TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

CORE DEGRADATION PROGRAM

VOLUME 1

HYDROGEN STUDY

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DISCLAIMER

The study of degraded core accidents at the Sequoyah Nuclear Plant, their effects, and their mitigations, described in the accompanying report was performed in the 6-week period concluded April 15, 1980. Of necessity, due to the limited duration of the study and the state-of-the-art information available to TVA, several limitations were in order.

First, the calculational techniques to analyze the course of degraded core accidents and their effects on containment were not available since those events are not included in the current plant design bases. As a result, simplifying assumptions and methods were relied on when necessary to estimate the accident consequences.

Second, the evaluations of the overall risk posed by Sequoyah with and without additional accident mitigations are somewhat subjective due to the limited time and data available for probability and consequence estimation. Only a limited number of accident sequences were examined and best-estimate judgments were made for the associated containment failure probabilities.

Third, the level of detail and degree of feasibility achievable in the conceptual designs of the mitigations were restricted by the available time and personnel. Further effort might result in somewhat different designs.

Fourth, the costs that were originally estimated for the design, construction, and operation of each of the mitigation concepts have not been included in this version of the report.

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SUMMARY

A preliminary evaluation of events beyond the design basis leading to reactor fuel element damage and generation of hydrogen at the Sequoyah Nuclear Plant has been performed. The purpose of this evaluation was to develop additional information about such events and to compare the relative merits of concepts that have been suggested to mitigate the consequences of such events. This information serves as a basis for further TVA action.

Our conclusions based on this study are:

1. The risk due to core damage events is sufficiently small to allow continued operation of Sequoyah. This is based on the following findings:

   a. The risk from these events is comparable to that at Surry which was used in the Rasmussen study (WASH-1400); recent information has indicated that the risk is less than originally estimated in that study.

   b. The differences in containment design between Sequoyah and PWR's with large "dry" containments do not lead to increased risk.

   c. The Sequoyah containment can be expected to survive a Three Mile Island-type event with burning of hydrogen up to at least that produced by 20-percent metal-water reaction.

   d. TVA has taken substantial measures since TMI to reduce the likelihood of core damage due to small loss-of-coolant accidents.

2. If the licensing process allows, TVA should not proceed with implementing a mitigation concept at this time. This is based on the above assessment and the following findings.

   a. The one concept (inerting containment with nitrogen during operation) that fully prevents hydrogen combustion has serious safety problems and major costs due to large plant downtimes.

   b. Other concepts provide partial protection, but involve additional risks or major technical problems.

   c. Because no concept was identified which provides substantial additional capability to handle hydrogen without major additional risks or technical problems, further study of options could lead to a better decision.

3. Venting of containment, especially filtered vented containment, provides very little protection for hydrogen effects. The pressure increase rate is too fast for realistic vent sizes. However, these concepts can provide protection for other slower overpressure transients.

4. If a mitigation concept must be selected now, we recommend installation of the ignition sources. Even though igniters do not provide complete protection, they would give the plant operators a way to ensure that the hydrogen burns at low concentration, at least for those events where the buildup of hydrogen is slow. Work needed to implement this concept includes:

   a. A development program for the igniters.

   b. A program for analyzing a range of events involving core damage to identify timing of events, the rate of release of hydrogen to containment, and the time and spatial dependence of hydrogen and oxygen concentrations.

   c. Selection of locations for igniters.

   d. A testing program to verify the system.

   e. A program for analyzing the effects of combustion on equipment in containment and the containment itself.

   f. Procurement of ignters and additional hydrogen analyzers.

   g. Installation and testing.

Our preferred approach is:

1. For Sequoyah unit 1, full power license, demonstrate by analysis that the risk is no greater than for current operating PWR's and verify by more detailed analysis the containment's capacity to tolerate limited hydrogen burning.

2. Find out if some of the technical problems with the concepts can be resolved and the additional risks minimized through a research and development program.

3. Continue our investigation of the basic events to improve our understanding. This will serve as a basis for later decisions and may give additional insight into how the events may be prevented.

4. Actively participate in NRC's proposed rulemaking proceedings by developing and sharing information on the subject. A probable outcome is a decision on how severe an event (degree of core damage/core melt) should be designed for.

Purpose

The purpose of this report is to explore the phenomena associated with the generation and release to containment of substantial amounts of hydrogen due to events leading to core damage and to evaluate the relative potential of various concepts put forward to control these phenomena. The report addresses these matters for the Sequoyah Nuclear Plant. The overall objective is to provide guidance for further TVA action in regard to these matters.

Background

Light water cooled nuclear power plants are designed to prevent and mitigate accidents which could lead to significant core damge (i.e., reduced fuel pin integrity due to overheating and oxidation cf the zirconium leading to cladding failure). Major design measures, analysis, and equipment have been used to provide assurance that such damage won't occur. In addition to the attention given the subject by the nuclear steam supply system vendors and the Nuclear Regulatory Commission (and its predecessor, the Atomic Energy Commission), TVA has expended considerable attention by inhouse experts on accident prevention and mitigation. Nonetheless, significant core damage did, in fact, occur at Three Mile Island unit 2 (TMI-2). This event called for a reassessment of the likelihood and consequences of such an event even though the consequences at TMI-2 were much less severe than other events considered in plant design.

Hydrogen Effects

In order to produce reasonable thermal efficiencies, reactor coolant systems operate at moderately high pressures and temperatures. If the reactor coolant is lost due to rupture of the system or other reasons, cooling water must be injected to carry off the energy released by the decay of fission products contained in the irradiated fuel. If cooling water is not injected or is not effectively conveyed to the core, the fuel pins heat up. At elevated temperatures, the zirconium clad oxidizes (if steam is present), resulting in a loss of clad strength, release of energy, and production of hydrogen. The hydrogen is released to the containment building. If the amount is sufficient, the hydrogen can burn in the containment, resulting in a large energy release to containment, increasing the containment temperature and pressure. If the burn is big enough and fast enough, the containment could fail due to excessive pressure, releasing fission products to the environment.

If core cooling is reestablished in time (as was done at Three Mile Island), the damage can be limited; if not, progressive core damage occurs, leading to complete collapse of the core (core melt).

If complete core melt occurs, the hydrogen and other noncondensible gases could cause containment failure due to overpressure even without

hydrogen combustion.

Events Which Could Lead to Core Damage

Core damage cannot occur if core cooling systems function after an
accident. The cooling systems are redundant and independent with
redundant sources of power. Therefore, multiple equipment failures
or operator errors are required to occur for failure of the cooling
systems. We have studied the three most important initiating events:
the large loss-of-coolant accident; the small loss-of-coolant
accident, and failure of secondary side heat removal system. If these
inititing events are accompanied by failure of all core cooling
systems, core damage will occur. There is some time, before
significant core damage exists, for the operator to take remedial
action to restore cooling (from a few minutes to about a couple of
hours, depending on the event), thereby arresting damage. In this
time, the water in the reactor boils away and the core heats up.

If remedial action is not taken, oxidation of the fuel proceeds fairly
rapidly (in a few minutes time) until a significant fraction of the
clad is oxidized (50-75 percent). If remedial action is taken in
this time period, damage can be arrested while the core is a coolable
geometry. If remedial action does not take place, melting occurs
and the core fragments fall to the bottom of the reactor vessel.
After a period of time, vessel failure can occur due to heating and
melting, dropping the fuel-clad-vessel metal mixture (corium) into
the reactor vessel cavity where it reacts with water to produce steam
and with the concrete to produce noncondensible gases. Hydrogen is
produced by oxidation of the core supports, the vessel itself, and
the remaining clad.

It has been postulated that eventually the core could melt its way
through the containment concrete leading to releases of fission
products to the ground water. Recent information indicates that this
will probably not occur.

Sequoyah Containment Design

The Sequoyah containment, along with several other plants, differs
from the containment used at TMI-2. Sequoyah uses a pressure
suppression type containment, wherein a large bed of ice is used to
condense steam and remove energy. Some other plants use a pool of
water instead of ice. The pressure suppression concept leads to
smaller containments with lower design pressures (typically 1/2 to
1/3 the size and 1/3 to 1/4 the design pressure of "dry containments")
and increased compartmentation. It is expected that these
characteristics could lead to a different response to significant
hydrogen generation. The large, passive heat sink (ice or pool) is
an excellent means of removing heat; on the other hand, the lower
volume results in higher hydrogen concentrations (for the same amount
of hydrogen produced) while the lower design pressure allows less
pressure buildup in containment.

Prevention

Prevention is more useful than mitigation; TVA has responded to the TMI event (through its Nuclear Program Review) with a number of steps to prevent accidents like that at TMI. These steps included:

1. Increased training of operators, improved procedures, and improved information to the operator to: (a) reduce the likelihood of operator errors negating the safety features provided to prevent and mitigate accidents; (b) help the operator diagnose events; and (c) help the operator respond to unexpected conditions.

2. Features to remove gases from the reactor coolant system which might prevent cooling water from getting to the core.

3. Improved equipment to further prevent the occurrence of a small loss-of-coolant event like TMI-2.

4. A special test program at Sequoyah to demonstrate natural circulation cooling of the core and provide experience to the operators.

Risk

TVA has evaluated the risk due to core damage using methodology similar to that used in the Reactor Safety Study, WASH-1400. The results indicated that the risk for Sequoyah is roughly the same as for other pressurized water reactors; that is, the containment has somewhat less capability for accommodating noncondensibles, but this small additional risk is balanced by the presence of passive heat removal capability and other advantages. We believe that the preventative measures incorporated since the Three Mile Island event further reduce the risk.

Inherent Capability

Our analysis demonstrates that Sequoyah could safely withstand an event like Three Mile Island unit 2 which included combustion of hydrogen equivalent to 20 percent metal-water reaction. That is, if all the hydrogen produced as a result of oxidation of 20 percent of the fuel clad burned, the resulting containment pressure would be approximately equal to the predicted ultimate strength of the containment. When coupled with TVA's preventative measures, this inherent capability provides increased confidence that the risk from core damage events is acceptably low.

Mitigation Potential

Notwithstanding the above, we have performed a preliminary evaluation of various concepts suggested for mitigating the effects of core damage (including the effects of hydrogen). These concepts are listed in table 1 along with comments on their effectiveness and major drawbacks. In general, our conclusions were:

- Some concepts promise to be effective for protecting against failure due to hydrogen burning or failure due to other

overpressure causes, but no single concept is effective for both.

