HCOG) to NRC, "Revision 4 EPGs vs HCOG Scenarios," April 8, 1988.

- , HGN-131-P, letter, J. R. Langley (HCOG) to NRC, "Supplemental Discussion of Pressure Transmitter Survivability at Low Hydrogen Release Rates," April 5, 1988.
- , HGN-132, letter, J. R. Langley (HCOG) to NRC, "Final HCOG Hydrogen Release Histories," April 4, 1988.
- , HCOG/NRC meeting presentation, "BWR Core Heatup Code Results," August 28, 1984.
- , HCOG/NRC meeting presentation, G. R. Thomas (HCOG), "Review of Basis for Zircaloy Oxidation Cutoff in the BWR Core Heatup Code," January 14, 1985.
- International Technical Services (ITS), ITS/LWR/BNL 85-1, "Validity of the Use of a Temperature Cutoff for Zircaloy Oxidation," by H. Komoriya and P. Abramson, September 1985.
- , letter from H. Komoriya (ITS) to L. Lois (NRC), "Review of Oxidation Modeling in BWR Core Heatup Code," October 30, 1987.
- Institute of Electrical and Electronics Engineers, IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."
- Mississippi Power and Light Company, letter from L. F. Dale (Mississippi Power and Light Company) to H. Denton (NRC), "H₂ Igniter Environmental Qualifications Test Results," February 14, 1983.
- Sandia National Laboratory, letters from S. E. Dingman to A. Notafrancesco (NRC), June 26, August 10, September 3, September 4, and December 23 (two letters), 1987.
- Science Application Inc., SAIC/87/3114, "Hydrogen Generating Events for Boiling Water Reactors with Mark III Containments," by Atefi et al., April 15, 1988.
- U.S. Nuclear Regulatory Commission, Title 10, Code of Federal Regulations, 10 CFR 50.44, "Hydrogen Control Requirements."
- , letter from R. Bernero (NRC) to S. H. Hobbs (HCOG) dated June 4, 1985.
- , letter from R. W. Houston (NRC) to J. R. Langley (HCOG) February 21, 1986.
- , NUREG-0011, "Safety Evaluation Report Related to the Operation of Sequoyah Nuclear Plant, Units 1 and 2," Supplement 6, December 1982.
- , NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Rev. 1, July 1981.
- , NUREG-0831, "Safety Evaluation Report Related to the Operation of the Grand Gulf Nuclear Station, Units 1 and 2," Supplement 3, July 1982, and Supplement 5, August 1984.
- , NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station, Unit No. 1," Supplement 6, July 1986.

---, NUREG-0887, "Safety Evaluation Report Related to the Operation of Perry Nuclear Power Plant, Units 1 and 2," Supplement 6, April 1985.

---, NUREG-0979, "Safety Evaluation Report Related to the Final Design Approval of the GESSAR II BWR/6 Nuclear Island Design," Supplement No. 2, November 1984.

---, NUREG-0989, "Safety Evaluation Report Related to the Operation of River Bend Station," Supplement 4, September 1985.

---, NUREG-1150, "Severe Accident Risks: an Assessment for Five U.S. Nuclear Power Plants," Second Draft, June, 1989.

---, NUREG/CR-1659, Vol. 4 of 4, "Reactor Safety Study Methodology Applications Program (RSSMAP)," Grand Gulf Power Plant Unit 1, November 1981.

---, NUREG/CR-4866, "An Assessment of Hydrogen Generation for the PBF Severe Fuel Damage Scoping and 1-1 Tests," A. J. Cronenberg et al., EG&G, April 1987.

---, NUREG/CR-5079, "Experimental Results Pertaining to the Performance of Thermal Igniters," Sandia National Laboratory, to be published.

Vinjamuri, K., et al., "Severe Fuel Damage Test 1-4 Data Report," EG&G, September 1987.

APPENDIX D
ACRONYM LIST

ADS	automatic depressurization system
BWR	boiling-water reactor
BWRCHUC	boiling-water-reactor core heatup code
DWB	small pipe break in the drywell
EPG	emergency procedure guid-line
ESF	engineered safety feature
FMRC	Factory Mutual Research Corporation
HCOG	Hydrogen Control Owners Group (Mark III Containment)
HCU	hydraulic control unit
HDDL	hydrogen deflagration overpressure limit
HGE	hydrogen-generation events
HIS	hydrogen ignition system
HPCS	high-pressure core spray
IEEE	Institute of Electrical and Electronics Engineers
LCO	limiting condition of operation
LOCA	loss-of-coolant accident
LWR	light-water reactor
MWR	metal-water reaction
NRC	Nuclear Regulatory Commission
PWR	pressurized-water reactor
QSTF	quarter-scale test facility
RCIC	reactor core isolation cooling
RPV	reactor pressure vessel
RSSMAP	reactor safety study methodology applications program
RWCU	reactor water cleanup
SAIC	Science Applications Inc. (report designation)
SDLOCA	small-break loss-of-coolant accident
SBO	station blackout
SER	safety evaluation report
SNL	Sandia National Laboratory
SORV	stuck-open relief valve
SRV	safety relief valve

TAF top of active fuel
10 CFR Title 10 of the Code of Federal Regulations
TS technical specifications
Zr zirconium