Instructions for Inserting

Revision 3 of the

Hydrogen Control Program Plan

1. Replace existing page 3-3 with page 3-3 dated 7/9/85
2. Replace existing page 4-3a with page 4-3a dated 7/9/85
3. Replace existing page 4-7 with page 4-7 dated 7/9/85
4. Replace existing page 4-8 with page 4-8 dated 7/9/85
5. Replace existing page 4-9 with page 4-9 dated 7/9/85
6. Replace existing page 4-12 with page 4-12 dated 7/9/85
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80. Replace existing page 4-166 with page 4-166 dated 7/9/85
81. Replace existing page 4-167 with page 4-167 dated 7/9/85
82. Replace original Explanation of Symbols with new explanation dated 7/9/85
83. Replace task 6 network with new network dated 7/9/85
84. Replace task 7 network with new network dated 7/9/85
85. Replace task 7 network with new network dated 7/9/85
86. Replace page 1 of Task 9 network with new network dated 7/9/85
87. Replace page 2 of Task 9 network with new network dated 7/9/85
88. Replace page 1 of Task 10 network with new network dated 7/9/85
89. Replace page 2 of Task 10 network with new network dated 7/9/85
90. Replace task 11 network with new network dated 7/9/85
91. Replace task 12 network with new network dated 7/9/85
92. Replace task 13 network with new network dated 7/9/85
93. Replace task 14 network with new network dated 7/9/85
94. Replace existing page 5-1 with page 5-1 dated 7/9/85
95. Insert page 5-2 dated 7/9/85 after page 5-1
96. Replace existing Milestone Schedule with new schedule dated 7/9/85

Factory Mutual Research Center (FMRC)

FMRC was selected as the contractor to construct and erect the 1/4 scale test facility for the diffusion flame thermal environment program. FMRC will design, construct and manage the testing effort conducted in this facility. FMRC is directly responsible to the Research Program Manager, NSAC.

Other Contractors

The Hydrogen Control Owners Group has utilized a number of other contractors to analyze or provide consulting or research services. Following is a brief description of the work performed by these contractors which is described in greater detail in Section 4.0, Task and Subtask Descriptions.

Offshore Power Systems - A subsidiary of Westinghouse which developed and modified the CLASIX analysis code.

GMF Associates - Provides general consulting services and support related to the use of CLASIX and CLASIX-3.

General Electric - Provides consulting services as requested and provided early evaluation of the hydrogen generation event.

Westinghouse - Current owner of CLASIX and CLASIX-3 code. Provided license agreement for use of CLASIX-3 after dissolution of Offshore Power Systems.

Combustion and Explosive Research, Inc. - Provided consulting services related to the behavior of hydrogen combustion.

Stone and Webster Engineering Corporation - Develops generic]3
models of equipment to be analyzed.]3

a recoverable HGE. These events are anticipated transients without scram (ATWS) and Station Blackout (SBO). Data will be evaluated to determine if these events have a significant]3
probability of resulting in recoverable HGEs.]3

used to identify the most representative accident scenario for a hydrogen generation event (HGE) and deterministic considerations were applied to define a realistic time period for termination of the HGE before significant core melt occurred. Using this methodology a spectrum of possible accident scenarios were produced. This evaluation was completed for HCOG by GE and the results included in "Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirments".

This report estimated the combined probability for all consoli-]3
dated events which could lead to core melt. The most probable]3
HGE sequence was modeled as a turbine trip with bypass and loss]3
of feedwater. Failure of all other makeup systems resulted in a
drop of RPV level due to inventory loss through the bypass
pressure control system until vessel isolation occurs. Reactor
pressure is then controlled by SRV action until the RPV is
depressurized by the operator per the emergency procedure.
Event recovery is started when ECCS injection is recovered and]3
water injection established. Variations in the timing of
operator actions to recover core makeup systems were included in
the study.

Responsibility - GE
Status - Complete

<u>1.5 Submit Report To NRC</u>

The work for Subtask 1.4 was completed in 1982 and submitted by HCOG to the NRC. The results of this work were described by "Report on Hydrogen Control Accident Scenarios, Hydrogen Generation Rates and Equipment Requirements" and submitted as an attachment to HGN-003. A revision to this report to incorporate

additional work requested by the NRC staff was submitted to the NRC as an attachment to HGN-006 dated September 9, 1982.

Responsiblity - HCOG
Status - Complete

1.6 Evaluate ATWS and SBO Accident Scenarios

The work under task 1.4 was completed in late 1982 and early 1983. Since that date, additional information within the nuclear industry has been developed on the probability of Anticipated Transients Without Scram (ATWS) and Station Blackout
(SBO) Events. The HCOG intends to submit a qualitative]3
discussion concerning the omission of ATWS and SBO's as HGE]3
initiators due to the low probability of either event]3
leading to recoverable degraded core accidents which threaten]3
containment integrity due to hydrogen combustion. This discus-]3
sion will be provided to the NRC staff for review.]3

Responsibility - HCOG]3
Status - In Progress]3

ACCEPTANCE CRITERIA FOR TASK 1
ESTABLISH MOST PROBABLE HYDROGEN GENERATION EVENT

1. Hydrogen generation events which involve amounts of hydrogen production up to the equivalent of oxidizing 75% of active fuel cladding shall be defined. Probabilistic considerations shall be used to determine the most probable initiating event. Plant systems available to mitigate the event will be consistent with assumptions regarding event progression and recovery of lost vessel inventory makeup systems. The frequency of occurrence for a variety of postulated initiating events shall be based on existing BWR/6 probabilistic data and considered in conjunction with loss of all core makeup to determine the probability of core melt. The consolidated events with the highest probability of leading to hydrogen generation without gross core melting shall be established.

2. The probability of Station Blackout and Anticipated Trans-]3
ients Without Scram leading to recoverable degraded core]3
accidents shall be evaluated. If the probability of these]3
events leading to recoverable degraded core accidents is lower]3
than the probability of other events, which could lead to]3
recoverable degraded core accidents these precursors need not be]3
considered further.]3

3. Operator actions to recover various ECCS systems and respond to plant conditions will be consistent with symptom based emergency operating procedures or guidelines. The consequences of reasonable variations in the timing for operator actions to depressurize the vessel and to recover core makeup shall be considered.

4. A recoverable core geometry shall be maintained by assuring that adequate core make-up is provided prior to the onset of

2.1 Evaluate Industry Experience

The HCOG began the process of selecting a hydrogen control system by reviewing the nuclear power industry's experience with systems designed to control large amounts of hydrogen. Three]3
nuclear power plant owners have contributed substantial information to the data available regarding hydrogen control systems. These plants were the Tennessee Valley Authority's Sequoyah Nuclear Power Plant, Duke Power Company's McGuire Nuclear Power Plant, and American Electric Power's D. C. Cooke Nuclear Power Plant.

These power plants have a Westinghouse NSSS with an ice condens-]3
er type of containment. Each plant separately committed to a distributed ignition system for burning hydrogen as it is generated and released to the containment.

The experience of these plants clarified the NRC's goals for]3
hydrogen control systems. The initial systems which were considered by Duke and TVA provide a reasonably comprehensive listing of potentially viable hydrogen control systems. Many of the criteria which were used by TVA and Duke for selecting the hydrogen ignition system are also applicable to the Mark III type containment.

Responsibility - HCOG
Status - Complete

2.2 Develop Selection Criteria

The HCOG developed a set of general system comparison criteria for the hydrogen control system. The criteria were originally developed by MP&L for the GGNS selection process and were based

TASK 3 - DESIGN HYDROGEN IGNITION SYSTEM

The detailed design of the HIS has been sufficiently developed to allow igniter installation prior to plant startup for all HCOG members. This design effort required selection of the specific igniter device, development of design criteria, establishment of igniter locations, support design for the igniter and finalization of control logic. To the greatest]3
extent practical, the HCOG has standardized the HIS design for all plants. Currently each HCOG member has common igniters, common design criteria and similar operational logic. Similar procedural guidelines will also be used by each HCOG member. Individual HCOG members will implement the design criteria in establishing igniter locations, designing supports and power supplies. This task is illustrated in Task Network 3.0.

environmental conditions enveloped all potential environmental conditions for specific HCOG member plants. The specified environment was compared to environments that were used for qualifying comparable equipment at the Sequoyah and McGuire plants. The igniters were required to withstand the effects of hydrogen combustion, submergence and operation in an all steam environment. Additional qualification analysis or testing as appropriate was then performed by each member utility to produce a plant specific qualification envelope.

Responsibility - Utility
Status - Complete

3.5 Qualify Igniters

Each member utility has conducted an environmental qualification program that verified that the selected igniter was capable of fulfilling their intended safety function. Some of the environmental qualification testing initially completed by MP&L for the Grand Gulf Nuclear Station igniters has been utilized by other HCOG members.

Responsiblity - Utility
Status - Complete

3.6 Specify Initiation and Operational Logic]3

The HCOG recommended system initiation logic, instrumentation and controls, operational requirements, and surveillance test requirements to individual utility members. The HCOG recommended that the ignition system should be manually initiated whenever there was a potential that hydrogen could be produced and released to the containment. HCOG felt that the operator would have sufficient time prior to onset of hydrogen

production to manually initiate the system. HCOG has recommended that the igniters should be actuated when the reactor pressure vessel water level reaches the top of active fuel (TAF). There should be a minimum of 10 minutes available between the time when the reactor pressure vessel water level reaches TAF and the time when significant hydrogen production commences. In addition, HCOG recognized that operation of the ignition system when hydrogen was not present would not affect the plants.

Responsiblity - HCOG
Status - Complete

3.7 Location, Redundancy and Separation Criteria

Design criteria for the igniter systems were developed generically by HCOG. Criteria for location, redundancy, and separation, were developed. Each member utility has also considered high energy pipe-whip and jet impingement in the igniter locations. The igniter location criteria also specify that hydrogen burns in the drywell, wetwell, upper containment and equipment rooms must be assumed. The igniters were spaced by each member utility to minimize the potential for hydrogen accumulation which could lead to local detonations assuming a single failure. Igniters will not be separated by more than 30 feet when all emergency safeguard feature (ESF) power supplies are operable. Operable igniters will not be separated by more than 60 feet when one ESF power supply is inoperable. Two]3
exceptions exist to these requirements. Igniters are more widely spaced in the large open regions above the refueling floor where hydrogen pocketing cannot occur and in the lower portions of drywell which are subject to flooding. Each member HIS is also designed to be tested during normal plant operation.

Responsiblity - HCOG
Status - Complete

are supplied with nominal 120 VAC power which is converted by a transformer to 12.0 V before being applied to the glow plug. Each utility has determined what failure modes are possible and considered the effects in the power supply design. In addition each member utility has added the igniter loads to appropriate ESF emergency power supplies.

Responsibililty - Utility
Status - Complete

3.11 Install Igniters

Each utility member is responsible for igniter and associated power supply and control system installation. This installation process is dependent on each plants projected startup and construction schedule. System testing and confirmation of operability will be completed prior to full power operation for each plant or as allowed by specific plant license conditions.

Responsibility - Utility
Status - In Progress

3.12 Develop Tech Specs]3

HCOG will review the plant specific Technical Specifications]3
related to hydrogen ignition systems subsequent to completion of]3
the 1/4 scale testing program, to determine if program results]3
would support a revision to the specifications. HCOG proposed]3
changes should be applicable to all member plants.]3

Responsibility - HCOG
Status - In Progress

designed to withstand the effects of the SSE and remain functional.

8. The hydrogen ignition system shall be designed and procured as Safety Class 2, Quality Group B equipment.

9. The use of A.C. power for the igniter system is acceptable provided that criterion 2 in Task 1 is satisfied with respect to station blackout being demonstrated to be sufficiently improbable.

10. Technical specification limits shall be established to assure that the hydrogen ignition system will be operable when
required. This will include requirements that sufficient]3
igniters are operable in the containment to assure ignition of]3
hydrogen at low hydrogen concentrations. Surveillance require-]3
ments shall be specified to assure that the system is operable during normal plant operation. Surveillance requirements shall also be established to assure that the igniters are capable of achieving the required surface temperature.

4.3 Burn Static or Dynamic

MP&L has completed an analysis for the GGNS which indicates that the pressure loadings produced by deflagrations occur over a period of several seconds. Since the pressure loading occurs over a relatively long period of time, the HCOG elected to use]3 this analysis as the basis for treating these loads as static loads.

Responsibility - HCOG
Status - Complete

4.4 Containment Ultimate Capacity Analysis

Each utility has completed a containment ultimate capacity analysis. This analysis was conducted using specified material properties or information from certified material test reports for the containment structure, the drywell, airlocks, and major penetrations. The capability of all local components such as airlock and hatch seals, penetrations, etc. has been evaluated to assure that these components have the capability to withstand at least the pressure determined to be limiting with respect to the overall structure. The containment ultimate capacity analyses also establish that the Mark III drywell will withstand large internal pressures. Either the ultimate capacity of the drywell structure has been established, or the drywell has been shown to have a pressure retaining capability which considerably exceeds the peak pressure which might be produced by combustion in the drywell. Individual reports for assumptions and methodology have been submitted by member utilities to the NRC]3 staff.

Responsibility - Utility
Status - Complete

4.5 Containment Negative Capacity Analysis

Each utility has conducted and completed a negative capacity
analysis to determine the maximum external pressure load on the
containment and drywell which could result from hydrogen
combustion or has documented the very large external pressure
capacity of the containment. In these cases utilities have shown]3
that the containment and drywell can withstand external pressure
loads which exceed the maximum external pressure which could
result from hydrogen combustion. Individual reports of assump-
tions and methodology have been submitted by member utilities to]3
the NRC staff.

Responsibility - Utility
Status - Complete

4.6 Consider Load Due to Local Detonation

The hydrogen igniters were located to preclude the accumulation of detonable concentrations of hydrogen in equipment rooms or other enclosed areas. The open geometry of the Mark III and turbulence further minimizes the possibility of local detonations. HCOG has concluded that local detonations will not occur in the Mark III containment. This conclusion was based upon extensive literature research completed by MP&L. This conclusion was documented in HCOG's response to an NRC request for additional information (RAI). The response to this RAI was submitted by letter HGN-011 dated May 11, 1983.

Responsibility - Utility/HCOG
Status - Complete

4.7 Document Exclusion of Local Detonation

Each utility had the responsibility to document to the NRC that]3

local detonations were not sufficiently probable to warrant consideration or that no significant containment pressure effect occurs. This included a discussion of plant specific features which preclude the accumulation of hydrogen concentrations to detonable levels.]3

Responsibility - Utility
Status - Complete

4.8 Verify Containment Capability

Each utility has verified that the peak containment pressure produced by postulated hydrogen combustion is below the containment ultimate capacity. The reports demonstrate that each element of the containment pressure boundary is capable of withstanding the peak pressures produced by hydrogen combustion. The reports also demonstrate that the drywell is capable of withstanding the peak pressure which might be produced by combustion in the drywell.

Responsibility - Utility
Status - Complete

4.9 Submit Ultimate Capacity Analysis to NRC

Each utility member has prepared an Ultimate Capacity Analysis report for the NRC. This report contained appropriate details of analytical methods, assumptions and evaluation results. The reports documented that the ultimate containment capacity exceeded the peak pressure which would be produced by hydrogen combustion.

Responsibility - Utility
Status - Complete

ACCEPTANCE CRITERIA FOR TASK 4
CONTAINMENT ULTIMATE CAPACITY ANALYSIS

1. The containment structural integrity shall be established by analyzing the ultimate internal pressure capacity. The ultimate internal pressure capacity shall be defined as that pressure where a general state of yield is reached by the limiting structural section or component. Local components such as containment air lock and hatch seals, and penetrations shall be shown to maintain their structural integrity at pressures which equal or exceed the ultimate internal pressure capacity.

2. The calculational method for determining ultimate internal pressure capability may include the use of actual material properties with suitable margins to account for uncertainties in modeling, in material properties, in construction tolerances, and so on. Another method which can be used is to demonstrate the following specific criteria of the ASME Boiler and Pressure Code are met:

   A. Steel containments shall meet the requirements of the ASME Boiler and Pressure Vessel Code Section III,
   Division 1, Subarticle N-3220, Service Level C Limits]3
   or an acceptable alternate criteria, considering]3
   pressure alone. The evaluation of instability is not required.

   B. Concrete containments shall meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, subarticle CC-3720, Factored Load Category, considering pressure and dead load alone.

3. For containments which do not have vacuum breakers to admit additional air mass into the containment volume, the containment structure shall be demonstrated to withstand an external

5.7 Submit Topical Report Verifying CLASIX to NRC

The CLASIX-3 code is an adaptation of the ice condenser containment code, CLASIX. The CLASIX code was used in the containment response analysis of the Sequoyah, McGuire and and D.C. Cooke plants. Formal verification of the base code was completed and has been submitted to the NRC in a topical report. Verification of the CLASIX-3 code, has also been completed and submitted to the NRC in the form of a topical report as described in Subtask 5.9.

Responsiblity - HCOG
Status - Complete

5.8 Determine if Additional Verification is Required

The Hydrogen Control Owners Group has reviewed the verification work completed to date and has determined that additional analysis was required. Comparison with similiar codes for]3
containment response indicates that CLASIX-3 provides reasonable results for accidents which can be analyzed with other containment response analysis codes. In addition, a number of sensitivity studies have been completed which show that the code predicts reasonable variations in program output for variations in code input. Although all verification work completed to date indicates that CLASIX-3 adequately predicts deflagration type combustion in the Mark III containment, HCOG intends to evaluate CLASIX-3 predictions against large scale data relevant to the Mark III containment geometry. This data is limited to data which will be obtained in the Mark III 1/4 Scale Test Program. The specific CLASIX-3 comparisons will be made as part of Task 12 in this program.