- Some concepts, although effective for either hydrogen or overpressure, pose significant safety concerns.

- To provide substantial protection for all core damage events would require a combination of mitigating concepts (i.e., one concept to mitigate hydrogen combustion and one to mitigate other overpressure causes).

- The most widely publicized concept, filtered vented containment, provides little added protection from the effects of hydrogen but involves a very significant risk of its own since it uses deliberate release of radioactive gases to the environment.

- The potentially most effective mitigator of hydrogen effects (containment inerting) is used by TVA in its Browns Ferry Nuclear Plant. However, we have concluded that its use at Sequoyah could actually increase risk due to reduced inspections inside containment and increased hazards to personnel.

- It appears that with additional research and development, an acceptable mitigation scheme could be developed if required.

Mitigation Strategies

Based on what we know at this time, we believe that the following levels of added protection might prove feasible if required.

- Increase the Amount of Hydrogen that can be Accommodated

  1. Use distributed ignition sources to burn hydrogen at low concentration (4-6 percent) to minimize buildup.

  2. Use halon injection to suppress combustion.

  3. Provide an additional containment structure with a high capacity mixing system to reduce the concentration.

  These can be used individually or in combination. Item 2 requires research to identify the long range incontainment behavior of halon to assure that essential equipment would not be damaged. It appears that these steps singly or in combination could increase the capacity to withstand hydrogen effects by 50 to 300 percent. Ignition sources are preferred over halon injection because of concern over the affect of decomposition of halon in the long term.

- Provide Complete Protection from Hydrogen Combustion

  1. Only containment inerting will provide complete protection.

     Containment inerting has additional inherent risks (reduced safety system reliability because of reduced containment access for inspection, maintenance, and repair) and high cost.

- Provide Protection from all Core Damage/Core Melt Phenomena that
  can Lead to Overpressure Failure, Including Hydrogen Combustion

  1. Significant protection by providing ignition sources and/or
     halon injection and one of the vented containment concepts.

  2. Complete protection by providing containment inerting and
     additional containment

     The costs for these options would be on the order of   to
                  per plant plus an operating cost of up to
            dollars per year. Note that there are other
  containment failure modes (isolation failure, steam explosions,
  missiles) that would not be mitigated by these measures,
  although these failure modes are considered less likely than
  overpressure.

Summary

Our investigation shows:

- While ice condensers and other suppression type containments react
  differently than "dry" containments, the overall risk is not
  significantly different.

- Sequoyah has inherent ability to withstand the effect of burning
  the hydrogen produced by oxidation of about 20 percent of the
  fuel clad.

- Several concepts investigated appear to have some potential for
  mitigating the effects of hydrogen combustion or containment
  overpressure. However, all of the concepts capable of mitigating
  hydrogen effects pose additional safety concerns or have major
  technical problems.

- Some concepts investigated appear to be effective in preventing
  uncontrolled combustion; some appear to be effective in preventing
  overpressure failure due to noncondensible gases. No concept
  identified appears to be effective for both.

- A combination of concepts might provide additional protection,
  but substantial research and development will be required to obtain
  an acceptable design.

TECHNICAL SUMMARY

TABLE 1

MITIGATION CONCEPTS

Concept	Comments
Inerted Containment	Effective for preventing hydrogen combustion; significant safety concerns due to reduced access to containment equipment and hazard to personnel. Ineffective for containment overpressure protection.
Halon Injection	Moderately effective for preventing hydrogen combustion; questions exist in regard to Halon's long term behavior and effect on equipment; ineffective (may aggravate the situation) for containment overpressure protection.
Ignition Sources	Moderately effective for all but rapid releases; potential for misoperation; minor impact on containment overpressure.
Filtered Vented Containment	Limited benefit for protection from hydrogen combustion effects; moderately effective for containment overpressure protection; the deliberate release of noble gases is a major safety concern.
Additional Containment	Somewhat effective for protection against the effects of (and preventing) hydrogen combustion. Effective for containment overpressure protection; expensive.
Coupled Containment	Similar to additional containment but somewhat less effective because second containment may not be available at all times the first unit is operating; safety concern in regard to impact on nonaccident unit.

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1.0 Purpose

The purpose of this study was to assess current containment
capabilities and to identify and evaluate concepts for the mitigation
of the consequences of core damage, including hydrogen generation,
in order to prevent failure of the containment. The results are
intended to provide additional insight into the events that could
lead to significant hydrogen generation, the effects of such
quantities of hydrogen, and the relative benefit, cost, and drawbacks
of various mitigation concepts. While the subject is applicable to
all TVA nuclear plants (particularly those incorporating pressure
suppression type containments), Sequoyah plant characteristics were
used in this study.

2.0 Scope

The scope of this study included six major areas of effort:

2.1 Identification of key accident sequences - - A discrete
number of accident sequences were selected for use as representative
of a wide range of types (small loss-of-coolant accident (LOCA), large
LOCA, etc.) and severities (moderate core damage, severe core damage,
etc.) of potential accidents at Sequoyah.

2.2 Analysis of present Sequoyah containment for key accident
sequences - - The physical consequences of the key accidents and their
effects on the present SQN containment were analyzed. The current
risks from these key accidents without additional mitigation were
also estimated.

2.3 Identification of mitigation concepts for key accident sequences
- - Concepts for mitigation of various consequences of the key
accidents were identified based on prior industry
experience,feasibility, or current regulatory attention.

2.4 Development of mitigative conceptual designs - Conceptual designs
were produced for the identified mitigation schemes. This included
general arrangement of components, bill of materials, and
identification of principal engineering factors.

2.5 Evaluation of mitigative concepts- - The conceptual designs were
each evaluated from several standpoints - physical effectiveness,
risk reduction, cost, design and construction schedules, nuclear
safety, operational impact, feasibility, etc.

2.6 Recommendations for future study - - The experience gained in
these efforts has made it possible to recommend specific areas where
further research and development would be useful.

3.0   Introduction

One of the design basis accidents for Sequoyah is a large LOCA. The consequences of this accident would include large mass and energy releases from the primary system rupture followed by the generation of small amounts of hydrogen in the reactor core and in the containment. Sequoyah safety systems currently include both active and passive measures to mitigate these accident consequences. To mitigate the primary fluid loss and the accompanying pressure and temperature effects, high and low pressure injection systems actively reflood and cool the primary system, containment spray systems actively cool the containment atmosphere, the ice condenser passively cools the containment atmosphere, and the containment structure itself is designed to retain its integrity at pressures above those expected. To handle the hydrogen produced, recombiners remove it from the containment atmosphere. These systems have the capability to alleviate all of the consequences of the current design basis accidents with adequate margin.

The accident at Three Mile Island involved moderate mass and energy releases from the primary system through a stuck-open relief valve. This was effectively a small LOCA, and the appropriate safety systems were automatically initiated as designed. Manual override of the high pressure injection system by the operators at TMI led to uncovering the core for longer periods than in any of the design basis accidents. This resulted in extensive core damage accompanied by hydrogen generation significantly above that considered in design. For this reason, TVA is reassessing the effect of hydrogen on plant safety.

The best approach to prevent hydrogen from jeopardizing plant safety is to prevent its formation in significant quantities by maintaining adequate core cooling. The emergency core cooling systems are redundant, qualified systems whose performance has been evaluated conservatively in great detail to ensure their effectiveness. They have received the most attention of any plant system. Nothing in the TMI experience indicated that they were not adequate. TVA therefore believes that the Emergency Core Cooling Systems (ECCS) at Sequoyah are sufficiently reliable to perform their intended functions, and that complete failure of the ECCS is of sufficiently low probability. However, TVA is responding to the TMI accident through commitments made in its TVA Nuclear Program Review ("Blue Book" and "Green Book") to make improvements in several areas. These include instrumentation changes to make more information available to the operator, hardware changes to make improved system operating modes available for the operator's use, and intensified training of the operators to respond more effectively to inadequate core cooling incidents. As a result of these post-TMI efforts to incorporate improvements in both hardware and procedures, we feel that the likelihood of Sequoyah experiencing inadequate core cooling has been significantly reduced. The Nuclear Safety Analysis Center of the Electric Power Research Institute has estimated that risk has already

been reduced by a factor of three to five for U.S. light water
reactors following TMI, and further improvements will reduce the risk
by another factor of two to four below WASH-1400 (3-1) estimates.

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3-2

4.0 Background of the Original Design

Sequoyah incorporates numerous features to provide for safe operation and to mitigate a wide range of accident and transient sequences. These features have been studied previously to determine their effectiveness in dealing with the plant design basis events. This section briefly describes the present design bases, analytical tools for evaluating events, and the implications of TMI on the present design bases.

4.1 Physical Plant Features

There are three principal barriers to the release of radioactivity during normal operation and accidents. These barriers, the fuel rod cladding, the reactor coolant system, and the containment perform independently and have been designed to be highly reliable and effective for current design basis events. A detailed discussion of the fuel and reactor coolant system may be found in Appendix A.1.

4.1.1 Containment

The containment is a large structural volume designed to contain the radiation, mass, and energy released from a breach of the reactor coolant system and the nuclear fuel cladding. The Sequoyah containment is a free-standing steel structure with a free volume of 1.2 million cubic feet. The design pressure of this structure is 26.4 psia which exceeds the calculated pressures resulting from any of the present design basis events.

The containment is divided into three compartments: The lower compartment, the upper compartment, and the ice condenser compartment (figures 4-1 and 4-2). The lower compartment completely encloses the reactor coolant system equipment. The upper compartment contains the refueling canal, refueling equipment, and the polar crane used during refueling and maintenance operations. The ice condenser connects the lower compartment to the upper.

4.1.2 Passive Containment Heat Removal Systems

The ice condenser contains approximately two and one-half million pounds of ice. This quantity of ice is capable of removing all of the heat released by a large LOCA for the first hour. The steel containment shell and the walls and equipment located inside containment are heat sinks that remove energy from the containment atmosphere, thereby reducing the containment pressure and temperature during an accident.

4.1.3 Active Containment Heat Removal Systems

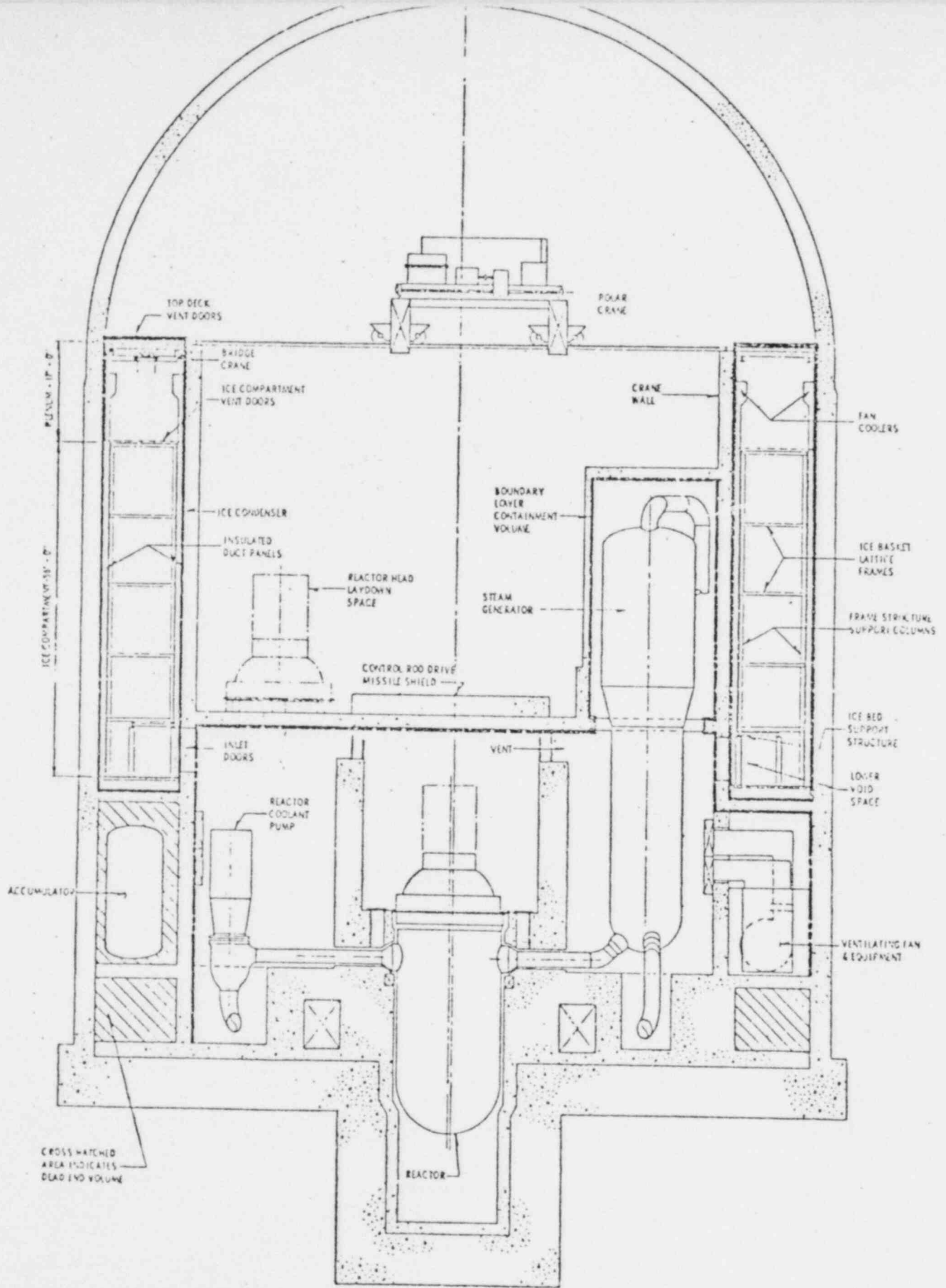In addition to the ice condenser, several systems remove heat from

FIGURE 4-1
CONTAINMENT CROSS SECTION
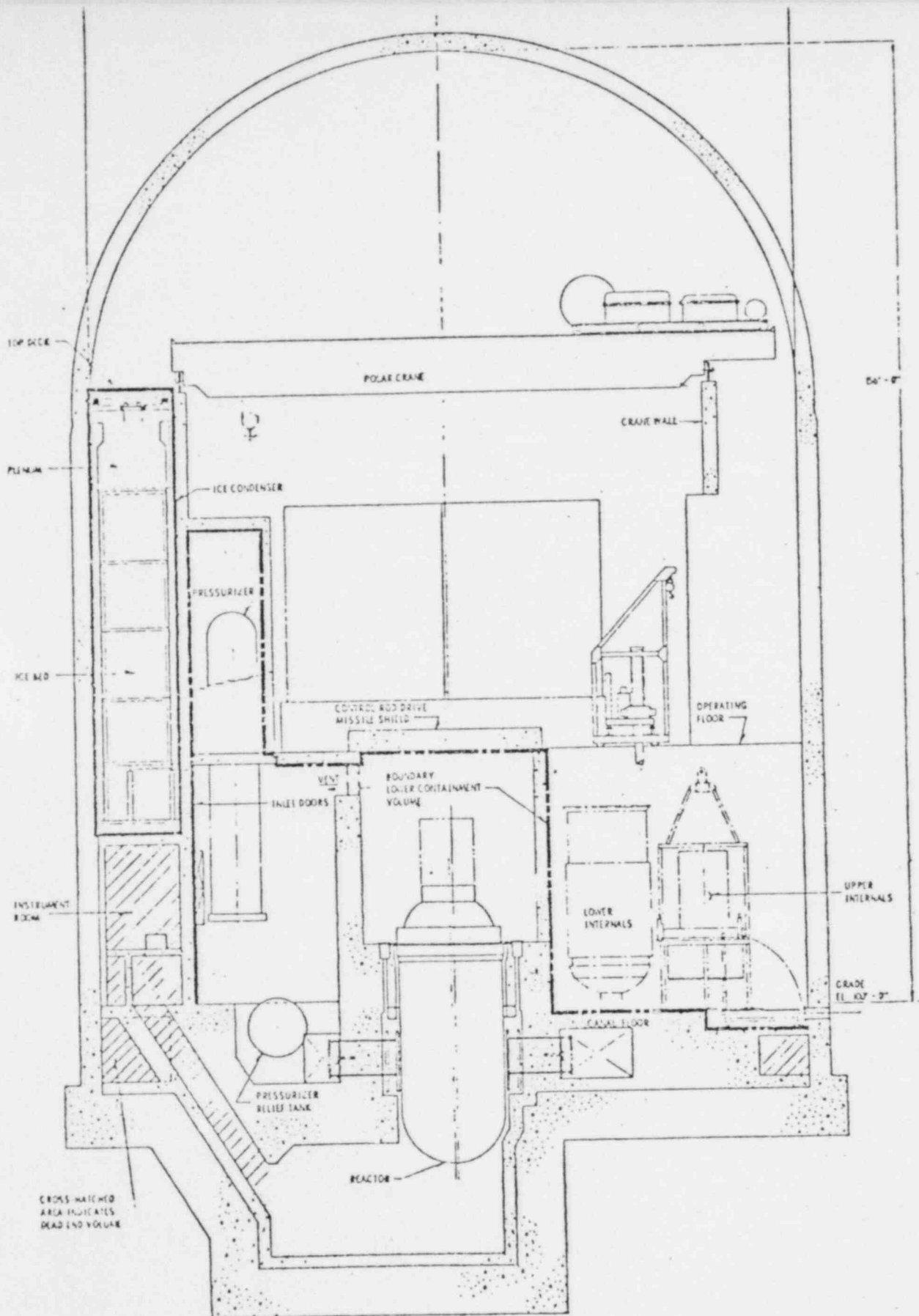
FIGURE 4-2
CONTAINMENT CROSS SECTION

the containment during a LOCA. The containment spray system removes heat from the containment atmosphere initially by the addition of water from the refueling water storage tank (RWST). When the water supply in the RWST is exhausted, the supply for the pumps is changed from the tank to the emergency sump located in the containment. Water from the sump is run through heat exchangers in the residual heat removal (RHR) system and the containment spray system to remove heat from the containment sump water and thereby from the containment.

4.1.4 Hydrogen Control Systems

Several safety systems are incorporated into the design of the Sequoyah containment to mitigate the effects of hydrogen production following the original design basis accident. The hydrogen mitigation systems function following the accident to detect, mix, and consume the hydrogen produced. Each subsystem presently included in the Sequoyah design is briefly described in Appendix A.5 as to function and operation principle. The systems employed will adequately process all of the hydrogen postulated in the original design basis accident scenario with 100-percent margin. They include hydrogen analyzers for postaccident monitoring, electric hydrogen recombiners for removal of hydrogen, a containment mixing system to prevent pockets of hydrogen at higher concentrations than the general containment airspace, and a small hydrogen purge system as a backup to the recombiners.

4.2 Present Design Bases

The containment is designed to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. The design basis for the containment is generally taken to be the complete severance of a large reactor coolant system pipe at a location which results in the worst consequences for the containment. See Appendices A.3 through A.3.2 for further discussion of the containment design bases including mass and energy release and formation of hydrogen.

4.2.1 Accidents

The limiting design basis accidents for the containment are loss-of-coolant accidents (LOCA) due to their potential release of radioactivity. A spectrum of LOCA's are studied from small pipe breaks (pipes less than 2" in diameter) to a break in the largest main coolant pipe (approximately 30" in diameter). Most studies performed to date concentrated on the large pipe break because it was believed to present the greatest challenge to the fuel cladding, the containment, and the plant safety systems. The analysis of such accidents is extremely complex and requires the consideration of heat transfer, fluid flow, neutronics, and the interaction of safety systems.

4.2.1.1 Mass and Energy Releases

Following a postulated rupture of the Reactor Coolant System (RCS),

steam and water are released into the containment. Initially, the water in the RCS is subcooled at a high pressure. When the break occurs, the water passes through the break where a portion flashes to steam at the lower pressure of the containment. These releases continue until the RCS depressurizes to the pressure in the containment (end of blowdown). At that time, the vessel is refilled by water from the accumulators and Safety Injection (SI) pumps. Water flowing from the accumulators and SI pumps start to fill the downcomer causing a driving head across the vessel which forces water into the hot core. When a level two feet below the top of the core is reached, the core is assumed to be reflooded and totally quenched, leaving only decay heat to generate steam.