Responsibility - HCOG
Status - Complete

hydrogen flammability tests conducted as part of Subtask 6.17 and observations and behavior observed in the 1/20 scale flame visualization test conducted as Subtask 6.18. Results confirmed that ignition limits of hydrogen rich mixtures agree with flammability limits and that steady diffusion flames could be established above the suppression pool.

Responsibility - HCOG
Status - Complete

6.20 Prepare Final Hydrogen Rich Flammability Limit Test Report

A final test report to document the test results obtained in Subtask 6.17 was prepared by the HCOG. This report included a description of the experimental setup, method, and results. Each test condition, the data obtained and overall conclusions were included in this report. This report was prepared by the Whiteshell Nuclear Research Establishment under contract to the Electric Power Research Institute. This report was titled "Ignition Effectiveness of the GMAC 7G Glow Plug In Rich Hydrogen-Air-Steam Mixtures".

Responsibility - HCOG
Status - Complete

6.21 Submit Final Flammability Test Report to NRC

The final report prepared in Subtask 6.20 was submitted to the NRC as an attachment to HGN-017 dated June 7, 1984.

These tests confirmed that hydrogen rich mixtures of hydrogen, air, and steam can be ignited by the GMAC 7G glow plug as long

to evaluate igniter performance in a spray environment. These
tests monitored surface temperature of Tayco A.C. igniter and
GMAC model 7G glow plug igniters in a spray environment. HCOG
has received the results of the TVA tests on the Tayco igniter;]3
however, it was determined that these tests are not applicable]3
to the GM glow plug used in the BWR Mark III containment. HCOG]3
is in the process of receiving results from Sandia concerning]3
tests run for the NRC. HCOG will investigate these results and]3
determine their applicability in the hydrogen ignition system.]3
HCOG will document to the NRC the basis for concluding that the]3
TVA tests on the Tayco igniter are not applicable to the Mark]3
III containment and the results of the group's evaluation of the]3
Sandia tests.]3

Responsibility - HCOG
Status - In Progress]3

- Check of coding logic

- Check of encoded representation of models]3

Technical

- Improvements in radiant heat transfer models of core internals

- Add ability to start core heatup from any initial core water level

- Add fuel channels and control blade surfaces to metal inventory available for oxidation

Documentation

- Improve code documentation

These improvements have been completed and incorporated into the BWR Core Heatup Code. The code has also been modified to include a detailed energy balance and to include modeling of blockages produced by melting of zircaloy cladding. Additional review of the code by the NRC staff in Subtask 7.12 may result in other changes being incorporated.

Responsibility - EPRI
Status - Complete

7.7 Define Recoverable Accident Scenarios

Accident scenario assumptions which are necessary for input parameters to the BWR Core Heatup Code have been identified. The vessel was assumed to be at an initially low pressure corresponding to the pressure following operation of the Automatic Depressurization System. The core is assumed to be

3/4 uncovered and core heatup is assumed to start approximately 2000 seconds after scram. Any changes in accident sequence assumptions from Subtask 7.14 or from emergency procedure development in Subtask 13.10 which will affect the hydrogen release rates will be reflected in the final release histories.

Responsibility - HCOG
Status - Complete

7.8 Calculate Degraded Core Hydrogen Release Histories

Using the modified BWR Core Heatup Code produced by Subtask 7.6 and the accident scenario assumptions defined in Subtask 7.7, calculations of various hydrogen release rates were performed. These analyses investigated the effect of various reflood flow rates and reflood initiation time. This information was provided to Subtask 9.7 to aid in generation of a draft test matrix.

Responsibility - EPRI
Status - Complete

7.9 Complete BWR Core Heatup Code Sensitivity Study

The sensitivity of the BWR Heatup Code to input parameter
variation and modeling assumptions has been assessed and]3
documented to NRC by HCOG letters HGN-032 dated April 16, 1985]3
and HGN-043 dated June 15, 1985. The changes in hydrogen]3
release rates due to variation of reflood flow rates, reflood
initiation timing, vessel pressure, initial core water level,
core wide radiant heat transfer modeling, core nodalization,
oxidation cutoff temperature, decay energy, and fuel-clad gap]3
conductance were determined. The sensitivity of code predic-]3
tions to timing for depressurizing the reactor vessel has been]3
evaluated. The effect of varying the amount of zircaloy]3
inventory melt considered to be recoverable has been evaluated.]3

These sensitivity runs demonstrate that the controlling parame-]3
ters for peak hydrogen generation rate, duration of hydrogen]3
production, and total hydrogen production are the reflood]3
injection rate and reflood initiation timing. Resolution of]3
questions in Subtask 7.12 has been considered in the sensitivity]3
studies.]3

Responsibility - EPRI
Status - Complete]3

7.10 Submit BWR Core Heatup Code Details to NRC

The Nuclear Regulatory Commission staff requested additional details on the BWR Core Heatup Code prior to a review meeting on the use of the code. The Hydrogen Control Owners Group has submitted the "User's Manual and Details of Modeling for the BWR Heatup Code" to the Nuclear Regulatory Commission staff for review. This manual was submitted to the staff by letter HGN-020 dated September 5, 1984. The manual discusses assumptions used in the code, equations solved by the code, required input, available output, and solution schemes employed.

Responsibility - HCOG
Status - Complete

7.11 HCOG/NRC Meeting to Review Code Application

The Hydrogen Control Owners Group has reviewed the modifications to the BWR Heatup Code implemented in Subtask 7.6 and the input assumptions for recoverable accidents with the Nuclear Regulatory Commisssion staff. The results of sensitivity studies performed with the BWR Core Heatup Code version proposed for generation of final hydrogen release histories for input into the 1/4 scale testing program have also been reviewed with

the staff. In addition, this meeting permitted the Nuclear Regulatory Commission staff to identify their concerns with the BWR Core Heatup Code modeling. This meeting was held on October 3 and 4, 1984.

Responsibility - HCOG
Status - Complete

7.12 Resolve Questions on BWR Heatup Code

The Nuclear Regulatory Commission staff review of the modeling and input assumptions implemented in the BWR Heatup Code and the meeting conducted as part of Subtask 7.8 has generated questions concerning the use of the code. Questions were identified relating to steam in the bypass region of the core, decay energy effect of initial core water level, effect of the variation in fuel-clad gap conductance and oxidation cutoff temperatures.
Additional sensitivity studies were completed as part of subtask]3
7.9 to assess the effect on hydrogen production from varying all]3
of these parameters except the amount of steam present in the]3
bypass region. Separate calculations were completed to]3
demonstrate that the amount of steam predicted by the BWR Core]3
Heatup Code in the bypass region was correct. These]3
calculations were reviewed with the NRC staff in November, 1984.]3
Clarifications or changes to achieve resolution should be completed before final hydrogen release histories are selected for the 1/4 scale test facility in Subtask 7.17. Any changes or clarifications will be input to Subtasks 7.6 and 7.9 and their effect determined.

The NRC staff has employed the MARCH code to verify hydrogen]3
generation predictions from the BWR Core Heatup Code.]3
Comparisons between the MARCH code and the BWR Core Heatup Code]3

resulted in additional questions on the basis for concluding]3
that the reactor pressure vessel is depressurized before]3
hydrogen production, the fraction of hydrogen which is produced]3
by oxidizing the fuel channels, and the effect on hydrogen]3
production of pressure variations in the pressure vessel. The]3
HCOG provided responses to the first two of these items]3
by letter HGN-043 dated June 15, 1985. The response to the]3
third item was provided by letter HGN-046 dated June 28, 1985.]3
The NRC staff and HCOG agreed resolution of these questions]3
would proceed in parallel with scoping testing in the 1/4 scale]3
test program.]3

Responsibility - HCOG
Status - Complete

7.13 HCOG/NRC Meeting to Review Accident Sequences

The hydrogen release histories which will be used in the 1/4 scale test program must be specified. In order to accomplish this task, the accident sequences which will be evaluated for the containment and containment systems must be determined. This involves defining the timing for system depressurization, timing for reflood initiation, systems available to provide

reflood, and effect of plant emergency procedures. The assump-]3
tions for these accident sequences must be consistent with the conditions that lead to hydrogen production and recoverable degraded core conditions. The Hydrogen Control Owners Group reviewed the proposed accident sequences with the Nuclear Regulatory Commission Staff. The reflood flow rates which should be considered in defining the input to the 1/4 scale test program were also discussed. The meeting was held on January 30, 1985.

Responsibility - HCOG
Status - Completed

7.14 Resolve Questions on Accident Sequences

It is anticipated that the Nuclear Regulatory Commission Staff's review of the accident sequences proposed for use in the 1/4 scale test facility could generate questions regarding the assumptions and reasoning which supports the sequences. Any changes or clarifications must be complete before use of the hydrogen release histories in the 1/4 scale test program. Any effect on Subtask 7.7 will be addressed to determine if any assumptions used to determine hydrogen generation release histories are affected.

Responsibility - HCOG
Status - Complete

7.15 Calculate 75% MWR Hydrogen Release History

A hydrogen release history which results in total hydrogen]3
production equivalent to oxidizing 75% of the active core]3
zircaloy cladding has been calculated. A non-mechanistic model]3

was developed to allow prediction of hydrogen production]3
equivalent to the required 75% MWR since analyses of degraded]3
core accidents using the BWR Core Heatup code demonstrated that]3
a fully mechanistic model does not predict hydrogen generation]3
for recoverable degraded core accidents equivalent to the amount]3
mandated by the new rule. The hydrogen release history also]3
provides some representation of the hydrogen production which]3
might occur in a BWR core analogous to the hydrogen production]3
on which the rule is based. This report was submitted to NRC by]3
letter HGN-034 dated May 14, 1985. This hydrogen release]3
history will be used in the 1/4 scale test program and addition-]3
al containment response analysis to be completed under subtasks]3
8.9 and 8.12. The release history will also be used in analysis]3
of the drywell response to degraded core accidents in subtasks]3
10.8 and 10.9.]3

Responsibility - HCOG
Status - Complete]3

<u>7.16 Prepare Hydrogen Release History Report</u>

A report detailing the methodology and assumptions used to generate hydrogen release histories using the BWR Core Heatup Code will be prepared. The BWR Core Heatup Code sensitivity runs will be included in this report along with a discussion of the effect on code output from varying initial parameters.
HCOG's response to an open question identified at subtask 7.13]3
related to the basis for the irreversible oxidation cutoff used
to terminate zircaloy oxidation in a given node HCOG's response
to this question was initially documented in letter HGN-032]3
dated April 16, 1985. The HCOG committed to evaluate the]3
buildup of an oxide layer on the cladding during the oxidation
transient. As long as this oxidation layer remains thin, an

irreversible oxidation cutoff temperature of 2400°K should be]3
acceptable. HCOG's evaluation of oxide layer buildup will be included in this report. This report will be submitted to the Nuclear Regulatory Commission staff in Subtask 7.17 to document

the results of HCOG's use of the BWR Core Heatup Code.]3

Responsibility - HCOG
Status - In Progress

7.17 Submit Hydrogen Release History Report to NRC

The report prepared as part of Subtask 7.16 will be submitted to the Nuclear Regulatory Commission staff to document the assumptions and modeling used to generate the hydrogen release histories used in the 1/4 scale test program. This submittal will include a description of input parameters and assumptions used in the generic analysis. The report will provide final documentation for the sensitivity of code results to variation in input assumptions.

Responsibility - HCOG
Status - Not Started

7.18 Select Hydrogen Release Histories For Input To 1/4 Scale Testing

Based on the work completed for Subtasks 7.1 through 7.10 preliminary selection of hydrogen release rates for input to the 1/4 scale test facility has been completed. As a result of the completion of Subtasks 7.11 through 7.14, information necessary to either confirm the previously selected hydrogen release rates or to select final hydrogen release time histories is available.]3
This process will insure the individual members of the Hydrogen Control Owners Group and the Nuclear Regulatory Commission staff understand the basis for hydrogen release histories which will be used in the 1/4 scale test facility. The HCOG initially]3
determined that one hydrogen release history used in the 1/4]3

scale test facility should represent the most probable hydrogen]3
generation event. This event entails loss of core inventory,
failure of all makeup systems, core heatup, hydrogen production
and termination of the event by recovery of a large flow ECCS.
Another hydrogen release history used in the 1/4 scale facility]3
should be a release history which results in a limiting diffu-
sion flame thermal environment. This release history must be
produced by a system which would be available during a degraded
core event. The event should produce sustained diffusion]3
flames. A third hydrogen release history should be used to
validate the containment response analysis code selected under
task 5. Selection of hydrogen release histories for inclusion]3
in the 1/4 scale test program was initially documented in letter]3
HGN-031 dated March 13, 1985.]3

During a meeting between the HCOG and the NRC staff on May 22,]3
1985, the HCOG committed to modify the hydrogen release]3
histories which will be used in the 1/4 scale production testing]3
in task 9.23. The HCOG agreed to include the 75% MWR hydrogen]3
release history calculated in subtask 7.15 in the 1/4 scale test]3
program. The HCOG also agreed to modify the "A" and "B"]3
hydrogen release histories identified in HGN-031 so that the]3
total amount of zircaloy in the active core region which is]3
allowed to melt will be 50% instead of 30%.]3

By letter dated June 24, 1985 the NRC Staff proposed hydrogen]3
release histories different from those proposed by HCOG in]3
previous correspondence. The staff indicated that the following]3
release histories would be acceptable. Case "A" calculated with]3
an injection rate of 150 gpm at 3100 sec., similar to that]3
documented in run HCOG 23 in letter HGN-018, Case "B" calculated]3
with an injection rate of 5000 gpm at 50% zircaloy melt, and]3
Case "C" consisting of Case "A" with a constant 0.1 lbm/sec]3
release following the Case "A" hydrogen release until the total]3
amount of hydrogen released equaled a 75% metal water reaction.]3
HCOG will use these release histories for the 1/4 scale test]3
program.]3

Responsibility - HCOG
Status - Complete

7.19 Provide Basis For Selection To NRC

The basis for selecting final hydrogen release histories was]3
provided to the Nuclear Regulatory Commission staff before
scoping tests began in letter HGN-031 dated March 31, 1985.]3
Previous information exchanges in Subtasks 7.11 through 7.14
resolved most Nuclear Regulatory Commission staff questions and]3
this submittal was intended to confirm the Hydrogen Control]3
Owners Group plans just prior to initiation of Scoping Tests in
Subtask 9.16. The NRC subsequently provided guidance to the]3
HCOG on the hydrogen release histories to be used in testing as]3
noted in subtask 7.18. HCOG intends to utilize the guidance]3
provided by the NRC as the basis for selecting release histories]3
for use in the 1/4 scale test program.]3

Responsibility - HCOG
Status - Complete]3

7.20 Resolve Questions On Selection Basis

The Nuclear Regulatory Commission staff review of the final]3
hydrogen release histories selected for inclusion in the 1/4]3
scale test program generated some additional questions. The]3
questions on the hydrogen release histories were clarified]3
during a meeting between HCOG and the NRC staff on May 22, 1985.]3
During this meeting HCOG indicated that scoping testing would]3
procede with hydrogen release history "A" identified in letter]3
HGN-013 dated March 13, 1985. HCOG reevaluated the hydrogen]3
release histories to be used in production testing and is]3
modifying the production testing hydrogen release histories as]3
identified in subtask 7.18. This has assured that the Nuclear]3

Regulatory Commission staff agrees on the accident sequences]3
considered, and hydrogen release histories before production]3
testing is started.

Responsibility - HCOG
Status - Complete

assumed to remain depressurized throughout the subsequent core heatup, hydrogen production, recovery of vessel injection, and quenching of the core. The operator will be assumed to devote all of his efforts to recovering vessel makeup once he has recognized that inadequate core cooling conditions exist. Variations in the time for completion of operator efforts to depressurize the vessel and to recover core makeup flow shall be analyzed to determine the effect on hydrogen production.

3. The code utilized to predict hydrogen generation shall provide a reasonable representation of degraded core behavior. The codc shall model the oxidation rate as a function of zircaloy temperature and provide for oxidation termination at high temperatures. A conservatively high value of 2400 K as a]3
maximum shall be used for this irreversible oxidation cutoff. Justification for use of the irreversible oxidation cutoff shall be provided by evaluating the accumulation of an oxide layer on the cladding surface. If a small oxidation layer is formed on the cladding surface, the 2400 K oxidation cutoff shall be]3
valid. The code shall not be required to accurately model core geometry deformations which occur during core heatup including molten zircaloy relocation.

4. Mechanistic Hydrogen generation histories evaluated for input to the 1/4 scale test program shall be limited to hydrogen generation histories produced by recoverable degraded core accidents. In order for an accident to be recoverable, the core must be maintained in a coolable geometry. Once 50% of the]3
total zircaloy inventory in the active fuel region exceeds the melting temperature of zircaloy, i.e., 2170 K, a coolable core]3
geometry can no longer be assured.

5. The sensitivity of code predictions of hydrogen generation histories for variations in significant code input parameters

8.1 Establish Preliminary Burn Parameters

Preliminary burn parameters for the generic analyses were established based on the work performed for the Sequoyah and McGuire nuclear plants and based on the results of phenomena testing conducted by Fenwal and Singleton Laboratories. These preliminary burn parameters included values for hydrogen concentration necessary for combustion, hydrogen concentration necessary to propagate burns, fraction of hydrogen consumed, oxygen concentrations required to support combustion, oxygen concentration required for ignition, flame velocity and burn times.

Responsibility - HCOG
Status - Complete

8.2 Establish Generic Mark III Base Cases

The generic Mark III base cases have used plant specific input parameters from the Grand Gulf Nuclear Station. Two cases were identified for analysis. The first case involved an unspecified initiation transient which resulted in SRV actuation with the 3
failure of one SRV to close. The second case involved a small break LOCA which introduced a portion of the total hydrogen produced into the drywell.

Responsibility - HCOG
Status - Complete

8.3 Perform Generic Mark III Base Case Analysis

The two base cases identified in Subtask 8.2 have been analyzed with CLASIX-3 and documented in a report submitted to the NRC in Subtask 8.6. The resulting generic transient responses have

that CLASIX-3 provides consistent results.

Responsibility - OPS
Status - Complete

8.6 Submit CLASIX-3 Report to NRC

A report summarizing the sensitivity studies completed with the CLASIX-3 computer code was prepared. This report detailed the input to the CLASIX-3 computer code for each case and contained the output from each sensitivity run including pressures, temperatures, and gas concentrations for the drywell, wetwell and containment. General conclusions regarding the effect of individual parameter changes were identified. This report was submitted to the NRC by HCOG letter HGN-001 dated January 2, 1982.

Responsibility - HCOG
Status - Complete

8.7 Resolve Questions on Sensitivity Studies

The Nuclear Regulatory Commission staff has requested additional information to document the use of CLASIX-3 and the selection of hydrogen burn parameters used in the base case analyses submitted by the Hydrogen Control Owners Group. The Nuclear Regulatory Commission staff Requests for Additional Information (RAIs) were initially answered by the Hydrogen Control Owners]3
Group's submittal HGN-011 on May 11, 1983. Additional questions exist on the effect of several parameters on temperature predicted by CLASIX-3 in the wetwell. Parameters of concern include spray carryover fractions, assumed beam lengths, heat transfer effectiveness of sheet flow, use of mass mean spray droplet size and characteristic length used to determine burn time. Questions on the heat transfer methods used in CLASIX-3,

the effects of sensible heat loss from the RPV, and the effects
of bypass leakage which were submitted to each utility member of
HCOG are being addressed generically by HCOG. In addition,
testing results from Subtask 6.21 have provided information to]3
answer portions of the NRC concerns.

Responsibility - HCOG
Status - In Progress

8.8 Determine If Additional Generic Analysis Is Required

In response to Nuclear Regulatory Commission staff comments,
CLASIX-3 has been modified to include NUREG-0588 heat transfer
correlations as detailed in Subtask 10.1. Also the Nuclear
Regulatory Commission staff has questioned other parameters and
assumptions as delineated in subtask 8.7. Consequently, HCOG]3
has decided to perform additional generic analyses. The results]3
from new analyses from Subtask 10.9 may also provide results
which affect previous work. Spray initiation timing and the
effect of other operator actions evaluated in Subtask 13.10 as
part of the emergency procedures may also necessitate completion
of additional generic analyses. In addition, the HCOG has]3
agreed to review deflagration data from burns conducted at the
Nevada Test Site as part of Subtask 14.5. These tests may
provide a basis for further revising some of the assumptions in]3
the CLASIX-3 analyses. A hydrogen release history which results]3
in total hydrogen production equivalent to 75% MWR was developed]3
in subtask 7.15.]3

Responsibility - HCOG
Status - Complete]3

8.9 Define Additional Analyses

If additional analyses are determined to be required at decision
point 8.8, then the extent of these analyses will be identified.

Sensitivity runs which exercise the parameter(s) to be investigated will also be identified. For any new analyses which are completed, the hydrogen and steam source terms used in the CLASIX-3 analyses will correspond to the hydrogen release history calculated at subtask 7.15. Any additional analysis which is completed will be performed with the version of the CLASIX-3 code that has been modified to include NUREG-0588 heat transfer methods. The results of any new analyses will be input into Subtask 11.8 to generate the containment deflagration thermal environment.

Responsibility - HCOG
Status - In Progress]3

8.10 Plant Specific Runs Required

Each utility has determined that a plant specific hydrogen]3
deflagration analysis is required. This will involve making a comparison of plant specific systems, containment geometries, release rates and other plant specific technical assumptions
with those used in the generic runs. As each utility has]3
determined a plant specific analysis is required, this work will]3
be completed in Subtask 8.12.]3

Responsibility - Utility
Status - Complete]3

8.11 Establish Common CLASIX-3 Assumptions

HCOG will prepare a list of common assumptions for use by
utilities completing plant specific hydrogen deflagration]3
analyses. This will insure that parameters which have been investigated by HCOG are treated in a conservative and consistent manner in analyses performed by the individual utilities.

Each utility will justify any plant unique assumptions in the report prepared in Subtask 8.13.

Responsibility - HCOG
Status - In Progress]3

8.12 Complete Plant Specific Runs

Each HCOG member utility will specify plant unique data such as]3
containment geometry, hydrogen release rates as calculated in]3
subtask 7.15, heat sinks, and use of sprays or containment]3
coolers for input into the modified deflagration analysis code. The resulting containment pressure/ temperature response will be used in the determination of plant equipment survivability. These analysis results will be included in the report written for Subtask 8.13. Any additional analysis using CLASIX-3 as the deflagration analysis code will be performed with the version of the CLASIX-3 computer code that has been modified to include NUREG-0588 heat transfer methods.

Responsibility - Utility
Status - In Progress]3

8.13 Prepare Utility Report

Analytical results from the work completed as part of Subtask 8.12 will be documented along with generic and plant specific assumptions in a report. This report will also justify any utility's decision from Subtask 8.11 that additional plant specific deflagration analyses runs are not required.

Responsibility - Utility
Status - Not Started

anticipated the Nuclear Regulatory Commission staff would request additional information on the facility's goals, design and use. Several Requests for Additional Information (RAI) were received by HCOG on December 8, 1983 and responses were provided in letter HGN-016 dated April 2, 1984. In addition, other questions relating to the facility have been addressed to the Hydrogen Control Owners Group during the meeting identified in
Subtask 9.4. The HCOG agreed to assess the spray carryover]3
fraction to the wetwell. HCOG intends to measure the total mass]3
flow of water reaching the wetwell and determine the mass flow]3
of spray reaching the wetwell in the form of droplets. These]3
two values will then be used to determine the spray carryover]3
fraction.]3

The development of responses to the Nuclear Regulatory Commission staff has had impact on the facility goals, design and use. Therefore, feedback from this activity to program elements 9.1 and 9.2 is shown.

Responsibility - HCOG
Status - In Progress

9.6 Construct Test Facility

The 1/4 scale test facility is being constructed for HCOG by the Electric Power Research Institute (EPRI) and Factory Mutual Research Corporation (FMRC). Initial construction was started in August, 1983 and was completed in January, 1985. The 1/4 Scale Test Facility is located at FMRC's remote test site in West Gloucester, Rhode Island. Planning and construction control was provided by FMRC with EPRI retaining overall management and budget authority.

Responsibility - EPRI/FMRC
Status - Complete

9.7 Draft Test Matrix

The 1/4 scale testing will be divided into shakedown, scoping and production tests. The shakedown tests are intended to verify proper operation of all systems and instruments in the test facility. Scoping tests will evaluate the effect of a number of parameters which might affect the production tests. Parameters which will be evaluated during scoping tests include test repeatability, the effect of simultaneous hydrogen and steam discharge, the effect of release through both the LOCA vents and the SRV spargers, effect of the presence of grating near the suppression pool surface, the effect of containment cooling system differences, the effect of geometry differences between two HCOG member plants, and the effect of changing the blockage fraction in the facility. The production tests will be used to define the full scale thermal environment produced by diffusion flames in the wetwell. The production tests will include variation on the location of the sparger which is assumed to be open, a test without containment sprays in operation and two tests with a hydrogen release rate below the threshold for steady diffusion flames. Production tests will be completed for at least three containment geometries. The final production test matrix will be dependent upon the outcome of the scoping tests. The hydrogen release rates used for the various tests are provided by Subtask 7.8.

This draft test matrix is complete and has been discussed with the Nuclear Regulatory Commission Staff. However, it is

anticipated that resolution of Nuclear Regulatory Commission questions in Subtask 9.9 may affect the final test matrix.

Responsibility - HCOG
Status - Complete

9.8 Submit Test Matrix To NRC

The draft test matrix was submitted to the Nuclear Regulatory Commission by HGN-018 dated July 6, 1984. The purpose of this submittal was to provide the Nuclear Regulatory Commission staff with the technical basis for the draft test matrix. This matrix was reviewed in a meeting between HCOG and the Nuclear Regulatory Commission staff on August 28 and 29, 1984. It is anticipated that questions on the draft test matrix will be identified by the NRC staff. Resolution of questions will be completed as part of Subtask 9.9.

Responsiblity - HCOG
Status - Complete

9.9 Resolve Questions on Test Matrix

Resolution of Nuclear Regulatory Commission staff questions on the draft test matrix assumptions and methodology should be]3 complete before the beginning of scoping tests in the 1/4 scale facility. This is to insure agreement on the important parameters which should be established by the various scoping and production tests. This will establish test conditions which will yield realistic diffusion flame thermal environments. Resolution of Nuclear Regulatory Commission staff questions will provide input to Subtask 9.7. In addition, input from Subtask

13.10 may provide effects from the EPG review which will affect the draft test matrix.

Responsibility - HCOG
Status - In Progress

9.10 Develop Test Procedures

Test procedures which define the logical progression of each]3
test and the use of the 1/4 scale test facility have been]3
generated. Revision 4 of these procedures was issued on April]3
19, 1985 by FMRC. These procedures are now being implemented]3
and will be revised as required during the testing program.]3

Responsiblity - EPRI/FMRC
Status - Complete]3

9.11 HCOG/NRC Final Facility Walkdown

After construction is essentially complete, the Hydrogen Control Owners Group offered the Nuclear Regulatory Commission staff the opportunity to conduct a final facility walkdown and review instrument locations. This walkdown and review allowed the utility sponsors and the Nuclear Regulatory Commission staff to inspect the finished facility to assure it is designed and constructed as previously described. Also the facility's ability to exercise various parameters (such as hydrogen release rate) was reviewed to assure the test matrix can be implemented. The detailed instrumentation location was reviewed. The meeting was held on December 18, 1984.

Responsibility - HCOG
Status - Complete

9.12 Complete Shakedown Testing

After construction of the test facility was completed, shakedown]3
testing was performed to initially calibrate, test, and place]3
various mechanical, electrical, and instrumentation systems in]3
working order. The final shakedown test was performed on May 6,]3
1985. The results indicate that the facility does respond to]3
the test sequence as designed and that instrumentation is]3
functional. Integrated tests have been completed and assure]3
that all of the facility systems including containment sprays,]3
instruments, hydrogen injection, and steam injection systems]3
function in an integrated manner. No significant facility]3
changes were necessary as a result of this testing.]3

Responsibility - EPRI/FMRC
Status - Complete]3

9.13 Identify Any Facility Changes to NRC

HCOG has completed the shakedown testing of the facility as]3
described in subtask 9.12. It was determined that systems and]3
instrumentation function as required, therefore no substantial]3
modifications to the test facility are required. This will also]3
was reported to NRC via letter HGN-038 dated June 17, 1985.]3

Responsibility - HCOG
Status - Completed]3

9.14 Prepare Final Design Report

To document the 1/4 scale facility design (which has been]3
modified since the submittal of the draft design report), a]3
final 1/4 Scale Test Facility design report will be prepared. Significant changes in the facility geometry, containment cooling systems modeled and instrumentation will be incorporated

into the final design report. The role of a complex geometry]3
calorimeter provided by the Hydrogen Control Owners Group will]3
also be discussed in the final document. This activity will be
completed concurrently with other Subtasks. The start of
scoping or production tests will not be delayed for this]3
activity.

Responsibilty - HCOG
Status - Not Started

<u>9.15 Submit Final Design Report to NRC Staff</u>

The report documenting the design and configuration of the facility as used for testing will be submitted by the Hydrogen Control Owners Group to the Nuclear Regulatory Commission staff. The Nuclear Regulatory Commisssion staff has been apprised of all facility changes by various HCOG/NRC meetings. This Subtask will provide a reference for facility design features.

Responsibility - HCOG
Status - Not Started

<u>9.16 Finalize Scoping Test Matrix</u>

The scoping test matrix for the 1/4 scale test facility has been]3
finalized to include 14 separate tests. The parameters to be]3
addressed by the tests are discussed below in subtask 9.17. The]3
hydrogen release history to be used for the tests was also]3
finalized. Both were transmitted to NRC by letter HGN-031 dated]3
March 13, 1985. The NRC staff has identified questions on]3
HCOG's proposed final scoping test matrix. In addition, as a]3
result of testing completed to date, HCOG has modified the]3
proposed final scoping test matrix. A revised final scoping]3
test matri.. will be provided to the NRC.]3

Responsibility - HCOG
Status - In Progress]3

9.17 Complete Scoping Tests

There are currently 14 scoping tests planned to confirm the effect of important parameters which affect the definition of the diffusion flame thermal environment. The following parameters will be addressed by scoping tests:]3

- Data repeatability
- Threshold for establishing continuous diffusion flames
- Effect of concurrent steam and hydrogen injection
- Simultaneous discharge through LOCA vents and spargers
- Effect of grating near suppression pool surface or the combustion transients
- Effect of changing the blockage fraction in the facility
- GGNS & PNPP geometry similarities

A complete set of test data will be recorded for each scoping test.]3

]3

Responsibility - FMRC
Status - In Progress]3

9.18 Analyze Scoping Test Results

The HCOG has made some specific assumptions for the proposed production test matrix. These assumptions are:

- The effects of hydrogen released through the spargers is limiting compared to releasing hydrogen through both the LOCA vents and spargers

- Effects of injecting steam and hydrogen simultaneously is negligible when compared to releasing hydrogen alone

- The Perry Nuclear Power Plant configuration and Grand Gulf Nuclear Station configuration are similar enough so that data obtained for GGNS is applicable to both plants

- No hydrogen pocketing will occur in the facility.

The results from the scoping tests will initially be analyzed to determine if these assumptions are correct. If necessary, changes in the production test matrix may be proposed for input into Subtask 9.19.

Responsibility - EPRI/FMRC
Status - In Progress]3

9.19 Production Test Matrix Acceptable?

A comparison of planned production tests and the scoping tests will be completed to determine if any modifications to the production test matrix are necessary based on scoping test results. The assumptions identified in Task 9.18 will be reviewed against scoping test results. A late input from Subtask 9.25 after formal submittal of scoping test results may

test progress and to discuss problems encountered during testing. Information concerning the nature of combustion will be provided including indications of the severity of thermal environment measured during the tests.

Responsibility - HCOG
Status - Not Scheduled

9.23 Complete Production Test

After the production test matrix is finalized in Subtask 9.20, production testing will be authorized to proceed. Fifteen production tests are planned. These tests will be conducted for]3
three different facility geometries reflecting the containment]3
arrangements of Grand Gulf Nuclear Station, Clinton Power Station, and River Bend Station. The tests for River Bend Station will include containment cooler effects. The following parameters will be addressed by production tests:

- Variation in containment geometry

- Variation in location of assumed stuck open SRV

- Absence of containment sprays

- Hydrogen release below the threshold for steady diffusion flames

Data from the first series of tests in the Grand Gulf Nuclear Station geometry will be used as early input to Subtask 11.7 to obtain a definition of the diffusion flame thermal environment.]3
Data from this Subtask will also be used in Subtask 12.6, 12.8,

12.9,12.10, and 12.15 to aid in validation of the analytical]3
methods used in the survivability analysis program.]3

Responsibility - EPRI/FMRC
Status - Not Started

9.24 Submit Preliminary Scoping Test Results

The scoping test data will be organized and correlated into a scoping test report and submitted to the Nuclear Regulatory Commission. This will serve to document the Hydrogen Control Owners Group's conclusions regarding the parameters investigated during scoping tests. Any changes to the production test matrix to reflect the observed behavior of the facility will also be addressed. Final scoping test data evaluations will be included in the Final Test Report prepared under Subtask 9.28.

Responsibility - HCOG
Status - Not Started

9.25 Resolve Questions on Scoping Test Results

It is anticipated that the Nuclear Regulatory Commission staff's detailed review of the scoping test data report may generate questions which could affect the production test matrix. Feedback to Subtask 9.19 to reflect any late changes in the production test matrix is considered a part of this task element. The Hydrogen Control Owners Group will also generate responses to any Nuclear Regulatory Commission staff questions regarding the scoping tests as part of this activity.