The LOCA analysis calculational model is typically divided into three phases: (1) blowdown, which includes the period from accident occurrence (when the reactor is at steady state full power operation) to the time when zero break flow is first calculated; (2) refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum; and (3) reflood, which begins when water starts moving into the core and continues until the end of the transient.

4.2.1.2 Design Basis Noncondensible Gas Production

Another source of mass that must be considered in containment design is the production of noncondensible gases during the accident. It is possible for nitrogen gas to enter the containment from the passive emergency core cooling system components. This source, while not significant for containment pressurization, is considered in the containment analysis. More importantly, the production of flammable gases is of concern. The more severe loss-of-coolant accidents postulated for plant design may result in the production of excess hydrogen which ultimately ends up in the containment air space. Depending on the final volumetric concentration of the hydrogen and the quantity of oxygen present, the gas can be a fire hazard, an explosion hazard, or too diffuse to chemically react.

Several sources of hydrogen have been identified for the LOCA. These include dissolved hydrogen in the reactor coolant system, which is released during blowdown; radiolysis by fission product decomposition of the coolant; fuel cladding oxidation from high fuel cladding temperatures concurrent with steam flow over the clad; and corrosion of zinc-coated surfaces in the containment interior by spray water concurrent with high ambient containment temperatures. Approximate quantities of gas produced by each mechanism are shown in table 4-1. It should be noted for the design basis event that the zirconium cladding reaction is important in the short term and the other production mechanisms are important in the long term.

See Appendices A.3 through A.3.2 for further discussion of the containment design bases, including mass and energy release and hydrogen formation.

TABLE 4-1

DESIGN BASIS HYDROGEN SOURCES

Source	Potential H_2 (lbm)
Zirconium (43,204 lbm Zr) 3 percent RXN	57
Primary System Dissolved Gas	6
Aluminum Corrosion (long term)	142
Zinc (long term - paint, galvanized steel corrosion)*	1800

*Most of the zinc is in the ice condenser components which remain at a low temperature and low corrosion rates until late in the accident.

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4.2.2 Analytical Tools

Some of the most difficult problems to analyze are the thermal
hydraulic problems encountered in reactor safety. Even with advanced
computers, it is difficult to completely analyze all but one or two
dimensional thermal-hydraulic problems. A brief introduction to the
analytical tools available to study the loss of coolant accident and
their limitations is necessary to a complete overview of the predicted
plant response to the events described in this report.

4.2.2.1 Primary System Thermal Hydraulic Programs

The primary system pressure, temperature, and flow response to various
postulated accidents is calculated using state-of-the-art computer
codes. The software tools numerically approximate the mass, momentum,
and energy conservation equations for a series of interconnected
volumes and flow paths simulating the reactor system. The Westinghouse
WFLASH (4-1) and SATAN (4-2) codes typify the design tools used for
Sequoyah primary system analysis. These codes, as well as more
recently developed programs that encompass greater detail, are
discussed in Appendix A.4.1.

The currently available codes treat the behavior of steam and water
mixtures but do not follow the presence of a noncondensible gas (e.g.,
nitrogen, hydrogen, and other noncondensibles) in the primary system.
This limitation is not a problem for the design basis accidents since
the production of large quantities of noncondensibles is not predicted.

4.2.2.2 Containment Thermal Hydraulic Programs

The current design basis analyses performed to determine containment
design parameters consider the addition of water or steam from a
spectrum of pipe breaks in either the RCS or the secondary side of
the steam generators. Computer codes have been written which again
use the conservation of mass, energy, an momentum to evaluate the
pressure and temperature inside the cont inment at any point in time.
Heat transfer coefficients, ice melt rates, and other critical factors
required to evaluate the performance of the ice condenser are
determined empirically for various mass and energy release rates.
The LOTIC (long term ice condenser code) computer code (4-3) was
developed by Westinghouse specifically to evaluate ice condenser
containments and was used to evaluate the Sequoyah containment. As
in the case of primary system computer codes, the LOTIC code has not
been written to treat hydrogen separately. A detailed discussion
of the capabilities of LOTIC may be found in Appendix A.4.

4.2.3 Predicted Plant Response

Studies of the design basis LOCA at Sequoyah indicate approximately
1-3 percent of the core zircaloy fuel cladding is calculated to

oxidize based on conservative 10CFR Appendix K computer models. Although the core is calculated to uncover and to heat up to the temperature where zircaloy oxidation rates are high, the core is quickly reflooded and cooled by the emergency core cooling systems, thus limiting oxidation. Production of hydrogen from the oxidation of core metal is therefore not expected to exceed 60 pounds of gas. No hydrogen generation is predicted from the vessel steel or from stainless steel cladding of the vessel since these materials do not heat up to reaction temperatures during the accident. The containment spray system is designed to operate throughout the accident, and the combination of spray water and containment temperature will corrode the containment painted surfaces and any aluminum present during the long term. Additional hydrogen will be produced by fission product radiolysis of water in the core and sump. This source also generates oxygen during the process which must be considered for inerted containment designs.

The initial release of hydrogen from core cladding oxidation after a design basis LOCA is accommodated by dilution in the containment air space. Long term hydrogen sources are accommodated by the use of the hydrogen recombiners which heat the hydrogen/air mixture initiating chemical recombination to produce water vapor. Typical hydrogen production rates from the above sources are shown in figure 4-3. Assuming at least one recombiner is started in the first day following the accident, the containment hydrogen does not exceed the 4 percent flammability limit as indicated in figure 4-4. If no recombiners are available, the hydrogen purge system can be started several days into the accident to maintain hydrogen concentrations within acceptable limits.

In the event of the design basis LOCA, the maximum containment pressure is 11.8 psig. Figures 4-5 and 4-6 provide the containment pressure and temperature response curves for this event. As can be seen in the pressure response curve, the containment rapidly pressurizes due to large mass and energy releases that occur during the blowdown period. The containment pressure starts dropping as the steam condensation rates in the ice condenser exceed the steam releases from the primary system due to decay heat and steam generator sensible heat. This trend continues until ice melt at approximately 4000 seconds. Over the next several hundred seconds, the containment pressure rises to its maximum value. The containment spray systems then maintain the pressure within the containment design limits once the ice has melted and in the long term gradually reduce the pressure to near atmospheric.

4.3 Ramifications of TMI on Current Design Bases

The accident that occurred at the Three Mile Island facility on March 28, 1979, has led to the reassessment of the design basis hydrogen production at Sequoyah. The event was primarily a small break LOCA with inadequate core cooling. The LOCA resulted from a stuck- .