Responsibility - HCOG
Status - Not Started

9.31 <u>Verify Adequacy of 1/4 Scale of Heat Loss Modeling</u>]3

The HCOG is evaluating the theoretical comparative heat losses]3
between the test facility and the full scale Mark III plants. A]3
study is being completed to estimate the heat losses from the]3
1/4 scale test facility to heat sinks in the facility and to the]3
facility structures versus heat losses to full scale plant]3
equipment and heat sinks. This study will assure that the gas]3
temperatures measured in the 1/4 scale facility will be conserv-]3
ative or comparable to the temperature which would be produced]3
in a full scale plant. The intent of the study is to show that]3
the Froude modeled 1/4 scale test facility provides an accept-]3
able representation of temperatures in a full scale facility.]3
This study will be provided to the NRC staff for review.]3

Responsibility - FMRC/EPRI]3
Status - In Progress]3

transfer to the containment sprays. The sprays shall be demonstrated to produce bulk atmosphere mixing patterns which are representative of the bulk atmosphere mixing patterns expected at full scale.

The total heat transfer characteristics of the scaled test facility shall be conservative with respect to the heat transfer characteristics of all full scale Mark III containment or representative of the actual heat transfer characteristics. It shall be demonstrated that the heat losses to the gratings, walls and other heat sinks in the scaled test facility shall not exceed the heat losses to gratings, walls, equipment and other heat sinks in the full scale facility.

The scaled test facility must have the capability to simulate variable hydrogen and steam flow into the facility. The
hydrogen flow rate into the facility must be variable over the 3
range of expected hydrogen production rates. The facility shall have the capability to simulate steam flows associated with the hydrogen release history which produces the limiting diffusion flame thermal environment throughout the hydrogen generation event. The hydrogen shall be injected into the facility at locations which correspond to the points of hydrogen release in the full scale plant.

4. The repeatability of the test data shall be evaluated. A set of tests with comparable initial conditions and identical geometry shall be completed. Acceptable repeatability for the test data will be determined by comparing the gas temperatures, velocities, and radiant heat fluxes.

scaled test facility. For each parameter which is evaluated, the test including that parameter shall be completed identically with tests which are conducted to assess repeatability. Each parameter may be judged to have no effect on the combustion transient. If the temperatures, gas velocities and radiant heat flux measured in the test facility for all instruments at the HCU floor level are within the data scatter defined during the repeatability tests, or these parameters are within 15% of the mean value established during repeatability testing, the parameter will be assumed to have no effect. If the parameters are shown to have a significant effect, these parameters shall
be addressed in the testing. The parameters varied in the]3
scoping tests will be evaluated to determine their significance]3
when considering a cumulative or synergistic effect with other]3
test-evaluated parameters.]3

7. The thermal environment including gas temperature convective heat flux and radiative heat flux shall be defined for all areas in the wetwell and containment where equipment required to survive hydrogen combustion is located. A limiting thermal environment in each area of the containment shall be established. The possible release points for hydrogen through
stuck open safety relief valves, open ADS valves, and release]3
through LOCA vents shall be considered in establishing the limiting thermal environment. The effect of having two open relief valves under a hot chimney, the effect of a single open relief valve below HCU floor grating, and the effect of simultaneous hydrogen release through the SRV spargers and LOCA vents shall be evaluated in establishing the limiting thermal environment.

8. Sufficient data on gas distribution throughout the wetwell and containment shall be obtained from tests with and without operation of containment sprays to evaluate the effectiveness of

hydrogen mixing with the containment atmosphere. A gas sampling system shall be included in the facility which is capable of measuring hydrogen, oxygen and water vapor concentrations at a number of different locations in the wetwell and containment. The mixing data will be evaluated to demonstrate that mixing of the containment gases prevents significant accumulation of hydrogen. Hydrogen concentrations shall be shown to not exceed 8 v/o at elevations above the first row of igniters and that detonable mixtures do not exist.]3

9. Data shall be obtained from the test facility to validate analytical methods which are used by HCOG. An instrumented calorimeter with complex geometry shall be included in the facility. This calorimeter shall be used to validate the heat transfer methods used in calculating the temperature response of the component.

Data shall also be obtained to validate the methodology used to define the thermal environment for evaluating equipment survivability. Convective heat flux shall be determined and compared with values calculated using measured gas velocities and temperatures. Radiative heat flux shall be measured and compared with values calculated using standard text book methods.

10.1 Modify CLASIX-3 To Utilize NUREG-0588 Assumptions

The Nuclear Regulatory Commission staff review of CLASIX-3
included comparisons with analyses of the drywell compartment
run with the CONTEMPT-LT containment response code. These
comparisons highlighted that CLASIX-3 was written to predict
conservative containment pressure and realistic temperature
responses following non-design basis degraded core accidents.
The CONTEMPT-LT code incorporates conservative heat transfer
methods for predicting temperatures corresponding to the
methodology described in NUREG-0588. To provide a more
conservative prediction of the thermal environment prior to]3
hydrogen ignition, an option to use heat transfer models based
on NUREG-0588, Branch Technical Position CSB 6-1 and the
CONTEMPT-LT program was included in the CLASIX-3 code. The use
of this modified code may affect the decision for Subtask 8.8.

Responsibility - OPS
Status - Complete

10.2 Define Accident Sequences

Preliminary results from Task 7.0, Generation of Hydrogen
Release Histories, indicates that hydrogen production rates are
relatively independent of the conditions that lead to hydrogen
production. However, specific drywell break sequences lead to]3
variations in estimates of the drywell thermal environment
before hydrogen production begins. These sequences include a]3
definition of break size, primary system heat loads, operator
actions, and available plant equipment which will affect the
drywell thermal environment and mass balance before hydrogen
production begins. HCOG has defined the accident sequences]3
which should be evaluated to define the drywell's response to]3
degraded core accidents. These sequences will be reported in]3
task 10.27.]3

Responsibility - HCOG
Status - Complete]3

10.3 Select Approach For Defining Blowdown History

There is a large body of existing analyses for drywell break cases which could be used as input into the CLASIX-3 code for the drywell blowdown history prior to depressurization of the
primary system. It has been determined that degraded accident]3
sequences are not properly modeled by existing analyses;]3
consequently, a new blowdown history will be calculated in]3
Subtasks 10.5 and 10.6 using the MAAP code.]3

Responsibility - HCOG
Status - Complete]3

10.4 Use Existing Analysis

As existing drywell break analyses are not appropriate for]3
defining the drywell blowdown history, a new history will be]3
determined by use of a different code. Consequently tasks 10.5]3
and 10.6 will be performed in lieu of task 10.4.]3

Responsibility - HCOG
Status - Complete (superseded by tasks 10.5 and 10.6)]3

10.5 Select Code For Blowdown Analysis

Since a new analysis to predict drywell blowdown under degraded]3
core conditions was determined to be required from Subtask 10.3]3
a survey of appropriate codes was conducted and the MAAP code]3
was selected to perform the required analyses. The MAAP code]3
has been used in the IDCOR program to calculate responses of the]3
plant to degraded core accidents. The hydrogen production]3
module in MAAP has been benchmarked against the BWR Core Heatup]3
Code. This approach should limit questions on the selection and]3
use of this code. MAAP will be used at Subtask 10.6 to conduct]3
predictions of blowdown to the drywell under degraded core]3
accident sequence assumptions prior to depressurization of the]3
primary system.]3

Responsibility - HCOG
Status - Complete]3

10.6 Complete Blowdown Analysis

Using realistic drywell initial conditions, the selected break]3
sizes, and accident sequences leading to degraded core condi-]3
tions, a drywell blowdown analysis up to the point at which the
vessel is depressurized will be completed using the MAAP code]3
selected in Subtask 10.5. This analysis will define the time
history of break flow into the drywell including mass added to
the drywell, energy added to the drywell, and temperature
response until the reactor vessel is depressurized by the
operator.

Responsibility - HCOG
Status - In Progress]3

10.7 Define ADS Timing

Flow through a postulated drywell break will continue until the
reactor is depressurized by the operator using the ADS. The
time that ADS actuation will occur will depend on the operator's
response to action limits in the emergency procedures based on
drywell temperature, reactor vessel level, and suppression pool]3
heat capacity, among other parameters. The operator is expected]3
to maximize the time available before hydrogen production using
steam cooling. Before hydrogen production commences, the vessel
will be fully depressurized. This establishes the point where
hydrogen and steam production from the degraded core can be
predicted by the BWR Core Heatup Code. The hydrogen and steam
flow will be split between the open ADS SRV's and the drywell
break after the vessel is depressurized using the ADS.

Responsibility - HCOG
Status - In Progress]3

10.8 Calculate Drywell Break-SRV Flow Split

The division of steam and hydrogen between the drywell break and
the open SRV's will be affected by the drywell pressure history
after ADS, the number of ADS valves open, suppression pool
level, the reactor vessel pressure, the break size, and the]3
timing of ADS. Based on these factors, a realistic flow split
of hydrogen and steam will be determined. Using the steam and
hydrogen release history generated as part of subtask 7.15, a
blowdown history of steam and hydrogen through the break into
the drywell will be calculated. This blowdown history of steam
and hydrogen and the pre-ADS steam blowdown history from 10.6]3
will be used to define the steam and hydrogen release history to
the drywell from the time of break until core recovery is

completed. This composite release history will be used as input to analyze the drywell thermal response in Subtask 10.9.

Responsibility - HCOG
Status - In Progress]3

10.9 Analyze Drywell/Containment Pressure Using CLASIX-3

The CLASIX-3 program is a multi-compartment containment response analysis code which will predict temperature, pressures and concentrations of gases. To represent a drywell break, a time history of the steam, hydrogen and fission product energy along with associated enthalpies must be input into the code. This is provided by Subtasks 7.15 and 10.8 for this analysis. Emergency procedure actions determined from Subtask 13.10 will also be used for this analysis. To determine if deflagrations in the drywell are possible, the CLASIX-3 code or other acceptable deflagration analysis code will be run for the drywell and]3
containment using the blowdown history from Task 10.8 and]3
appropriate combustion parameters for the drywell. The analysis will account for the effects of drywell bypass leakage on the wetwell and upper containment response. If deflagrations occur then the effect of this drywell pressure spike will be determined. This analysis will also allow the Hydrogen Control Owners Group to determine if inverted diffusion flames can be established at the exit of the CGCS compressor or from other oxygen sources in the drywell. The results from this analysis will define the environment which equipment in the drywell must survive if the Hydrogen Control Owners Group concludes that inverted diffusion flames do not occur in the drywell. Thus, this Subtask provides input to Subtask 11.6 in the equipment survivability analysis program. The resolution of Nuclear Regulatory Commission staff questions from Subtask 10.11 concerning this analysis will be assessed to determine their impact on this Subtask. Since previous CLASIX-3 runs have not

incorporated drywell bypass on the wetwell and upper containment response, information from this analysis will be considered in Subtask 8.8 to assist in determining if additional generic analysis is required.

Responsibility - HCOG
Status - Not Started

10.10 HCOG/NRC Meeting To Discuss Response Analysis Results

Results from the drywell analysis completed in Subtask 10.9 will be discussed with the Nuclear Regulatory Commission staff. The assumptions and methodology to analyze the drywell environment will be described along with the CLASIX-3 analysis results. Conclusions regarding conditions in the drywell will also be described excluding the potential for inverted diffusion flames.

Responsibility - HCOG
Status - Not Scheduled

10.11 Resolve Questions on Response Analysis Results

It is anticipated that Nuclear Regulatory Commission staff review of the drywell response to degraded core conditions may generate questions and requests for additional information concerning assumptions and modeling used in the analysis completed for Subtask 10.9. This Subtask will provide needed resolution for any Nuclear Regulatory Commission staff questions.

Responsibility - HCOG
Status - Not Started

10.12 Define Criteria For Existence Of Inverted Diffusion Flames

The existence of inverted diffusion flames will be investigated by determining the appropriate conditions required to support this phenomena. This will include required hydrogen concentration in the drywell, oxygen content of the entering gas, effect of combustion products, local mixing, location and geometry of air inlet sources to the drywell, water vapor effects. A search of existing combustion literature for applicable test data defining the parameters needed to support inverted diffusion flames will also be conducted. This information will be used to define a conservative set of conditions and criteria which must be present in order for inverted diffusion flames to exist. This criteria will be used by Subtask 10.13 to determine if degraded core accident conditions in the drywell can support this type of hydrogen combustion.

Responsibility - HCOG
Status - In Progress]3

10.13 Determine If Inverted Diffusion Flames Can Occur

Using the drywell environment produced by Subtask 10.9 and the criteria defined in Subtask 10.12, it will be determined if drywell conditions support existence of inverted diffusion]3
flames at the point(s) of air entry into the drywell. As part of this task, a survey to identify all systems which can introduce air into the drywell or hydrogen into the containment other than through the LOCA vents or SRV spargers will be conducted. The air entry points into the drywell or drywell exits will be described and assessed for the ability to produce continuous flames. If continuous diffusion flames are determined not to exist, then this information will be used in]3

Subtask 10.18. If the criteria for continuous diffusion flames are satisfied, then the effect on the drywell thermal environment will be determined in Subtasks 10.14 through 10.17.

Responsibility - HCOG
Status - Not Started

10.14 Choose Approach To Evaluate Thermal Environment

If the results of Subtask 10.13 indicate that inverted diffusion]3
flames can exist in the drywell, then the effect of this type of combustion on the drywell thermal environment will be assessed. This will be accomplished by determining the total heat input into the drywell, defining the cone of influence from the flame, and determining what equipment is affected by radiant heat transfer. One approach could be to conduct a search of appropriate test data in existing combustion literature which will provide sufficient data to analyze this condition. Another approach could be to design an appropriate test which will yield]3
the necessary data to determine the characteristics and effect of an inverted diffusion flame in the drywell. One of these approaches will be selected by the Hydrogen Control Owners Group.

Responsibility - HCOG
Status - Not Started

10.15 Define Analytical Approach

A literature search of existing combustion data for inverted diffusion flames where an oxidant rich mixture is introduced into a fuel rich, oxidant absent atmosphere will be conducted. Description of flame characteristics, temperature, cone of influence, and radiation characteristics will be investigated.

Using this data and the predicted drywell conditions from Subtask 10.9, a heat transfer model of the inverted diffusion flame will be constructed. This model will be used in conjunction with CLASIX-3 to predict the behavior of the drywell after ignition of the inverted diffusion flame occurs. The effect on this Subtask from resolution of Nuclear Regulatory Commission staff questions in Subtask 10.20 will be assessed.

Responsibility - HCOG
Status - Not Started

10.16 Complete Analysis Program

Using the inverted flame model, the drywell pressure/temperature
response will be defined from the point in time of ignition of
the inverted flame until conditions supporting inverted flames
no longer exist. This information will be used to define the]3
drywell thermal environment for input into Subtask 11.6.

Responsibility - HCOG
Status - Not Started

10.17 Define Test Program

If insufficient data exist to adequately define the behavior of]3
an inverted diffusion flame, then consideration will be given to
conducting an appropriate test. This test will gather suffic-
ient information on flame characteristics, flame temperature,
radiant heat flux, cone of influence, thermal plume dimension]3
and other factors necessary to determine the effect of inverted
diffusion flames on the drywell thermal environment. Any

planned tests will adequately account for plant unique design features. This information will be used in Subtask 11.6 for input into the equipment survivability analysis program.

Responsibility - HCOG
Status - Not Started

10.18 Document Conclusions To NRC

If inverted diffusion flames cannot occur, then the Hydrogen Control Owners Group will provide justification for this conclusion. The technical basis for this conclusion will be summarized and submitted to the Nuclear Regulatory Commission staff for review.

Responsibility - HCOG
Status - Not Started

10.19 HCOG/NRC Meeting To Discuss Inverted Flame Program

The Hydrogen Control Owners Group will meet with the Nuclear Regulatory Commission staff to discuss the status of the inverted diffusion flame program. At this point the Owners]3
should have sufficient information to determine if inverted diffusion flames can occur in the drywell or if diffusion flames can occur elsewhere in the containment. Also, sufficient information should be available to determine if an analytical approach or scaled testing will be used to determine the effect of the inverted flame on the drywell environment.

Responsibility - HCOG
Status - Not Scheduled

10.20 Resolve Questions On Inverted Flame Program

After the meeting with the Nuclear Regulatory Commission staff to review the inverted diffusion flame program results in Subtask 10.19, it is anticipated the staff may make requests for additional information on assumptions and methodology used. This subtask will provide any needed responses or question resolution to the staff. Any changes in analysis methodology will be reflected in Subtask 10.15. This information will be used by the member utilities of the Hydrogen Control Owners Group in individual utility reports to be submitted to the Nuclear Regulatory Commission staff as part of subtask 10.27.

Responsibility - HCOG/Utility
Status - Not Scheduled

10.21 Evaluate Potential For Pool Swell Loading From Hydrogen Combustion

If the HCOG determines that inverted diffusion flames cannot occur in the drywell, then deflagrations in the drywell may]3
occur and force excess hydrogen from the drywell into the wetwell where a large supply of oxygen is present. An immediate and large deflagration in the wetwell could produce a large containment to drywell differential pressure. This pressure may result in forcing a jet of water from the area between the drywell wall and weir wall upward into the drywell. With sufficient velocity, this jet of water may produce loads on structures or affect safety related equipment above the weir area. The possibility for occurrence of this negative pool swell will be investigated.

In addition, drywell deflagrations may produce high pressure transient conditions in the drywell similar to LOCA pressure

increases. If these pressure increases are larger than the drywell pressure increase from a DBA LOCA, then pool swell loads produced by these deflagrations will be evaluated to assure that essential structures survive. If inverted diffusion flames occur in the drywell, it is expected that hydrogen will be burned to below the point necessary for a deflagration to occur,]3
thus precluding large pressure differences. Otherwise, it will be demonstrated that drywell deflagrations and possible containment to drywell differential pressure differences produce loads no greater than currently predicted for design basis accident (DBA) conditions using faulted stress allowables.

Responsibility - HCOG
Status - Not Started

10.22 Analyze Structures and Supports

If the containment to drywell differential pressure predicted by Subtask 10.21 produces reverse vent flow or loads in excess of those previously predicted for DBA conditions, then the specific utilities will determine if any drywell structures or equipment should be reanalyzed. This decision will be based upon a comparison of the predicted differential pressure from Subtask 10.21 and the subsequent reverse vent flow peak velocity and the value used for reverse vent flow peak velocity for DBA conditions. If a significant increase exists then this new value will be used to determine new load profiles above the weir wall.

Responsibility - Utility
Status - Not Started

10.23 Analyze Structures and Supports

A survey of drywell structures, supports and equipment which

might be impacted by a reverse pool swell water jet will be conducted. Equipment which is required to survive these accidents will be analyzed as necessary for the increased load. The effect on this effort due to the resolution of any Nuclear Regulatory staff questions in Subtask 10.26 will be assessed.

Responsibility - Utility
Status - Not Started

10.24 Demonstrate Acceptable Loading

Equipment analyzed for the increased loading in Subtask 10.23 will be shown to maintain its function. Any equipment projected to fail or not to perform its function will be addressed on an]3
individual basis.

Responsibility - Utility
Status - Not Started

10.25 Document Conclusions On Differential Pressure

If the results of Subtask 10.22 indicate that the differential pressure between the containment and drywell produces negative]3
pool swell loads less than current design loads, then no further analysis will be conducted. The analysis, assumptions and methodology will be documented to the Nuclear Regulatory Commission staff for plant specific structures by each utility.

Responsibility - Utility
Status - Not Started

10.26 Resolve Questions On Analysis

It is anticipated the Nuclear Regulatory Commission staff review

of the negative pool swell analysis submitted in Subtask 4.24 could generate questions concerning the assumptions and methodology used. This Subtask will provide for any required]3 responses and question resolution. Any changes in methodology will be reflected in Subtasks 10.15 and 10.23.

Responsibility - Utility
Status - Not Started

<u>10.27 Prepare Final Drywell Response Report</u>

The results of Subtasks 10.9, 10.11, 10.13, 10.16 and 10.21 thru 10.26 will be included in a final report prepared by each]3 utility and will document the results of the drywell response to degraded core conditions. This report will include the assumptions used to analyze the drywell response for degraded core accident sequences, justification of pre-ADS blowdown to the drywell, ADS timing justification, methodology for drywell break/SRV flow split, inverted diffusion flame criteria, any inverted diffusion flame testing results, and plant specific results from the negative pool swell evaluation.

Responsibility - HCOG/Utility
Status - Not Started

<u>10.28 Submit Final Report To NRC</u>

The final report documenting the results from the various Subtasks to define the drywell response to degraded core accidents will be submitted to the Nuclear Regulatory Commission staff. Due to planned interactions with the staff, this report

existing experimental data and analytical techniques or from a suitable test. Drywell essential equipment exposed to a potential inverted diffusion flame environment will be shown to meet the acceptance criteria of Task 11.

6. The pool swell transient shall be defined based upon expected combustion in the drywell. Drywell and containment structures and components shall be evaluated to determine that pool swell does not impose structure, equipment or support loadings greater than previously analyzed. This may be accomplished by demonstrating that pool swell loads do not exist or that pool swell loads are enveloped by the present design loads, or that essential structures and components survive the pool swell event. The LOCA design basis drywell to containment pressure differential will be compared to the differential pressure transient produced by hydrogen combustion. No pool swell loadings will be evaluated if the drywell to containment differential pressure for a design basis event exceeds the hydrogen combustion differential pressure for the length of the transient.

to the operator for initiating actions in accordance with the Emergency Procedures Guidelines

(5) Components whose failure could preclude the ability of the above systems to fulfill their intended function

The generic equipment list has been given to each Hydrogen Control Owners Group member utility for guidance in developing plant specific equipment lists.

Responsibility - HCOG
Status - Complete

11.3 Select Survivability Analysis Code

A computer code capable of analyzing the thermal response of a piece of equipment subjected to transient heat fluxes due to thermal radiation, convective heat transfer and conductive heat
transfer has been selected. The code is capable of solving]3
complex geometries in various coordinate systems. The ability to input the variable heat flux at component boundaries due to
the dynamic thermal environment was also required. HEATING-6]3
was selected by HCOG to perform the required analyses.]3

Responsibility - HCOG
Status - Complete]3

11.4 Establish Components To Be Analyzed

Plant specific lists of equipment required to survive these]3
transients have been established based on the criteria]3
identified in subtask 11.1, and have been reviewed to identify]3
common components. In addition, the physical geometry of]3
similar components was reviewed to determine if a single heat]3
transfer model can be used to represent a variety of similar]3
components. Finally, if two similar pieces of equipment were]3

included on the survivability lists of HCOG utilities, the more]3
thermally limiting piece of equipment was modeled to represent]3
both pieces of equipment. The result of this task was a]3
substantially smaller list of heat transfer models required to]3
adequately represent equipment in HCOG member plants which must]3
survive these transients.]3

Responsibility - SWEC]3
Status - Complete]3

<u>11.5 Develop Models Of Equipment To Be Analyzed</u>

Using the modeling instructions and format specified for the
survivability analysis code selected in Subtask 11.3, generic]3
geometric models of equipment identified in Subtask 11.4 will be
produced. This will insure a consistent approach by all
utilities and decrease the amount of plant specific work
required.

Responsibility - SWEC]3
Status - In Progress]3

<u>11.6 Establish Drywell Thermal Profile For Analysis</u>

Using the data provided by Task 10.0, Evaluation of Drywell
Response To Degraded Core Accidents, a time history of the local
thermal environment due to convective and radiative heat
transfer will be defined for each piece of drywell equipment to
be evaluated. This will be accomplished by using the predicted
thermal profiles from the CLASIX-3 runs completed as part of
Subtask 10.9 and/or any effects from inverted diffusion flames
from either Subtask 10.16 or 10.17. This information will be]3
used in Subtask 11.11 to predict the thermal response of
equipment and components exposed to drywell conditions.

Responsibility - HCOG
Status - Not Started

profiles. The thermal response for each piece of equipment will be predicted. Generic analysis will be completed by the Hydrogen Control Owners Group to the greatest extent possible. The methodology for generic analysis will be discussed with the Nuclear Regulatory Commission staff in Subtask 11.9. Questions resulting from this presentation and resolved in Subtask 11.10 will be assessed for possible impact on this Subtask. In addition, resolution of questions on the validation process in Subtask 12.12 will act as input to this task.

Responsibility - HCOG/Utility
Status - Not Started

<u>11.12 Peak Pressure Exceeds Equipment Qualification Pressure</u>

The peak pressure in the containment or drywell, as appropriate, established in Subtasks 8.13 and 10.27 will be compared against the pressure that each piece of essential equipment identified in Subtask 11.2 is qualified to withstand by its environmental]3
qualification documentation. If the peak pressure is less than the environment qualification pressure then the equipment is acceptable and this will be documented in the plant specific equipment survivability report. If the peak pressure exceeds the pressure it was previously qualified for, it will be further evaluated in Subtask 11.13.

Responsibility - HCOG/Utility
Status - Not Started

<u>11.13 Document Equipment Survival</u>

The equipment design will be reviewed for sensitivity to pressure transients if the environmental qualification pressure is exceeded by the peak pressure expected from hydrogen

equipment to a point with a milder local environment, replacement of the equipment with equipment qualified to an acceptable temperature, active cooling of the critical component or equipment, or protection of the component with insulation or structures resistant to high temperatures.

Responsibility - HCOG/Utility
Status - In Progress]3

11.18 Specify Method Of Assuring Equipment Survivability

Based on the potential survivability enhancements identified in Task 11.17, the utility will make a decision based on the economics of the various methods of assuring survival. If shielding or other protection enhancements are utilized, then the equipment model developed in Subtask 11.5 or by the utility will be modified and the equipment will be reanalyzed to demonstrate that acceptable protection has been provided. The method of protection will be documented in the utility report on equipment survivability prepared in Subtask 11.21.

Responsibility - Utility
Status - Not Started

11.19 HCOG/NRC Meeting On Survivability Analysis

The Hydrogen Control Owners Group will meet with the Nuclear Regulatory Commission staff to discuss the results from the survivability analysis program. The evaluation results for sample pieces of equipment, the equipment response and the conclusions drawn regarding survivability will be reviewed. Representative equipment enhancements utilized to assure

12.1 Develop CLASIX-3 Model of 1/4 Scale Test Facility

A specific CLASIX-3 input case using the 1/4 scale test facility
will be developed. The modeling of the 1/4 scale test facility
may be considered an extension of the code verification
completed under subtask 5.8. The specific treatment of
geometry, compartment volumes, heat sinks, spray flow, spray
carryover, intercompartment connections and other features of]3
the 1/4 test facility must be determined in order to obtain an
accurate model. This information will be used in Subtask 12.2
to specify an appropriate input case.

Responsibility - GMF]3
Status - In Progress]3

12.2 Specify CLASIX-3 Input

The steam and hydrogen flows, compartment initial conditions, burn parameters, flow path parameters, spray system parameters, heat sinks, and suppression pool level, must be determined to define an input case for CLASIX-3. Where appropriate, the same assumptions used in previous containment response analyses will be used in the 1/4 scale test predictions. For example, hydrogen combustion in a compartment will be initiated when bulk compartment hydrogen concentration reaches 8% and 85% of the hydrogen in the compartment will be assumed to burn. At least one analysis will also be completed with best estimate combustion parameters such as assuming combustion is initiated when compartment hydrogen concentration reaches 6% with 65% of the hydrogen being burned. The spray carryover fraction used in the CLASIX-3 analysis will be measured in the 1/4 scale test.

Responsibility - HCOG
Status - In Progress]3

12.3 Complete CLASIX - 3 Prediction

Using the CLASIX-3 model of the 1/4 scale test facility developed in Subtask 12.1 and the input data file from Subtask 12.2 a test prediction of the 1/4 scale test facility response will be made. This run will predict the containment gas temperatures, constituent gas concentrations, and containment pressure response for the deflagration event in the 1/4 scale test facility.

Responsibility - GMF]3
Status - Not Started

12.4 Design Complex Calorimeter

A complex calorimeter has been designed and installed in the]3
test facility. The complex calorimeter represents different]3
types of equipment geometries such as rectangular and cylindrical components. It also has several different materials with a]3
variety of coupled thermal masses. This device is sufficently instrumented to measure its response to the convective and radiant heat flux present in the facility. The ability to move the calorimeter to various locations in the 1/4 scale test facility has been provided in order to measure the change in]3
total heat flux as the distance from hot gas plumes is increased. Design details on the complex calorimeter, its location in the facility, and instrumentation near the calorimeter were supplied to the NRC staff in letter HGN-027 dated February 13, 1985.

Responsibility - EPRI
Status - Complete

12.5 Prepare Model of Complex Calorimeter

Using the modeling instructions and format specified by the survivability analysis code established at Subtask 11.3, a heat
transfer model of the complex calorimeter was prepared. This]3
model will be used in Subtask 12.7 and 12.9 along with the appropriate thermal environments to predict the response of the
complex calorimeter. The details of the model will be reported]3
in Subtask 12.13.]3

Responsibility - HCOG
Status - Complete]3

12.6 Compare CLASIX-3 Predicted Results with Measured Results

The CLASIX-3 predictions of temperatures, pressures and gas concentrations in the wetwell and upper containment shall be compared with measured test results. The amount of combustion which is predicted to occur in each compartment will be compared with the combustion observed in test data and on video tapes. An attempt will be made to relate temperatures measured in the facility locally to global temperatures predicted by CLASIX-3.

Responsibility - HCOG
Status - Not Started

12.7 Apply CLASIX-3 Temperatures to Complex Calorimeter Model

Using the mathematical model of the complex calorimeter prepared in Subtask 12.5 and the thermal response code selected in Subtask 11.3, a prediction of the complex calorimeter behavior will be made. The deflagration environment predicted for the 1/4 scale facility by CLASIX-3 will be used to make this prediction. This information will be used in Task 12.8 to

demonstrate conservatism in the use of CLASIX-3 output and thermal response models to predict the peak temperatures in equipment and components.

Responsibility - HCOG
Status - Not Started

12.8 Compare Measured Results with CLASIX-3/HEATING-6 Predictions]3

The data from Subtask 9.23 which defines the measured deflagra-]3
tion response of the complex calorimeter in the 1/4 scale test facility will be compared with the predicted response of the complex calorimeter model produced by Subtask 12.7. If the predicted response is conservative compared to the measured response of the complex calorimeter, then the analytical methodology previously used to evaluate equipment survivability for deflagrations will have been shown to be conservative. This result will be documented in Subtask 12.13. If the predicted response does not envelope the measured response of the complex calorimeter, then a review of deflagration modeling assumptions]3
and techniques will be conducted to determine what revisions are necessary. Changes in modeling assumptions or methodology will be documented in Subtask 12.13. This information will be considered in the equipment survivability analysis completed at Subtask 11.11.

Responsibility - HCOG
Status - Not Started

12.9 Apply Measured Diffusion Flame Environments to Complex Calorimeter Model]3

Using the data from Subtask 9.23, the diffusion flame thermal

environments in the vicinity of the complex calorimeter will be]3
determined. These thermal environments will be used as input to]3
the thermal response model of the complex calorimeter prepared
in Subtask 12.5. This will yield a prediction of the response
of the complex calorimeter in the diffusion flame environment]3
of the 1/4 scale test facility. This information will be used
in Subtasks 12.5 and 12.10 to validate the heat transfer]3
modeling assumptions and techniques.

Responsibility - HCOG
Status - In Progress]3

<u>12.10 Compare Measured Results With Thermal Response Predictions</u>

The data from Subtask 9.23 which defines the measured responses]3
of the complex calorimeter in the 1/4 scale test facility to]3
diffusion flames will be compared with the predicted responses]3
of the complex calorimeter model produced by the Subtask 12.9.]3
If the predicted responses from the model are conservative]3
compared to the measured responses of the complex calorimeter,]3
then validation of the techniques and assumptions used to]3
construct the model and the assumptions used in defining the]3
thermal environments will be achieved. These results will be]3
documented in the report prepared at Subtask 12.13. If the]3
predicted responses do not envelope the measured responses of]3
the complex calorimeter then a review of modeling assumptions]3
and techniques will be conducted to determine what revisions are]3
necessary. Changes in modeling assumptions or methodology will]3
be documented in the report prepared as part of Subtask 12.13.]3
This information will be used in the equipment survivability]3
analysis conducted at Subtask 11.11.]3

Responsibility - HCOG
Status - In Progress]3

not anticipated that any further questions will be identified following submittal of this report.

Responsibility - HCOG
Status - Not Started

12.15 Apply Measured Deflagration Thermal Environment to Complex Calorimeter Model]3

Using data from Subtask 9.23 the deflagration thermal environment in the vicinity of the complex calorimeter will be determined. This thermal environment will be used as input to the thermal response model of the complex calorimeter prepared in Subtask 12.5. This will yield a prediction of the response of the complex calorimeter in the deflagration thermal environ-ment of the 1/4 scale test facility. This will assure that the heat transfer modeling of components can provide reasonable predictions of a component's response to deflagration thermal environments.]3

Responsibility - HCOG]3
Status - Not Started]3

D. Minimum oxygen volume percent for ignition 5 v/o

E. Minimum oxygen volume percent to support combustion 0 v/o

F. Flame speed 6 ft/sec

Heat removal from the 1/4 scale facility shall be consistent with the methodology used for full scale containment analysis. The containment spray carryover fraction in the facility shall be determined.

A CLASIX-3 prediction shall be completed using the same assumptions as used in previous licensing analysis. Specifically, combustion shall be initiated when hydrogen concentration reaches 8 v/o with 85 % of the hydrogen burned. The conservative nature of these values, assumptions and]3
approaches will be verified by results of 1/4 scale tests, and]3
comparisons with CLASIX-3 calculations.]3

4. The CLASIX-3 predictions of 1/4 scale test temperatures and pressures shall be compared with measured temperatures and pressures. The intent of this comparison shall be to demonstrate that CLASIX-3 conservatively predicts compartment average peak temperatures and pressures. Temperatures produced by any localized hydrogen combustion shall be compared with the compartment averaged temperature response.

13.13 Prepare Final Report on EPG and Analysis Assumptions

A final report on the comparison of operator actions identified in the combustible gas control EPG versus licensing analysis]3 assumptions will be prepared. This report will formally document the evaluation completed at Subtask 13.10 and the resolution of questions completed at Subtask 13.12. Any changes in licensing analysis assumptions and the expected effect of these changes will be documented.

Responsibility - HCOG
Status - Not Started

13.14 Submit Final Report on EPG and Assumptions to NRC

The final report prepared in Subtask 13.13 will be submitted to the Nuclear Regulatory Commission staff by the Hydrogen Control Owners Group. Based upon previous interaction with the Nuclear Regulatory Commission staff and earlier resolution of questions, it is not anticipated that this report will generate any additional questions.