HYDROGEN PRODUCTION RATES
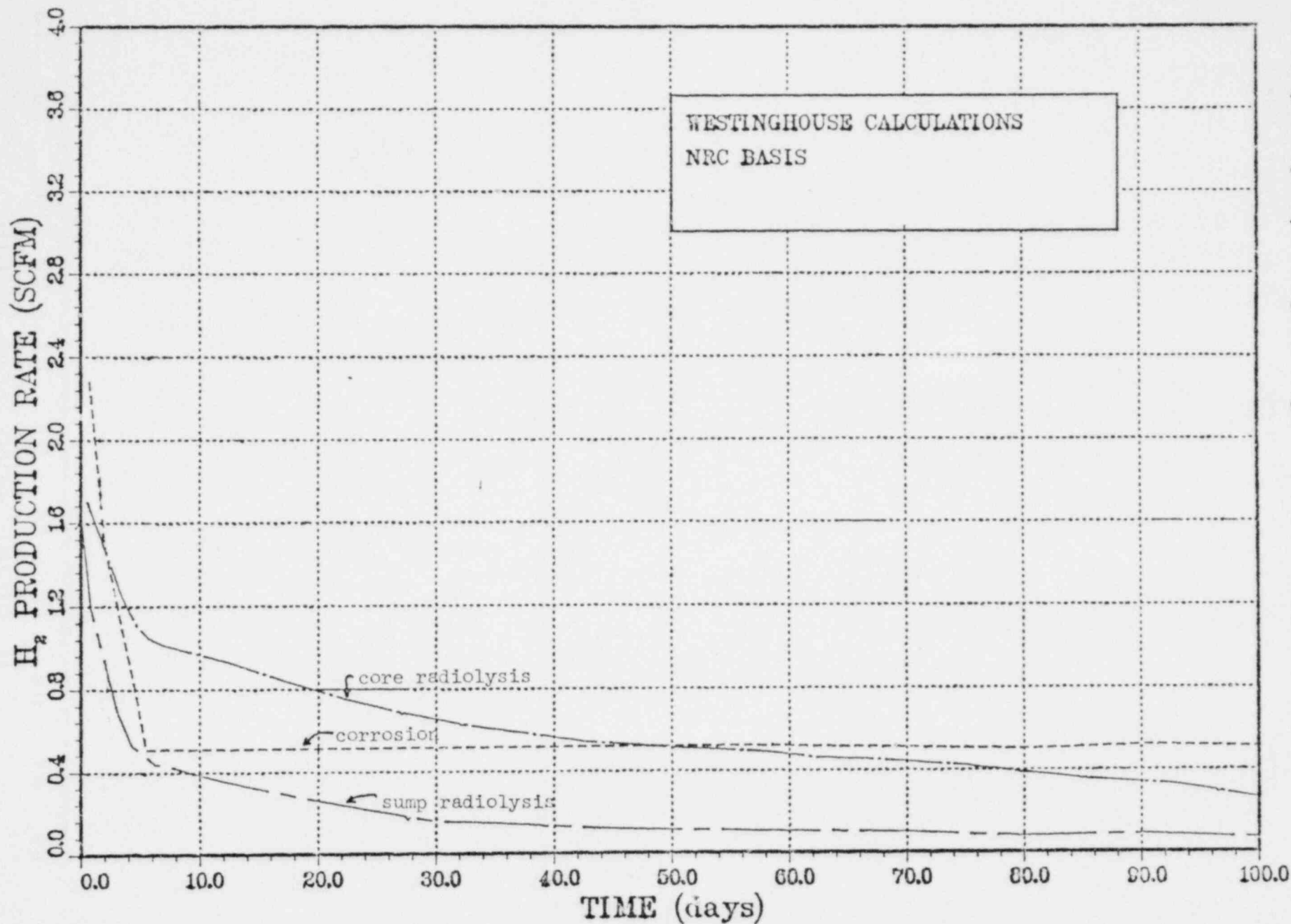
FIGURE 4-3

CONTAINMENT HYDROGEN VOLUME PERCENT
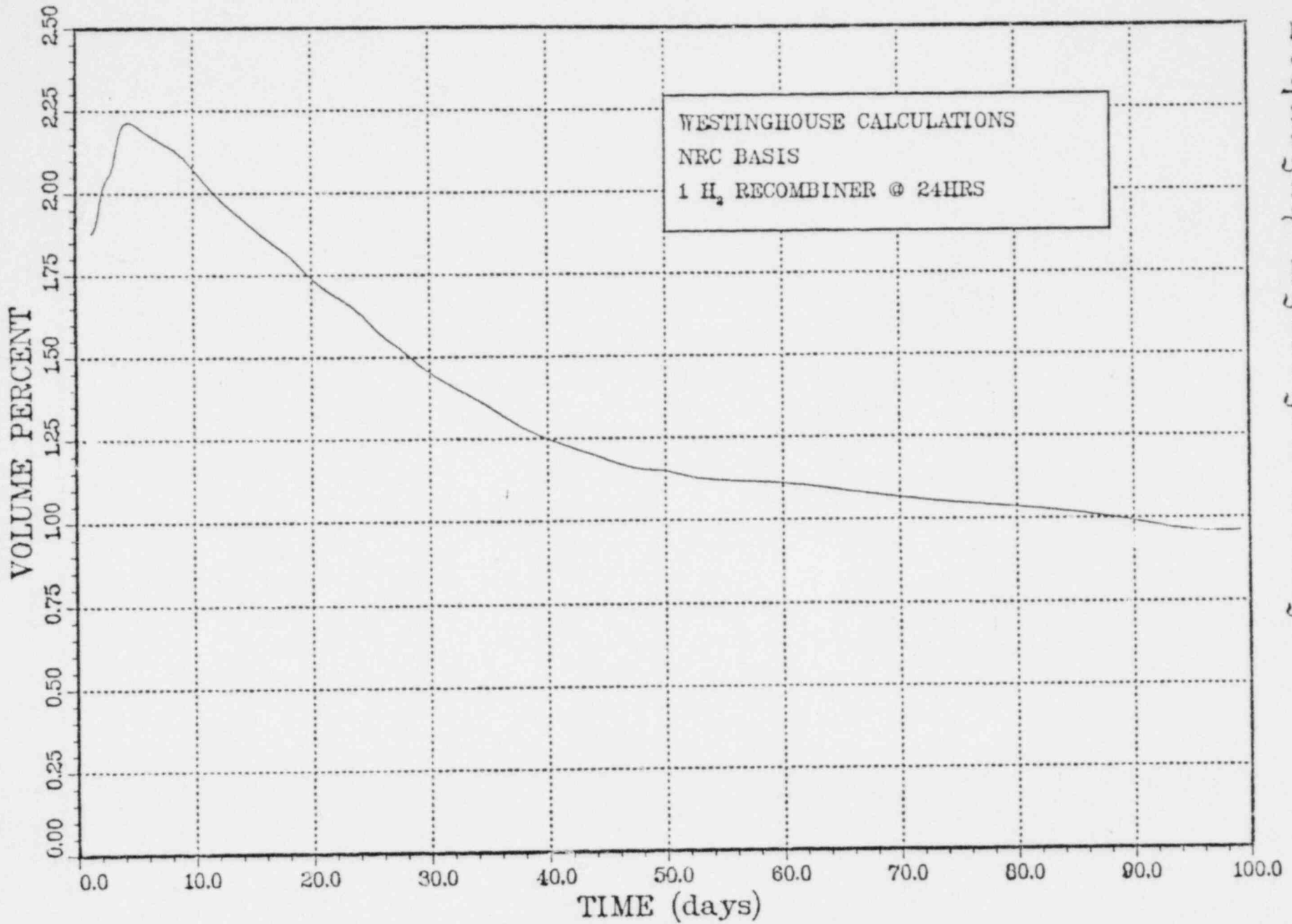
FIGURE 4-4

CONTAINMENT PRESSURE RESPONSE

DOUBLE ENDED PUMP SUCTION BRK
INCLUDING ICE MELT
SQN FSAR FIG. 6.2-20

PRESSURE (psig)

TIME (seconds)

FIGURE 4-5

4-11

Sequoyah Hydrogen Study

CONTAINMENT TEMPERATURE RESPONSE

DOUBLE ENDED PUMP SUCTION BRK
INCLUDING ICE MELT
SQN FSAR FIG. 6.2-30

Lower Compartment

Upper Compartment

TIME (seconds)
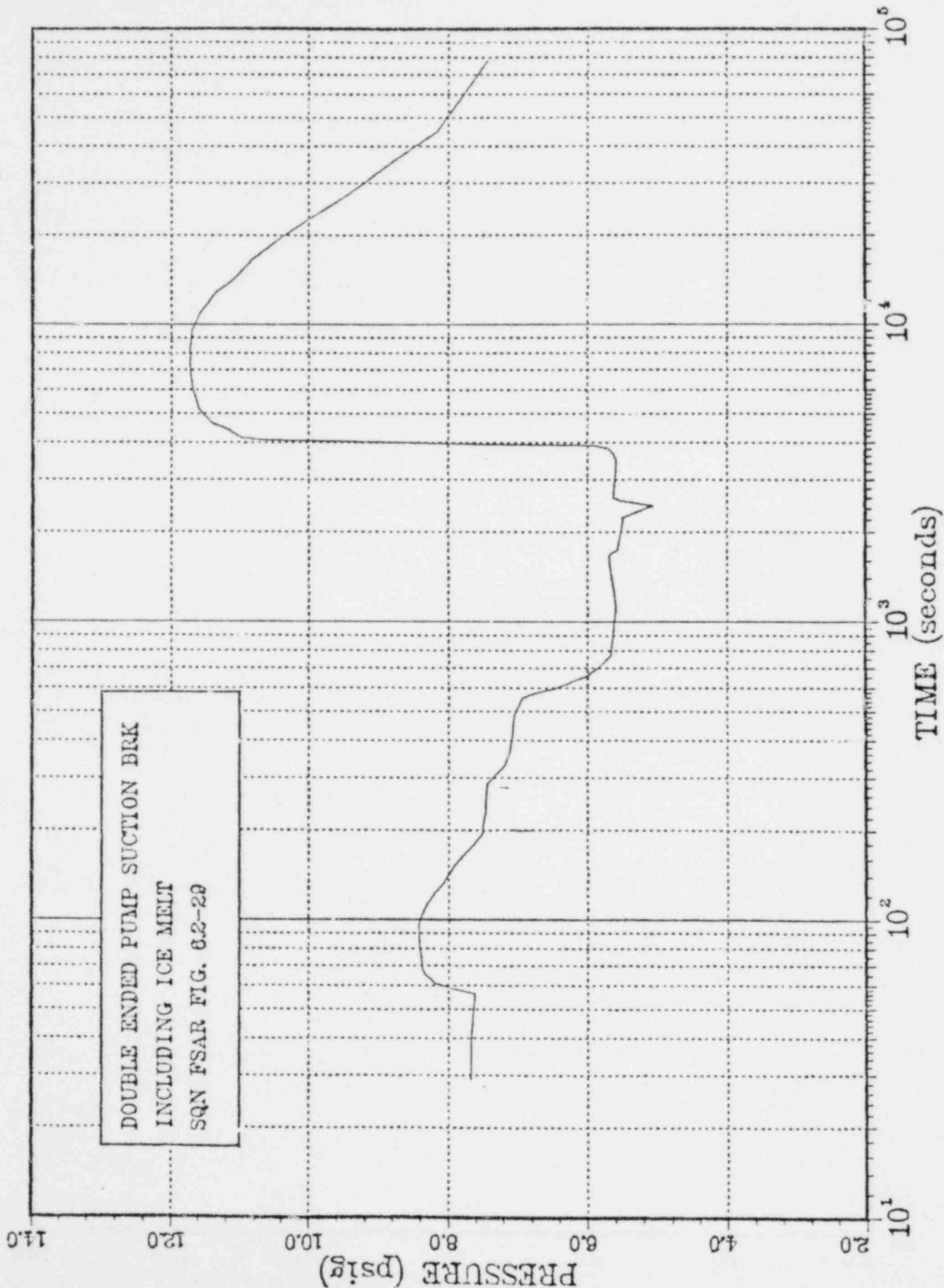
TEMPERATURE (°F)
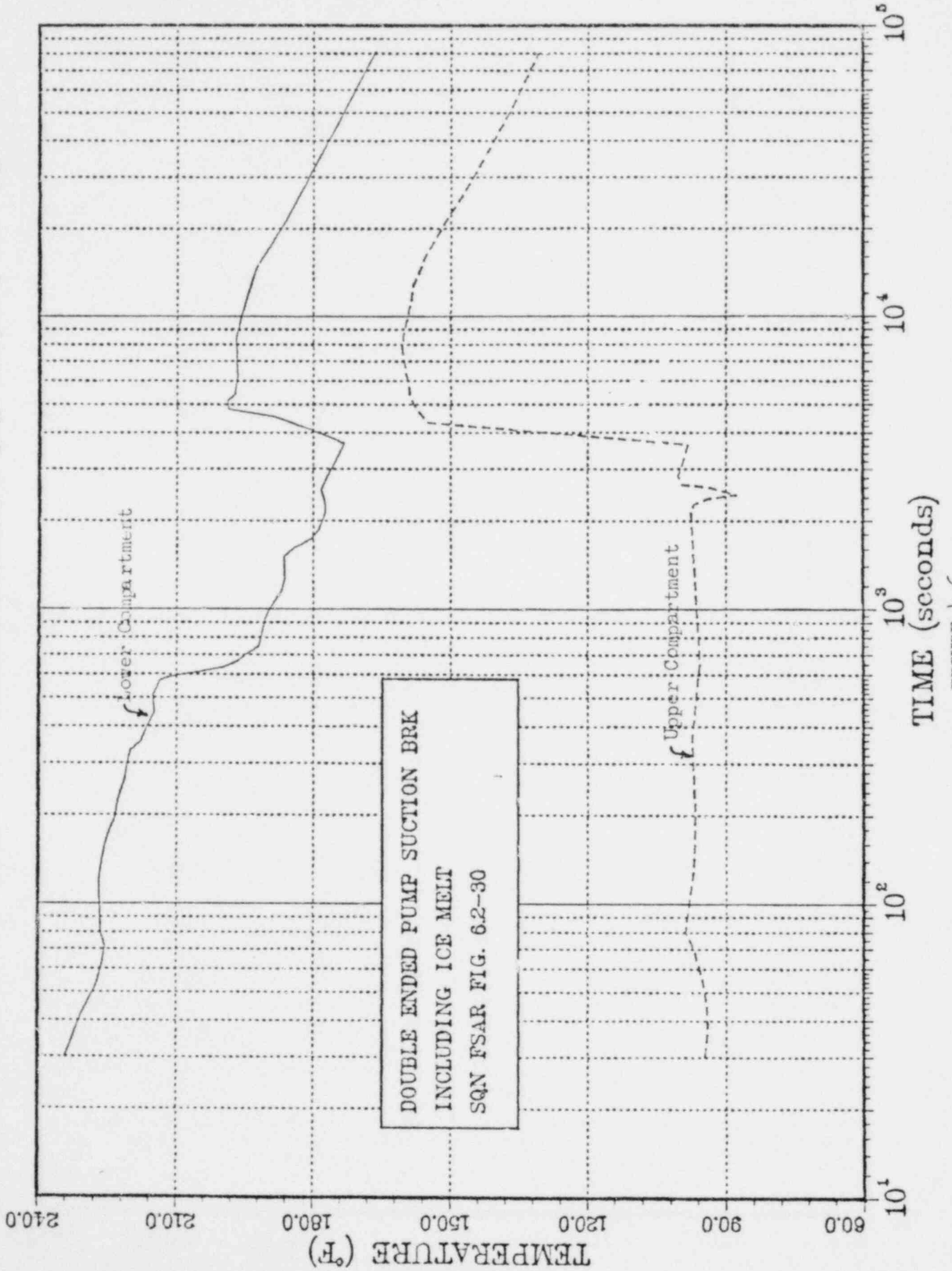
FIGURE 4-6