Responsibility - HCOG
Status - Not Started

14.1 Identify All NTS Tests Completed During Program

A review of all Nevada Test Site (NTS) tests has been conducted. This review identified each test for additional evaluation. All NTS tests which might provide potential information concerning hydrogen combustion assumptions used in licensing analyses or concerning behavior of equipment used in Mark III containments in hydrogen burn environments were noted for further evaluation. These tests were identified from HCOG's monitoring of EPRI research activities in Subtask 6.8 and this review was initiated to resolve an open item with the NRC staff.

Responsibility - HCOG
Status - Complete

14.2 Summarize Test Data

A summary of test data and preliminary data plots for the test]3
conducted at the Nevada Test Site (NTS) have been received from]3
EPRI. Also draft versions of the following final reports have]3
been received from EPRI:]3

- "Large-Scale Hydrogen Combustion Experiments"]3
  Research Project 1932-11]3
- "Large-Scale Hydrogen Burn Equipment Experiments"]3
  Research Project 2168-3]3

The above information will be used in Tasks 14.3 and 14.4.]3

Responsibility - HCOG]3
Status - Complete]3

14.3 Identify Applicable Tests and Equipment Used in Mark III Plants

The tests identified and summarized in Subtasks 14.1 and 14.2 will be examined to determine if test conditions are applicable to Mark III containment conditions. Tests where hydrogen

be discussed. The effect of any differences on previous analytical work will also be addressed.

Responsibility - HCOG
Status - Not Scheduled

14.8 Resolve Questions on Evaluation Results

It is anticipated that NRC staff review of the NTS data evaluation and the comparison with previous assumptions could]3
generate questions concerning the effect on previous analytical results. Resolution of any questions which affect previous licensing assumptions will be input to decision point 8.9 which relates to the need for completing any new generic CLASIX-3 containment response analyses and Subtask 14.6.

Responsibility - HCOG
Status - Not Started

14.9 Prepare Final Report on NTS Data Evaluation

A report detailing the review conducted in subtask 14.1 through 14.5 will be prepared. Specific effects on assumptions used in containment and survivability analysis work from Subtask 14.5 will be included in this report. The report will finalize information contained in the preliminary report submitted to the Nuclear Regulatory Commission at Subtask 14.6. This report will also contain responses to any questions or requests for additional information identified by the Nuclear Regulatory Commission and resolved at Subtask 14.8.

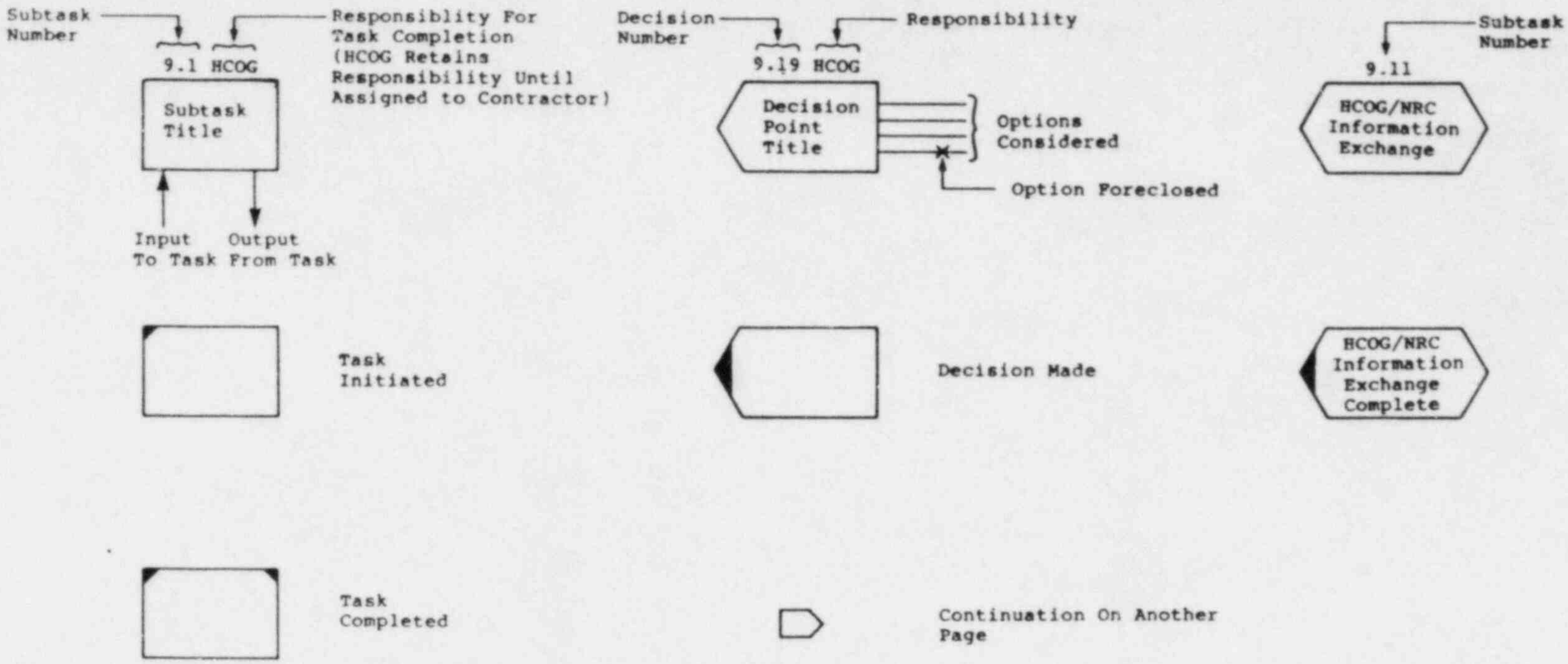
Responsibility - HCOG
Status - Not Started

14.10 Submit Final Report on NTS Data Evaluation to NRC

The final report documenting the NTS data evaluation and comparison with licensing assumptions prepared in subtask 14.9]3 will be submitted to the Nuclear Regulatory Commission staff. Any results which have resulted in changes in licensing assumptions or analysis methodology will be discussed in this report. Based on the previous interactions with the Nuclear Regulatory Commission staff, it is not anticipated that there will be any additional questions on this final report.

Responsibility - HCOG
Status - Not Started

EXPLANATION OF SYMBOLS



HCOG-Hydrogen Control Owners Group

Utility-Individual Members of HCOG As Applicable

GMF Associates-G. M. Fuls

FMRC-Factory Mutual Research Corp.

NRC - Nuclear Regulatory Commission

EPRI-Electric Power Research Institute

OPS-Offshore Power Systems

EPC-BWROG Emergency Procedures Committee

Enercon- Enercon Services, Inc.

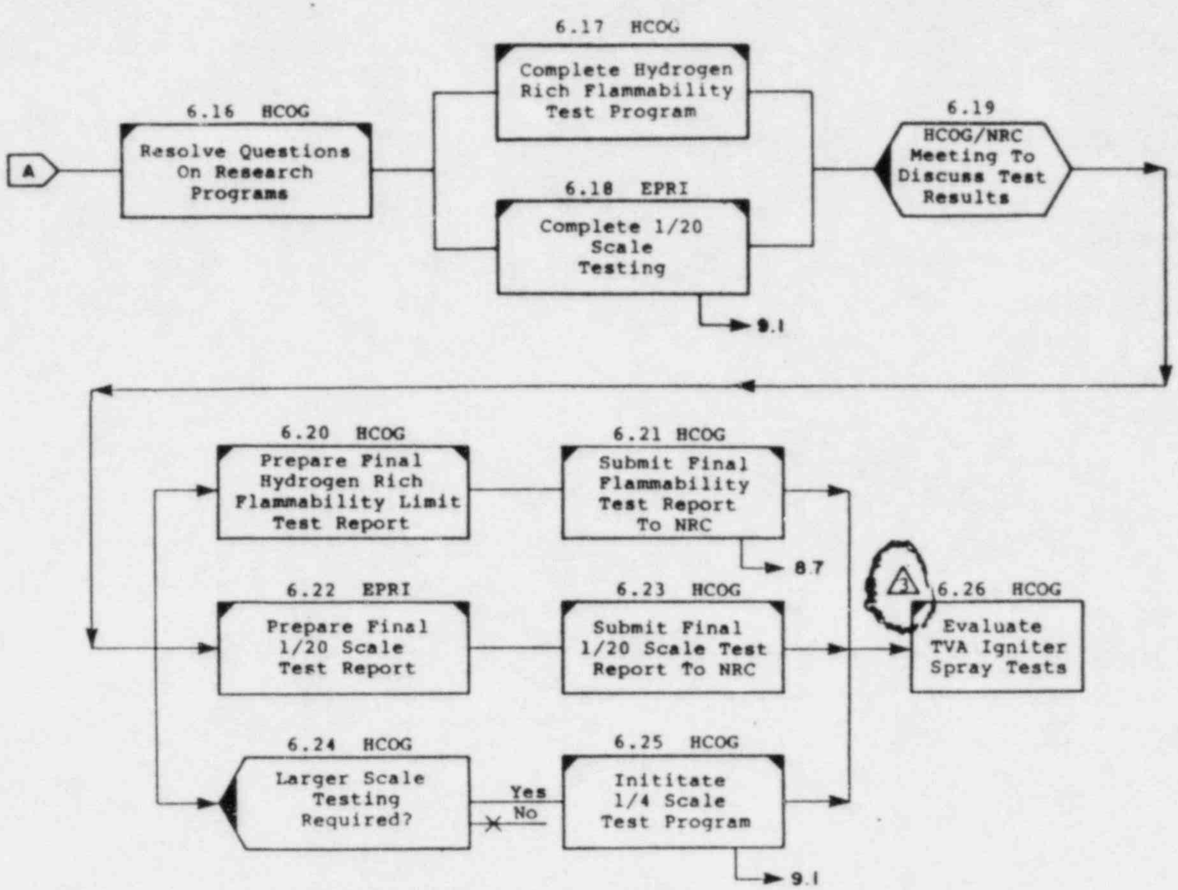
COMBEX - Combustion Explosives, Inc.