open pressurizer power operated relief valve which was not recognized
by the operators. The emergency core cooling was severely degraded
by improper operator action which was based on conflicting
instrumentation and training that did not cover the event. The
degraded cooling led to hydrogen generation which exceeded that
normally considered in plant design.

4.3.1 TMI Challenge to Containment Design

Following the accident at Three Mile Island, it was known that a
significant amount of hydrogen was produced and burned, generating
a 28 psig pressure spike in the containment as shown in figure 4-7
(4-4). This pressure spike, had it occurred at Sequoyah, would have
challenged the containment integrity since it exceeded the Sequoyah
design pressure. It is not expected that the containment would have
failed due to the design margin present. There is considerable
uncertainty concerning the actual quantity of hydrogen produced and
burned at TMI. Since the total hydrogen influences the maximum
containment pressure following a burn, the various estimates for
TMI are described below.

Several separate organizations have examined the plant instrumentation
recordings using various approaches to determine the amount of
hydrogen produced during the course of the accident. The original
analysis was performed by the NRC and indicates that 452 pounds of
hydrogen were burned at TMI (4-5). Other organizations more recently
have calculated a wide range of hydrogen burning from 400 pounds to
1160 pounds. Most agree that the data is sparse and not very
conclusive. For example, the containment temperature instrumentation
indicated a temperature increase to a maximum of 190°F during the
hydrogen burn, whereas a theoretical temperature of over 1000°F was
possible. The results of the various investigations are presented
in Table 4-2.

Several indications show that it is possible the hydrogen burn was
localized. This could result in significantly higher temperatures
in localized areas of the containment without affecting the general
atmospheric temperature throughout the volume thus explaining the
temperature dilemma. The presence of steel and concrete heat sinks
may also have absorbed significant amounts of energy.

It is important to note that not all of the hydrogen generated was
burned. Considerable hydrogen remained in the reactor coolant system
both in the hydrogen bubble above the core and dissolved in the
primary system water as can be seen in Table 4-2. Also, not all of
the hydrogen released to the containment atmosphere burned since
approximately 1.7 percent hydrogen by volume remained following the
burn. If the same conditions resulted at Sequoyah, more than 452
pounds of hydrogen could probably be accommodated without containment
failure.

THREE MILE ISLAND PRESSURE RESPONSE

Sequoyah Hydrogen Study

REACTOR BLDG. PRESSURE
TIME AFTER TURBINE TRIP
HYDROGEN BURN

TIME (hours)

FIGURE 4-7

PRESSURE (psig)

4-14
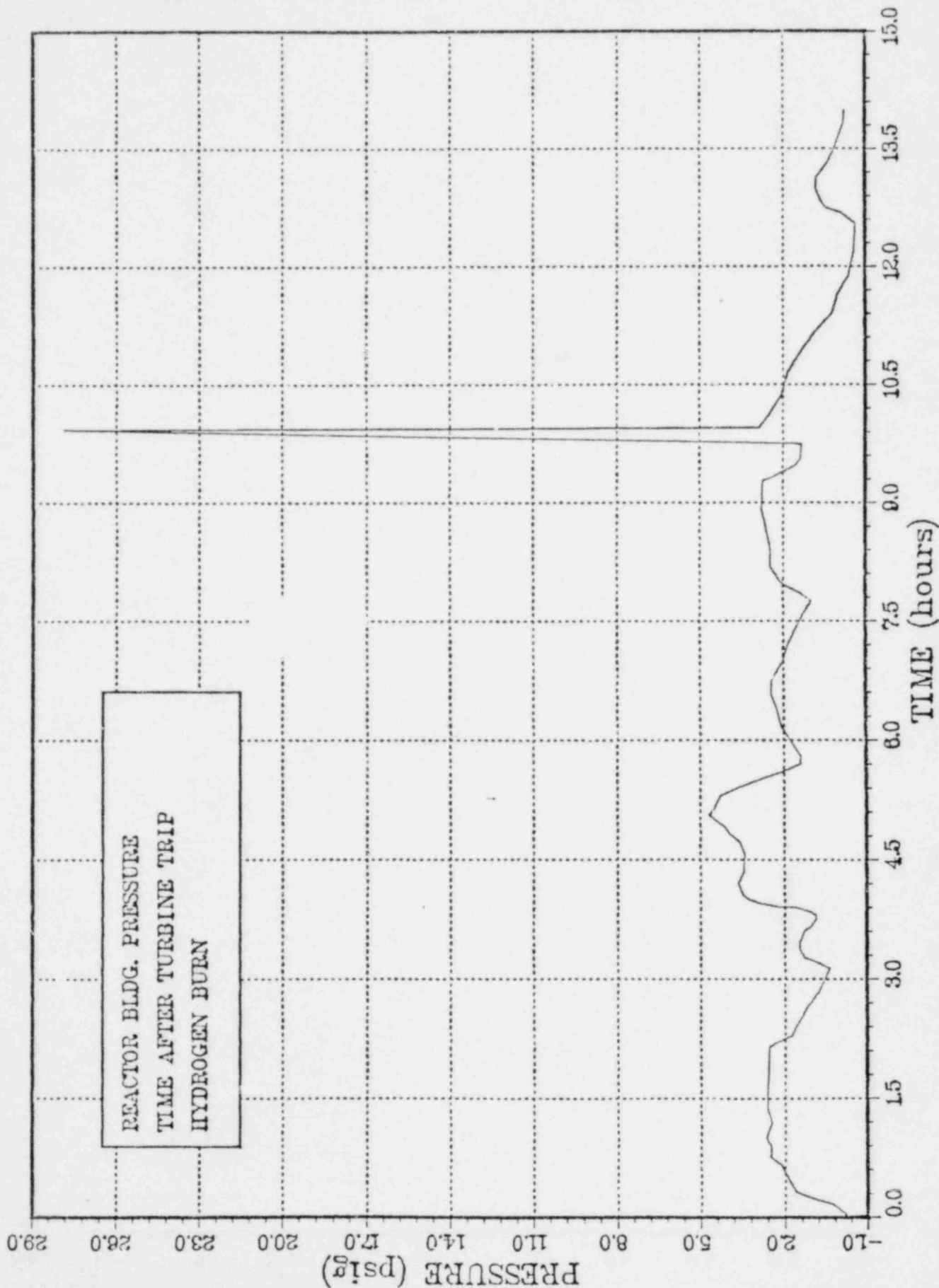
TABLE 4-2

THREE MILE ISLAND HYDROGEN PRODUCTION ESTIMATES

Method	Hydrogen Burned	Hydrogen Produced
Oxygen depletion	400-1160 lb.	670-1286 lb.
ΔP Method	564-588 lb.	1002 lb.
Core Heatup	Not Determined	440-1320 lb.

Details of Above:

Determination by	Type Analysis	H_2 Burned lb.	H_2 in Air (Unburned)	RCS Bubble	RCS Dissolved	H_2 Generated Total
NSAC 1. Preburn	O_2 Depletion	0	680	300	68	1048
2. March 31	"	820	144	172	72	1208
3. April 1-2	"	400	178	20	72	670
4. June 1	"	1080	52	0	2	1134
5. August 2	"	1160	52	0	2	1214
Kemony	ΔP	588	158 ± 8	184 ± 44	72	1002
	O_2 Depletion	872	158	184	72	1286
Sandia 1. Upper	Core Heatup					1320
2. Lower	"					440
3. Best Guess	"					770
Battelle, Columbus	ΔP	564				
	O_2 Depletion	1034				
NRC	Unknown	452				

References: 1. Nuclear Safety Analysis Center, "Analysis of Three Mile Island - Unit 2 Accident," Appendix HYD, Electric Power Research Institute, NSAC-1, July 1979.
2. Robert E. English, "Technical Staff Analysis Report on Chemistry," Technical Assessment Task Force, October 1979.
3. Randall K. Cole, Jr., "Generatio of Hydrogen During the First Three Hours of the Three Mile Island Accident," Sandia Laboratory, NUREG/CR-0913, July 1979.
4. R. O. Wooton, R. S. Denning, P. Cybulskis, "Analysis of the Three Mile Island Accident and Alternative Sequences," Battelle Columbus Laboratories, NUREG/CR-1219, January 1980.

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4-15

Essentially all the investigations that were performed ignore the temperature dilemma and use three basic techniques to determine the hydrogen generation at TMI:

1. Oxygen depletion - Chemical analysis of the containment air space indicates that the oxygen concentration at various times following the accident was less than that normally found in air. By assuming the depletion was caused by the burning of hydrogen, the amount burned can be determined.

2. Delta P - The hydrogen required to generate the pressure spike observed at TMI can be calculated theoretically since the energy released by burning is known.

3. Core heatup - Examination of the thermal hydraulics during the accident can be performed from the few data available to determine the core heatup, cladding oxidation, and hydrogen generation. The exact thermal hydraulic behavior is not well known, but the values generated using this technique agree reasonably well with the other methods.

4.3.2   Consideration of Inadequate Core Cooling

The primary objective during or following any accident is to maintain adequate core cooling so that the core integrity is preserved and most of the radioactivity is maintained within the fuel rods. Additionally, significant hydrogen can only form when the zirconium clad fuel rods are allowed to exceed temperatures of about 2200° F. The best approach in preventing core degradation, which results in the release of radioactivity and hydrogen generation, is to maintain adequate core cooling.

Diagrammatically, the events following a LOCA where adequate core cooling exists are shown in figure 4-8. Dotted lines are included for processes that are not expected to occur. The diagram is modified as shown in figure 4-9 in the case of inadequate core cooling. The problem now includes a significant number of undesirable effects not considered in the design basis analysis. Indeed, the key to the event is therefore the operability of the emergency core cooling systems. These systems have been evaluated conservatively to ensure their effectiveness and have received considerable attention in the past. One change made to the ECCS as a result of the TMI experience was to eliminate their actuation based on low pressurizer water level, thus improving reliability. TVA had identified concerns on its Bellefonte plant that, for certain small loss of coolant accidents where ECCS injection is limited because of the repressurization of the reactor coolant system, the operator might not have sufficient information to identify the appropriate corrective action. The accident at TMI bore out the validity of these concerns. As a result, the automatic status monitoring system which monitors components or systems important to safety, including the ECCS, is being extensively expanded. Other TVA

FIGURE 4-8
LOCA
WITH ADEQUATE CORE COOLING
(See Reference 6-7)

FIGURE 4-9
LOCA
WITH INADEQUATE CORE COOLING
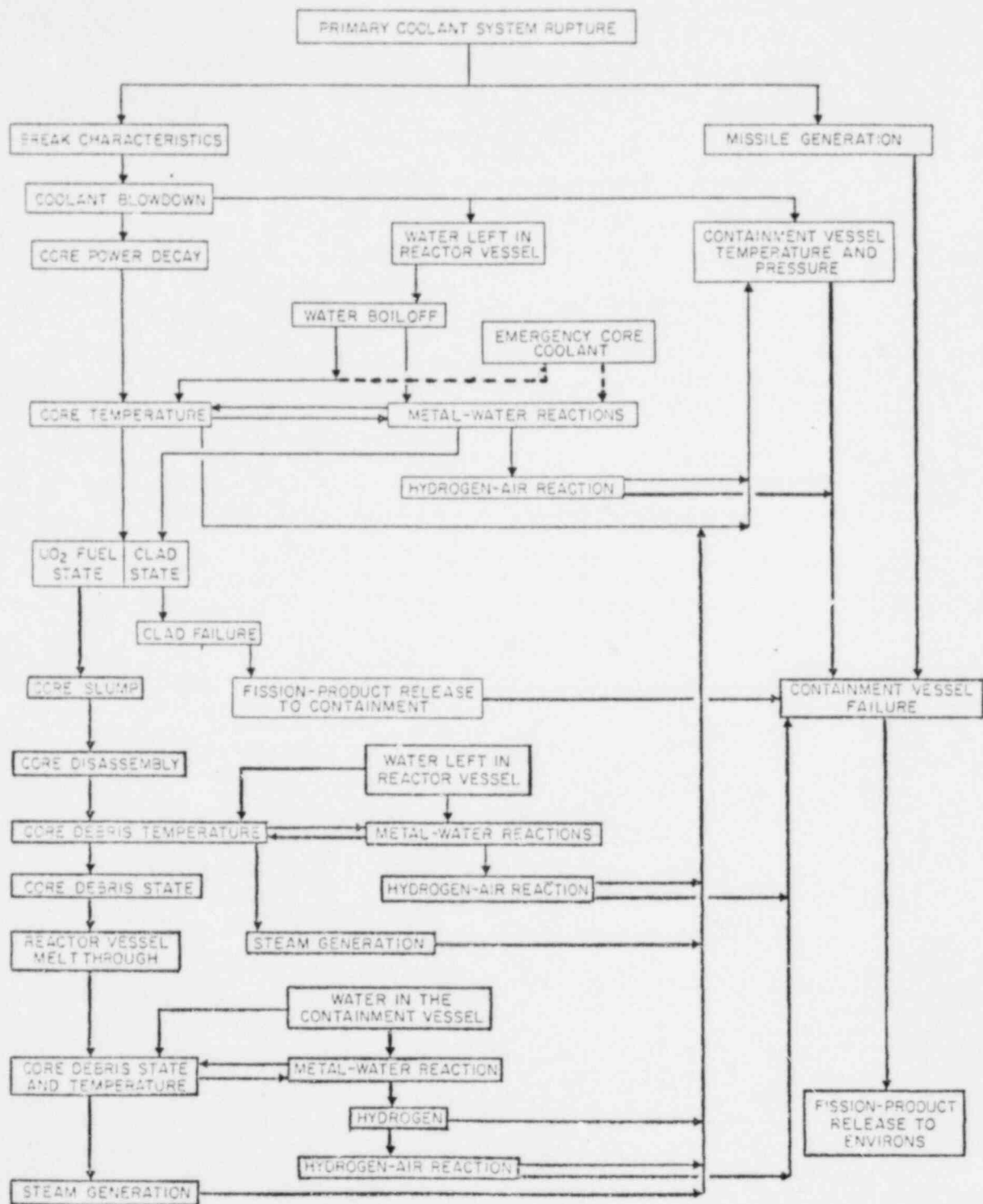
actions to prevent a similar occurrence at Sequoyah are described below.

In the area of plant transient analysis and operator training, it was noted that the TMI operators were relying on procedures that did not adequately address small LOCA's and were relying on instrumentation (pressurizer level) which indicated adequate core cooling even though the core was overheating as indicated by other available instruments. Small LOCA's have not received the extensive analysis attention given larger LOCA's. As a result, the operators were relatively unprepared for the events and conditions that occurred. To assure that this condition does not occur at Sequoyah, TVA is taking the following steps:

1. Improving operating procedures to assure they adequately reflect the design intent.

2. Participating in a joint effort with other utilities to study small LOCA's, inadequate core cooling, and similar events for improved understanding of the phenomena and to determine required mitigative actions.

3. Based upon the studies outlined in 2, additional training will be given to operators to prepare them to recognize the events and take proper mitigative action.

In addition to improved procedures and additional analyses, several hardware and instrumentation changes are planned. It has long been recognized that inadvertent opening of the pressurizer relief valves or safety valves or their failure to close could cause a small LOCA. For this reason, downstream "block" valves were installed on relief valve lines and temperature detectors were provided on the lines to detect an open relief or safety valve. At TMI, the relief valve stuck open, but the operator, for various reasons, did not promptly identify the problem. Sequoyah has direct, positive indication of relief valve position in the control room. This will provide the operator with reliable information to isolate the downstream block valve, terminating a transient if a valve should stick open. (Block valves are not installed on safety valve lines since this could prevent the safety valves from performing their essential function, and safety valves are much less prone to fail since the relief valves provide sufficient capacity for most events, reducing the times the safety valves operate.) TVA is providing acoustic monitors to alert the operator to an open safety valve so that he can take appropriate steps to mitigate the event.

Other hardware changes are planned to ensure that the operator has adequate knowledge that inadequate core cooling may be approaching or existing. These include:

1. The range of the incore thermocouples has been extended to give the operator a positive indication of temperatures in the core region even in an event where the core is inadequately cooled.

2. Instruments have been provided at Sequoyah which will give a direct indication to the operator when the primary system is approaching the boiling point.

3. Instruments to indicate in the main control room the water level in the reactor pressure vessel are being provided at Sequoyah and will be installed by January 1981.

4. Sampling capability of the primary system coolant will be provided. This capability will allow detection of failed fuel (4-6, 4-7).

DA03:NSAHYD.P2

5.0 Risk Analysis of Present Sequoyah Capability for Key Accident Sequences

5.1 General Description

There are several potential accident sequences which could lead to plant damage. Most of these accidents do not involve risk to the public. For this study, the important accidents are ones involving core damage. These events have a relatively low probability of occurrence but have relatively severe consequences. In this study, a risk evaluation of the present Sequoyah design was made by comparison with the reference plant (Surry Nuclear Plant) of the WASH-1400 study. Where applicable, the Surry results have been adjusted to Sequoyah assumptions in order to provide the greatest parity for comparing the results of the plant studies. The results of this risk evaluation provide an indication of the relative risk from the Sequoyah Plant and which accidents should be used as a basis for evaluation of mitigation techniques for degraded core phenomena such as hydrogen evolution, rapid energy releases, etc. These results are discussed in the remainder of this section. For an understanding of the methods used to arrive at risk in WASH-1400 and this study refer to Appendix B.

The dominant accident sequences resulting in core melt and containment failure for the WASH-1400 (Surry) study are shown in table 5.1. Sequoyah dominant sequences are shown in table 5.2. A comparison of the Surry-Sequoyah release category failure probabilities is shown graphically in figure 5-1.

5.2 Analysis Assumptions

The initiating event probabilities used in WASH-1400 for the large and small LOCA's were used in the Sequoyah analysis. The probability of interfacing check valve failure was reduced from $1X10^{-6}$ per reactor year to $1X10^{-8}$ per reactor year at both Surry and Sequoyah to account for improvements in monitoring procedures instituted since the WASH-1400 study was performed. The frequency of loss of offsite power coupled with the failure of the diesel generators to function was taken as $1X10^{-5}$ per reactor year at Sequoyah. The frequency of loss of offsite power at Sequoyah was based on data specific to the Sequoyah plant.

For large LOCA sequences and transient sequences, the probability of containment failure due to a steam explosion was taken to be $1X10^{-3}$ for both Surry and Sequoyah. For small LOCA sequences, the probability of containment failure due to a steam explosion was taken to be $1X10^{-4}$ for both Surry and Sequoyah. WASH-1400 used used $1X10^{-2}$ for the probability of failure due to explosion for Surry, but recent information has indicated the steam explosion risk is much less than believed in 1975. For containment leakage, the probability of containment isolation failure was taken as $1X10^{-2}$ for both Surry and Sequoyah. These assumptions are summarized in table 5-3. Symbols are explained in table 5-4.