GE - General Electric Company

SWEC - Stone and Webster Engineering Corporation 3

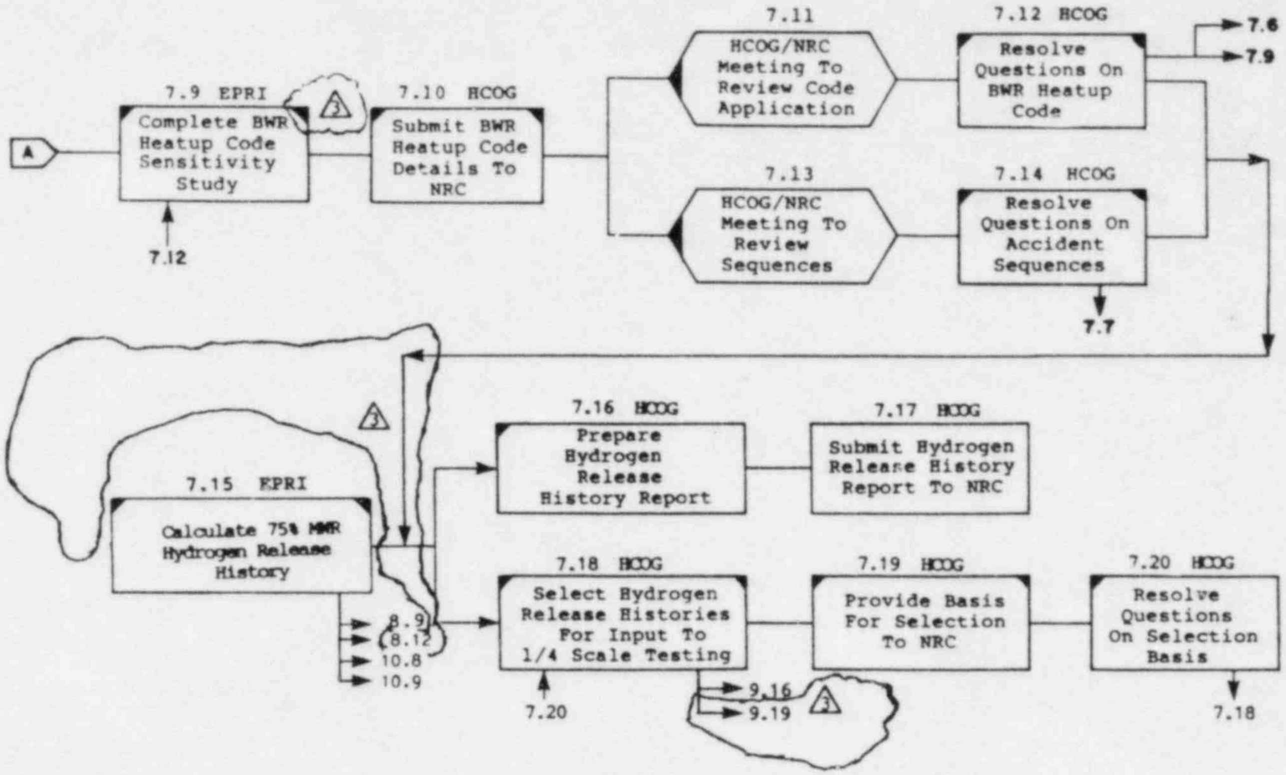
7/9/85

6.0 HYDROGEN COMBUSTION TESTING (CONT)

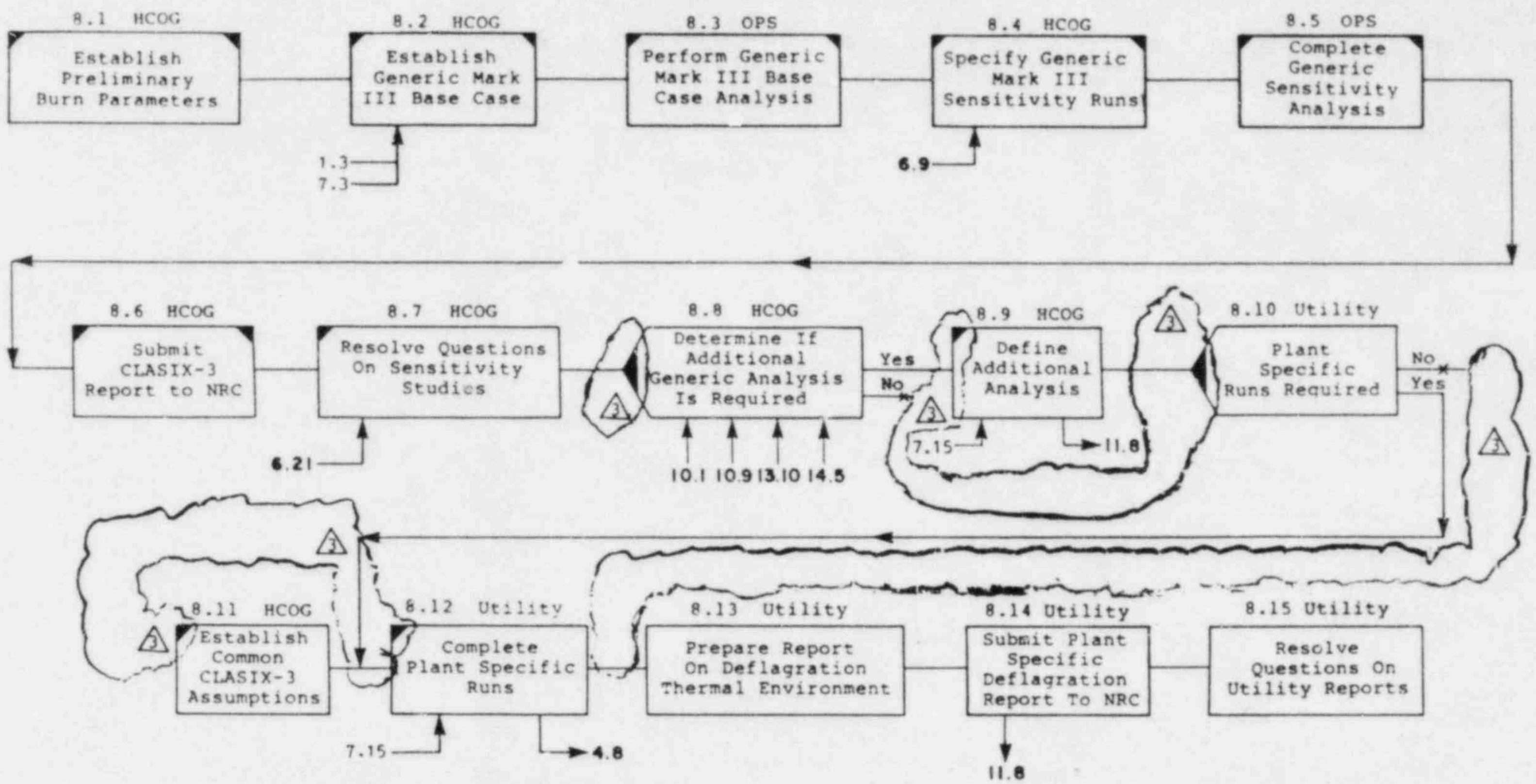


7/9/85

7.0 GENERATION OF HYDROGEN RELEASE HISTORIES (CONT)

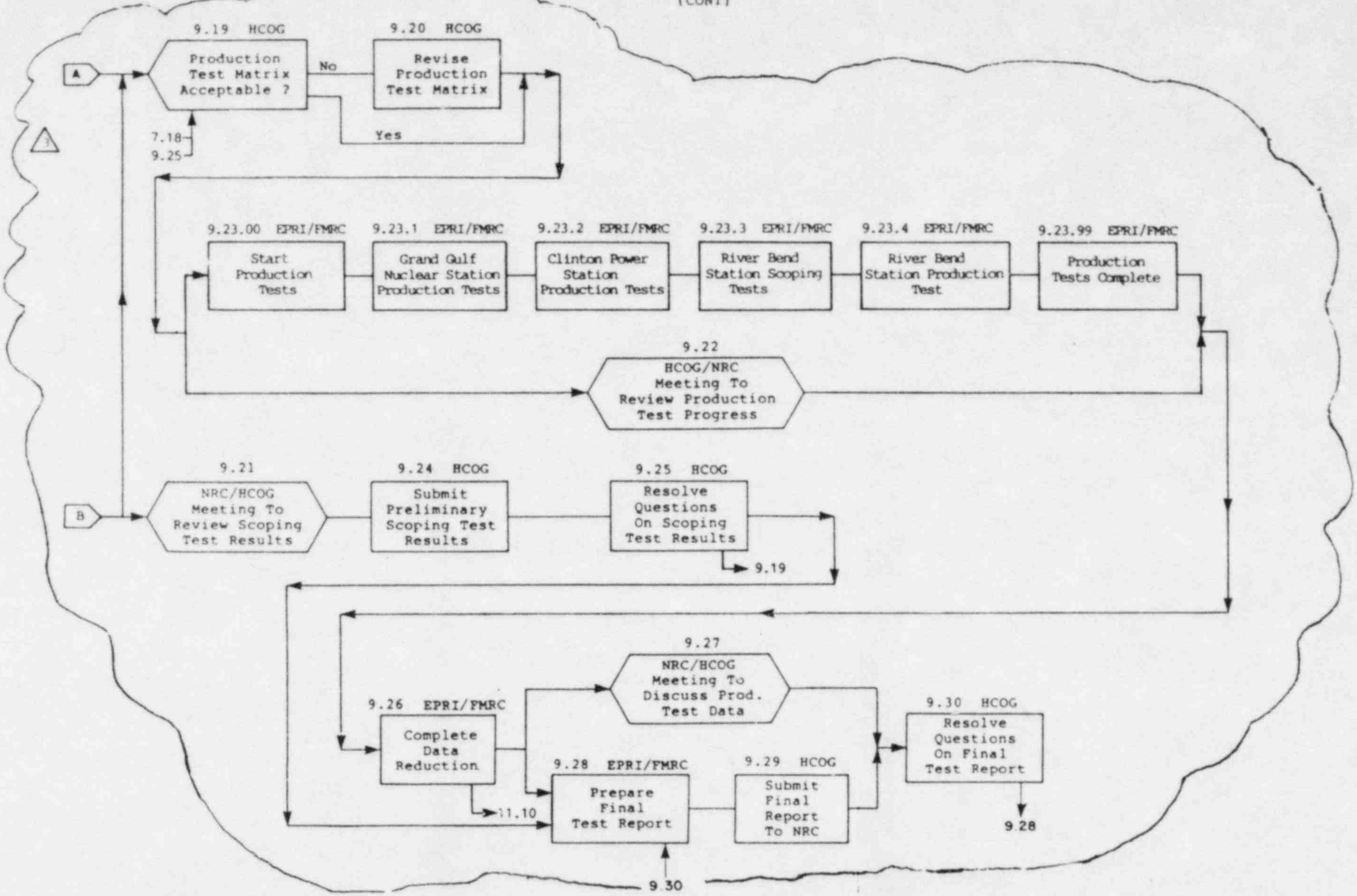


8.0 CONTAINMENT RESPONSE ANALYSIS



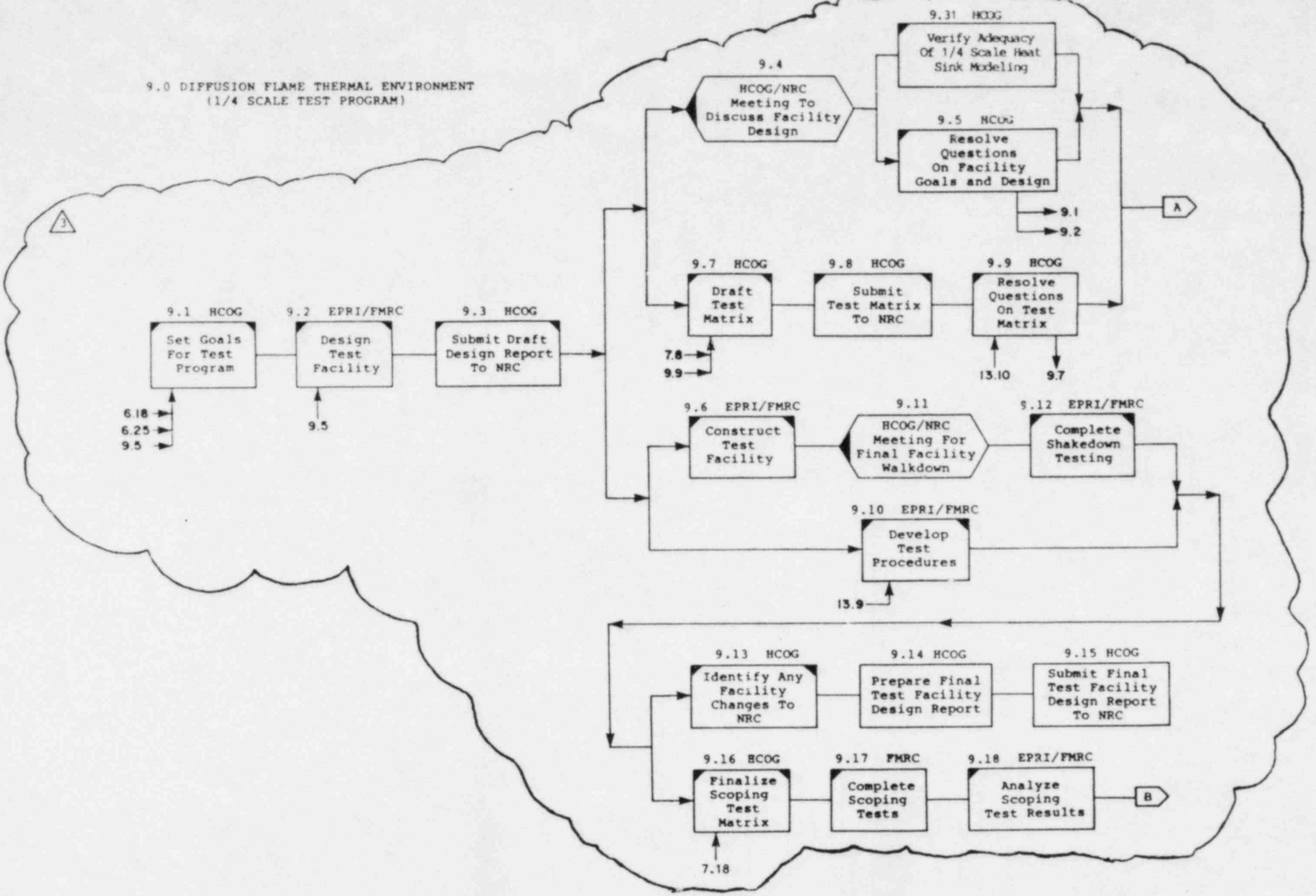
7/9/85

9.0 DIFFUSION FLAME THERMAL ENVIRONMENT
(1/4 SCALE TEST PROGRAM)
(CONT)



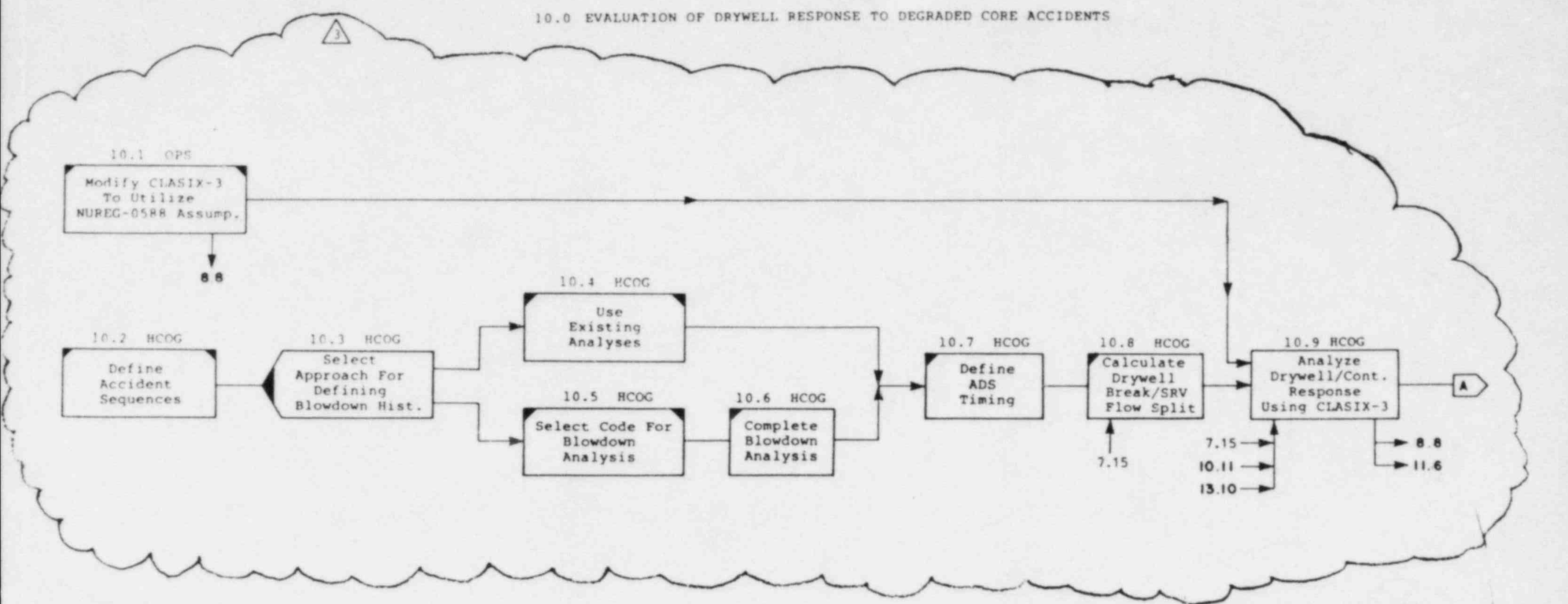
7/9/85

9.0 DIFFUSION FLAME THERMAL ENVIRONMENT
(1/4 SCALE TEST PROGRAM)



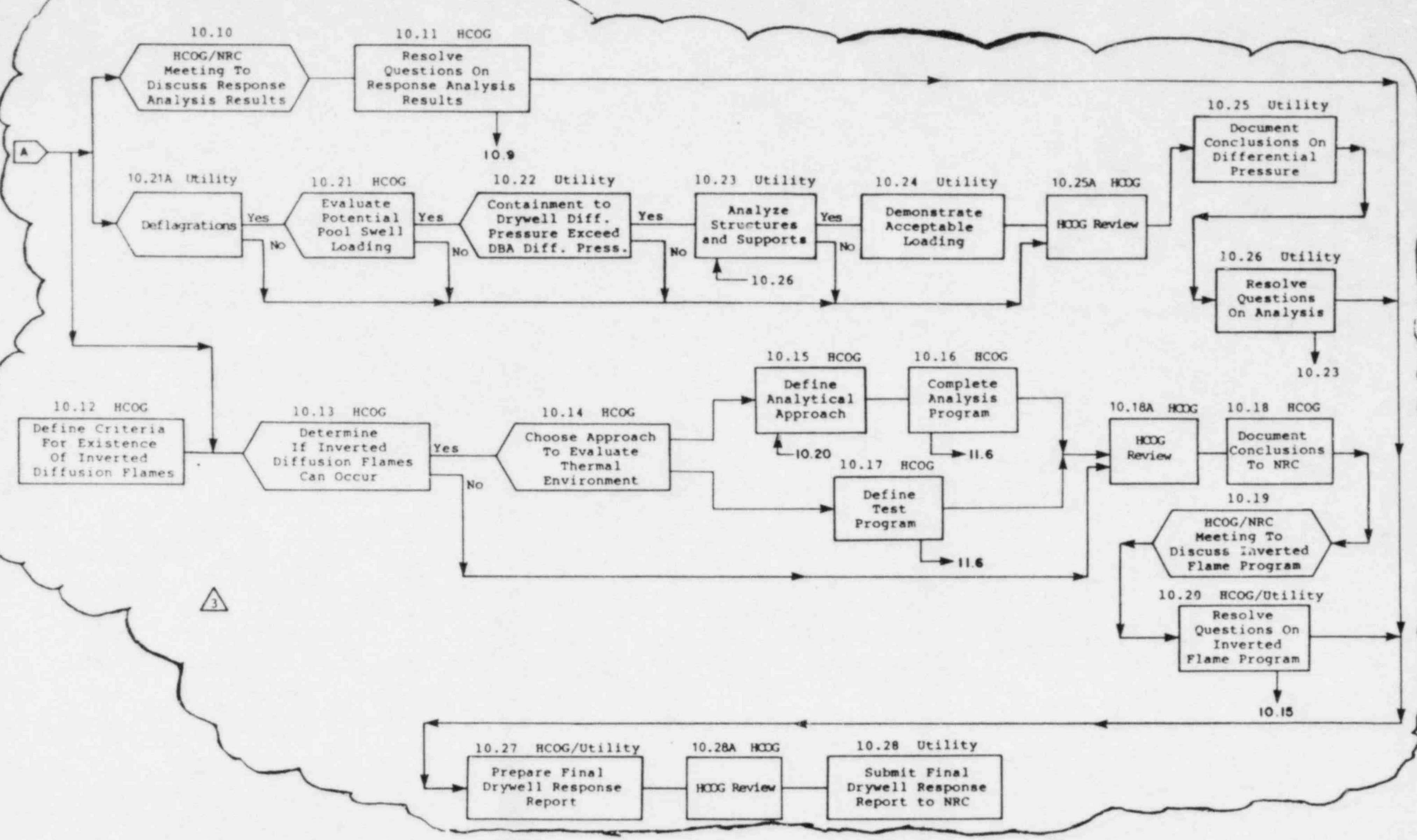
7/9/85

10.0 EVALUATION OF DRYWELL RESPONSE TO DEGRADED CORE ACCIDENTS



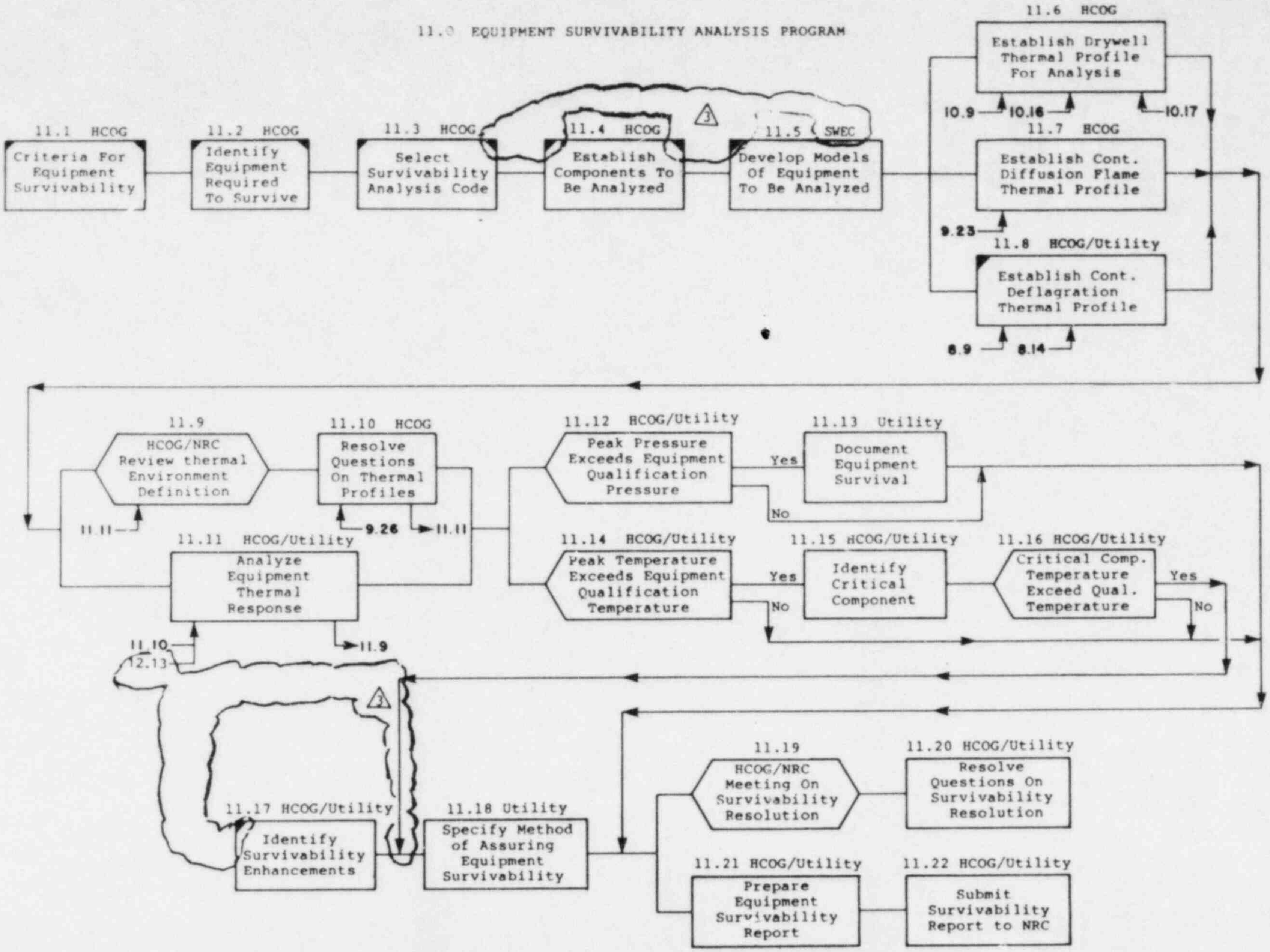
7/9/85

10.0 EVALUATION OF DRYWELL RESPONSE TO DEGRADED CORE ACCIDENTS (CONT)



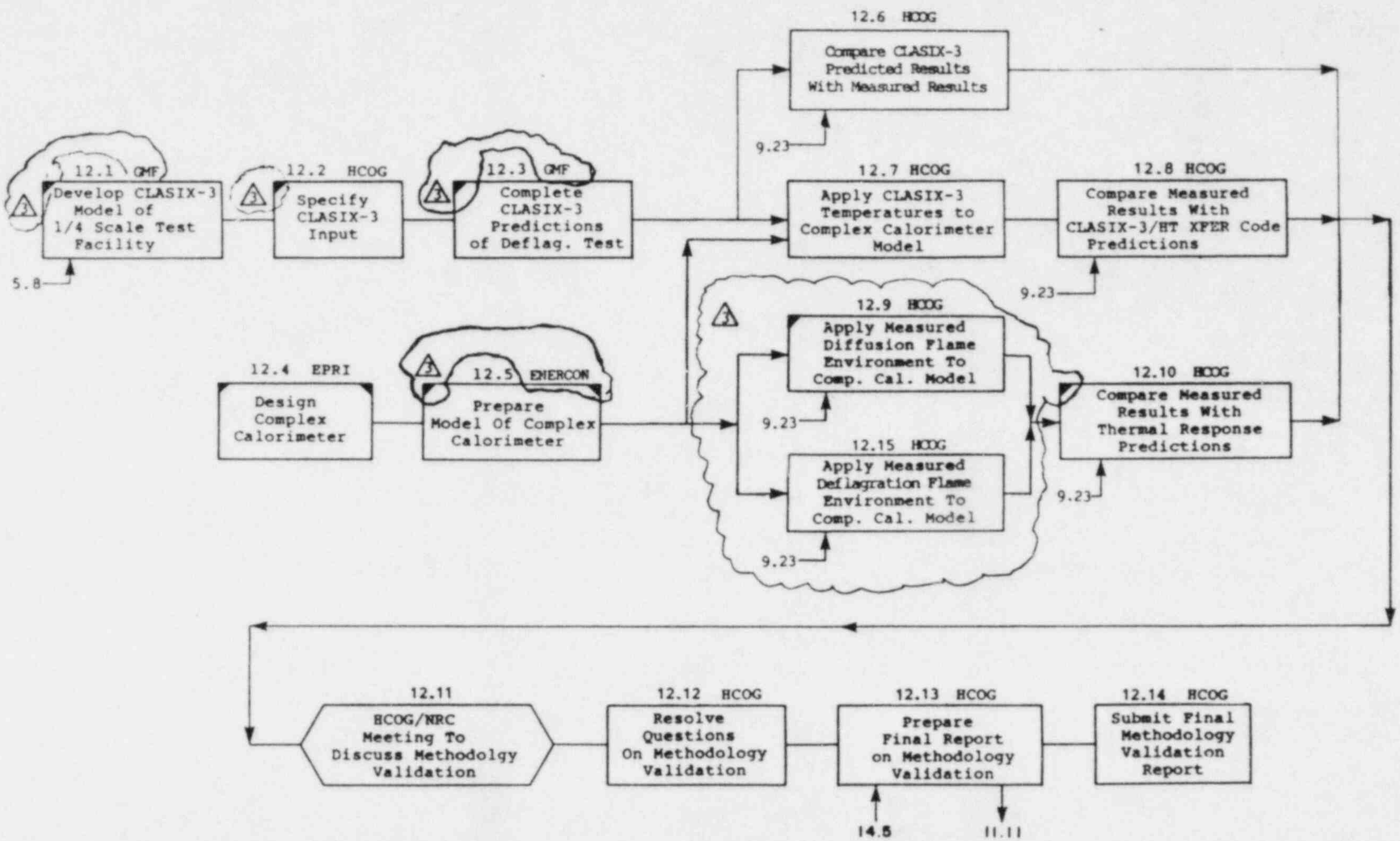
7/9/85

11.0 EQUIPMENT SURVIVABILITY ANALYSIS PROGRAM



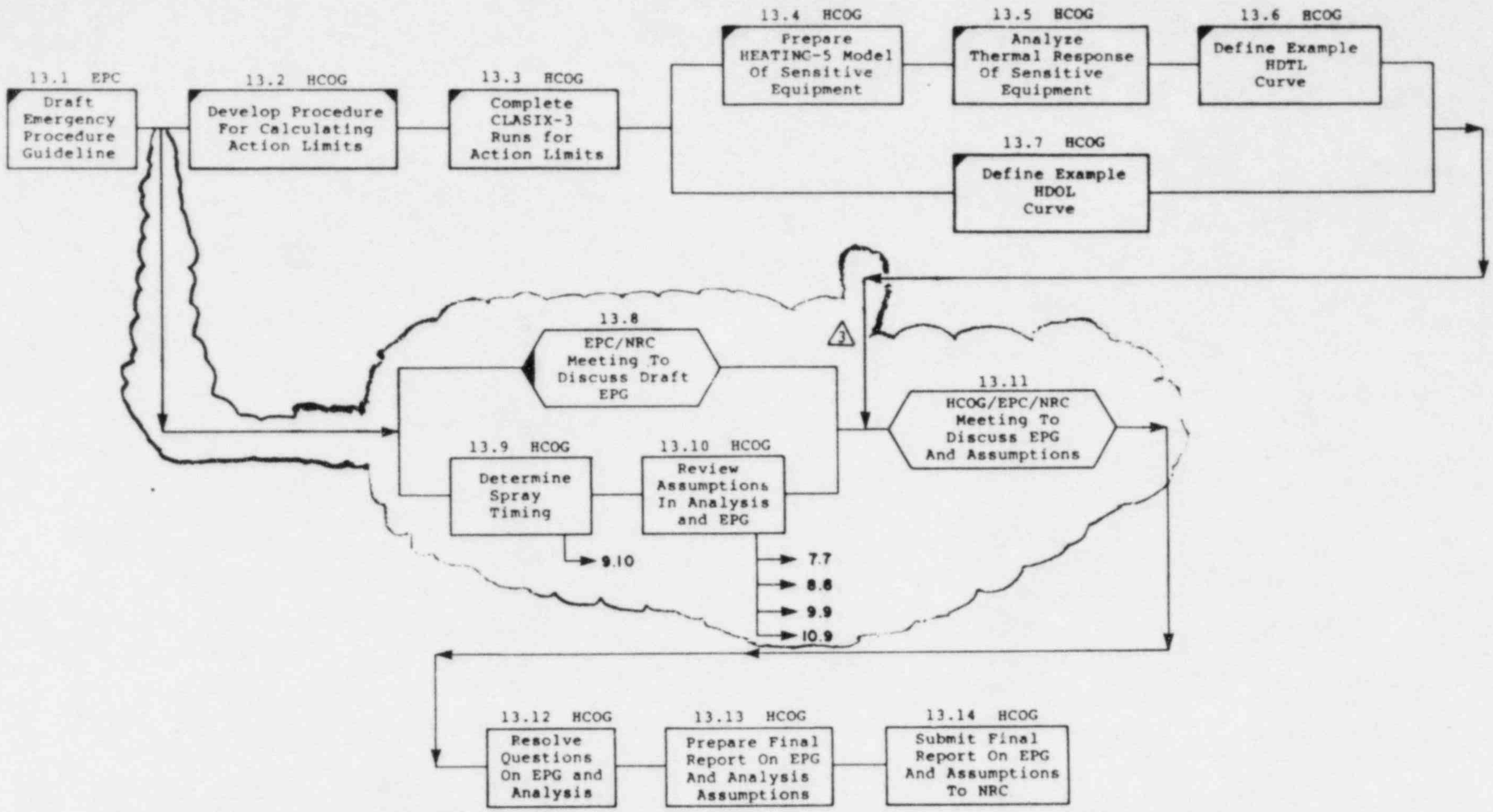
7/9/85

12.0 VALIDATION OF ANALYTICAL METHODS



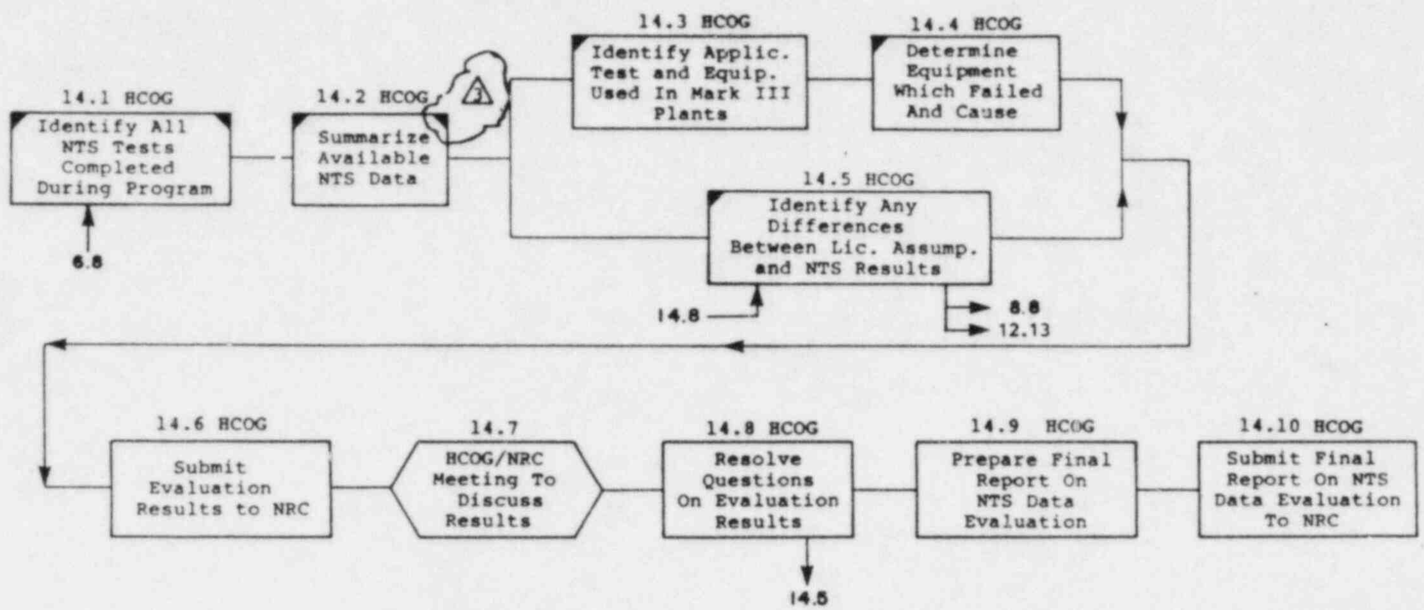
7/9/85

13.0 COMBUSTIBILE GAS CONTROL EPG



7/9/85

14.0 NEVADA TEST SITE DATA EVALUATION



5.0 SCHEDULE

A milestone schedule describing the duration of the tasks
identified in the Program Plan is shown on the attached figure.
Tasks that have been completed are no longer shown and tasks]3
that are utility unique will not be shown on this milestone]3
schedule but will be reflected in each utility's specific]3
schedule.]3

The milestone schedule identifies when the reports from the]3
various tasks will be submitted to the NRC and also when the]3
meetings between the NRC and HCOG which are identified in the]3
Program Plan are scheduled to occur.]3

The HCOG Schedule and Milestone Chart is based upon the]3
following assumptions. If any of these assumptions proved to be]3
otherwise the impact on the schedule could be significant.]3

- The Nuclear Regulatory Commission staff is responsive to HCOG meetings and review requests.

- The Nuclear Regulatory Commission staff provides prompt
  review of the Program Plan and provides a Safety]3
  Evaluation Report in a timely manner.]3

- The Nuclear Regulatory Commission staff provides acceptance of portion of the Program Plan as indicated on the attached milestone schedule.

- CLASIX-3 is an acceptable code to perform the Containment]3
  Response Analysis (Task 8)]3

- The drywell thermal environment is assumed to be]3
  calculable in Task 10, and a test program is not]3
  currently reflected in the schedule.]3

- Additional protection will not be required to meet the]3
requirements for Equipment Survivability (Task 11.0)]3

- The current Analytical Methods will be validated in Task]3
12.0.]3

- Current assumptions used in the various analysis and 1/4]3
Scale Testing Program will not be modified as the result]3
of the final review of the Emergency Procedure]3
Guidelines.]3

- Resolving the NRC's questions on the BWR Core Heatup Code]3
and the selection of scenarios will have no impact on the]3
schedule.

- The "parallel" effort to compare BWR Core Heatup Code]3
results with that of MARCH code will have no impact on]3
the schedule.]3

- Station blackout and ATWS scenarios need not be included]3
in the test matrices for the 1/4 scale test facility.]3

- Time Histories A, B and C for 1/4 scale testing will be]3
as indicated in NRC letter dated June 24, 1985.]3

- The test matrices for scoping and production tests as]3
submitted in HGN-031 (3-11-85) will not vary signifi-]3
cantly]3

Milestone Schedule

```
MLOG MILESTONES     1985                        1986                                            1987
------------------- JUN JUL AUG SEP OCT NOV DEC JAN FEB MAR APR MAY JUN JUL AUG SEP OCT NOV DEC JAN
       NAME         0   1   2   3   4   5   6   7   8   9   10  11  12  13  14  15  16  17  18  19

1.6                 0==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
3.12                0==============).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
6.26                0==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.12                0======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.14                0======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.16                .   S==============).   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.17 - TO NRC       .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.18                .   S======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
7.19                .   .   S======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
8.7-8.9             0==================).   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .
8.11                0===I.  .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.5                 0======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.14                0==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.15 TO NRC         .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.17,9.18,9.24      0==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
TO NRC              .   .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.21 NRC MTG        .   .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.25                .   .   .   )=======I.  .   .   .   .   .   .   .   .   .   .   .   .   .   .
9.19,9.20,9.23      .   .   S==================).   .   .   .   .   .   .   .   .   .   .   .   .
9.22 NRC MTG        .   .   .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .
9.26                .   .   .   S==================).   .   .   .   .   .   .   .   .   .   .   .
9.27 NRC MTG        .   .   .   .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .
9.28                .   .   .   .   .   .   .   .   S==============).   .   .   .   .   .   .   .
9.29 TO NRC         .   .   .   .   .   .   .   .   .   .   .   .   ♦   .   .   .   .   .   .   .
9.30                .   .   .   .   .   .   .   .   .   .   .   .   )=======I.  .   .   .   .   .
10.6-10.9           0==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
10.10 NRC MTG       .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
10.11-10.17         .   .   S==============).   .   .   .   .   .   .   .   .   .   .   .   .   .
10.18 TO NRC        .   .   .   .   .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .
10.19 NRC MTG       .   .   .   .   .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .
10.20               .   .   .   .   .   .   )=======I.  .   .   .   .   .   .   .   .   .   .   .
10.21-10.24         .   .   S==============).   .   .   .   .   .   .   .   .   .   .   .   .   .
10.25 TO NRC        .   .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .
10.26-10.27         .   .   .   .   S======================).   .   .   .   .   .   .   .   .   .
10.28 TO NRC        .   .   .   .   .   .   .   .   .   .   I   .   .   .   .   .   .   .   .   .
11.4-11.21          0======================================================================).   .
11.22 TO NRC        .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   I   .
12.1-12.10          0==============================).   .   .   .   .   .   .   .   .   .   .   .
12.11 NRC MTG       .   .   .   .   .   .   .   .   ♦   .   .   .   .   .   .   .   .   .   .   .
12.12               .   .   .   .   .   .   .   .   )=======I.  .   .   .   .   .   .   .   .   .
12.13               .   .   .   .   .   .   .   .   .   S======).   .   .   .   .   .   .   .   .
12.14 TO NRC        .   .   .   .   .   .   .   .   .   .   .   I   .   .   .   .   .   .   .   .
13.1-13.13          0======================).   .   .   .   .   .   .   .   .   .   .   .   .   .
13.11 NRC MTG       .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
13.14 TO NRC        .   .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .
14.2-14.5           0======).   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
14.6                .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
14.7 NRC MTG        .   .   ♦   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .   .
14.8-14.9           .   .   )==========).   .   .   .   .   .   .   .   .   .   .   .   .   .   .
14.10 TO NRC        .   .   .   .   .   I   .   .   .   .   .   .   .   .   .   .   .   .   .   .
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