For Sequoyah, failure of the reactor protection system scram function

TABLE 5-1

PROBABILITIES OF DOMINANT ACCIDENT SEQUENCES FOR WASH-1400 (ADJUSTED)

RELEASE CATEGORY

Initiating Event	1	2	3	4	5	6	7
A	$AB - \alpha$ $1-10^{-12}$	$AB - \gamma$ $1x10^{-10}$	$AD - \alpha$ $2x10^{-9}$	$ACD - \beta$ $1x10^{-11}$	$AD - \beta$ $4x10^{-9}$	$AB - \varepsilon$ $1x10^{-9}$	$AD - \varepsilon$ $2x10^{-6}$
	$AF - \alpha$ $1x10^{-11}$	$AB - \delta$ $4x10^{-11}$	$AH - \alpha$ $1x10^{-9}$		$AH - \beta$ $3x10^{-9}$	$ADF - \varepsilon$ $1x10^{-10}$	$AH - \varepsilon$ $1x10^{-6}$
	$ACD - \alpha$ $5x10^{-12}$	$AHF - \gamma$ $2x10^{-11}$	$AF - \delta$ $1x10^{-8}$			$AHF - \varepsilon$ $2x10^{-10}$	
	$AG - \alpha$ $9x10^{-12}$		$AG - \delta$ $9x10^{-9}$				
	$2.7x10^{-10}$	$2.4x10^{-9}$	$2.2x10^{-8}$	$5.9x10^{-9}$	$3.8x10^{-8}$	$3.0x10^{-7}$	$3.0x10^{-6}$
S_1	$S_1B - \alpha$ $3x10^{-13}$	$S_1B - \gamma$ $4x10^{-10}$	$S_1D - \alpha$ $3x10^{-10}$	$S_1CD - \beta$ $1x10^{-11}$	$S_1H - \beta$ $5x10^{-9}$	$S_1DF - \varepsilon$ $3x10^{-10}$	$S_1D - \varepsilon$ $3x10^{-6}$
	$S_1CD - \alpha$ $7x10^{-13}$	$S_1B - \delta$ $1x10^{-10}$	$S_1H - \alpha$ $3x10^{-10}$		$S_1D - \beta$ $6x10^{-9}$	$S_1B - \varepsilon$ $2x10^{-9}$	$S_1H - \varepsilon$ $3x10^{-6}$
	$S_1F - \alpha$ $3x10^{-12}$	$S_1HF - \gamma$ $6x10^{-11}$	$S_1F - \delta$ $3x10^{-8}$			$S_1HF - \varepsilon$ $4x10^{-10}$	
	$S_1G - \alpha$ $3x10^{-12}$		$S_1G - \delta$ $3x10^{-8}$				
	$6.9x10^{-10}$	$6.8x10^{-9}$	$6.1x10^{-8}$	$1.2x10^{-8}$	$6.2x10^{-8}$	$6.0x10^{-7}$	$6.0x10^{-6}$
S_2	$S_2B - \alpha$ $1x10^{-12}$	$S_2B - \gamma$ $1x10^{-9}$	$S_2D - \alpha$ $9x10^{-10}$	$S_2DG - \beta$ $1x10^{-12}$	$S_2D - \beta$ $2x10^{-8}$	$S_2B - \varepsilon$ $8x10^{-9}$	$S_2D - \varepsilon$ $9x10^{-6}$
	$S_2F - \alpha$ $1x10^{-11}$	$S_2HF - \gamma$ $2x10^{-10}$	$S_2 - \alpha$ $6x10^{-10}$		$S_2H - \beta$ $1x10^{-8}$	$S_2CD - \varepsilon$ $2x10^{-8}$	$S_2H - \varepsilon$ $6x10^{-6}$
	$S_2DC - \alpha$ $2x10^{-12}$	$S_2B - \delta$ $4x10^{-10}$	$S_2F - \delta$ $1x10^{-7}$			$S_2HF - \varepsilon$ $1x10^{-9}$	
	$S_2G - \alpha$ $9x10^{-12}$		$S_2C - \delta$ $2x10^{-6}$				
	$S_2C - \alpha$ $2x10^{-10}$		$S_2G - \delta$ $9x10^{-8}$				
	$2.3x10^{-8}$	$2.3x10^{-7}$	$2.3x10^{-6}$	$2.5x10^{-7}$	$2.5x10^{-7}$	$2.0x10^{-6}$	$2.0x10^{-5}$

TABLE 5-1 (Continued)

PROBABILITIES OF DOMINANT ACCIDENT
SEQUENCES FOR WASH-1400 (ADJUSTED)

RELEASE CATEGORY

Initiating Event	1	2	3	4	5	6	7
T	$TMLB'-\alpha$ 3×10^{-10}	$TMLB'-\gamma$ 7×10^{-7}	$TML-\alpha$ 6×10^{-10}		$TML-\beta$ 3×10^{-10}	$TMLB'-\varepsilon$ 6×10^{-7}	$TML-\varepsilon$ 6×10^{-6}
		$TMLB'-\delta$ 2×10^{-6}	$TKQ-\alpha$ 3×10^{-10}		$TKQ-\beta$ 3×10^{-10}		$TKQ-\varepsilon$ 3×10^{-6}
			$TKMQ-\alpha$ 1×10^{-10}				$TKMQ-\varepsilon$ 1×10^{-6}
	3.0×10^{-7}	3.0×10^{-6}	3.0×10^{-7}	4.0×10^{-8}	1.0×10^{-7}	1.0×10^{-6}	1.0×10^{-5}
R	$RC-\alpha$ 2×10^{-13}	$RC-\gamma$ 3×10^{-11}	$R-\alpha$ 1×10^{-10}				$R-\varepsilon$ 1×10^{-7}
		$RF-\delta$ 1×10^{-11}					
		$RC-\delta$ 1×10^{-12}					
	5.2×10^{-12}	5.0×10^{-11}	1.1×10^{-10}	1.1×10^{-10}	1.0×10^{-9}	1.0×10^{-8}	1.0×10^{-7}
V		V 4×10^{-8}					
	4.0×10^{-9}	4.0×10^{-8}	4.0×10^{-9}	4.0×10^{-10}			
TOTAL	3.3×10^{-7}	3.3×10^{-6}	2.7×10^{-6}	3.1×10^{-7}	4.5×10^{-7}	4.5×10^{-6}	3.9×10^{-5}

TABLE 5-2

PROBABILITIES OF DOMINANT ACCIDENT SEQUENCES FOR SEQUOYAH

Release Categories

Initiating Event	1	2	3	4	5
A	$AG - \alpha$ 6.0×10^{-11}	$AHF - \gamma,\delta$ 3.0×10^{-8}	$AG - \gamma,\delta$ 6.0×10^{-8}		$AH - \gamma,\delta$ 4.6×10^{-7}
	$AHF - \alpha$ 3.0×10^{-11}				$AD - \gamma,\delta$ 1.9×10^{-7}
	4.0×10^{-9}	3.9×10^{-8}	7.0×10^{-8}	7.3×10^{-8}	6.7×10^{-7}
S_1	$S_1G - \alpha$ 1.8×10^{-11}	$S_1HF - \gamma,\delta$ 9.0×10^{-8}	$S_1G - \gamma,\delta$ 1.8×10^{-7}	$S_1FG - \beta$ 5.9×10^{-12}	$S_1D - \gamma,\delta$ 1.05×10^{-6}
	$S_1HF - \alpha$ 9.0×10^{-12}			$S_1CFG - \beta$ 3.1×10^{-12}	$S_1H - \gamma,\delta$ 4.8×10^{-7}
	1.1×10^{-8}	1.1×10^{-7}	2.0×10^{-7}	1.7×10^{-7}	1.5×10^{-6}
S_2	$S_2G - \alpha$ 6.0×10^{-11}	$S_2HF - \gamma,\delta$ 3.0×10^{-7}	$S_2G - \gamma,\delta$ 6.0×10^{-7}	$S_2CDF - \beta$ 5.9×10^{-11}	$S_2D - \gamma,\delta$ 3.5×10^{-6}
	$S_2HF - \alpha$ 3.0×10^{-11}	$S_2DF - \gamma,\delta$ 1.2×10^{-8}		$S_2CHF - \beta$ 3.2×10^{-11}	$S_2H - \gamma,\delta$ 1.6×10^{-6}
	$S_2CDF - \alpha$ 5.9×10^{-13}	$S_2CDF - \gamma,\delta$ 5.9×10^{-9}		$S_2FG - \beta$ 2.0×10^{-11}	$S_2L - \gamma,\delta$ 3.0×10^{-7}
		$S_2CHF - \gamma,\delta$ 3.2×10^{-9}		$S_2CFG - \beta$ 1.0×10^{-11}	$S_2D - \beta$ 3.5×10^{-8}
		$S_2HF - \beta$ 3.0×10^{-9}			
	4.0×10^{-8}	3.9×10^{-7}	6.9×10^{-7}	6.1×10^{-7}	6.1×10^{-6}
		$T_B B_2 - \gamma,\delta$ 1.0×10^{-6}			
	$T_B B_2 - \alpha$ 1.0×10^{-10}	$T_A L - \gamma,\delta$ 8.0×10^{-9}		$T_B B_2 - \beta$ 1.0×10^{-8}	
T		$T_B LDO - \gamma,\delta$ 2.4×10^{-9}			

TABLE 5-2 (Continued)

PROBABILITIES OF DOMINANT ACCIDENT
SEQUENCES FOR SEQUOYAH

Initiating Event	Release Categories				
	1	2	3	4	5
				$T_A L - \beta$	$T_B LD - \gamma, \delta$
				8.0×10^{-11}	1.0×10^{-6}
				$T_B LDO - \beta$	$T_B QD - \gamma, \delta$
				2.4×10^{-11}	5.4×10^{-7}
	1.0×10^{-7}	1.0×10^{-6}	1.2×10^{-7}	1.8×10^{-7}	1.6×10^{-7}
V		V 4×10^{-8}			
	4×10^{-9}	4×10^{-8}	4×10^{-9}	4×10^{-10}	
TOTAL	1.6×10^{-7}	1.5×10^{-6}	1.1×10^{-6}	1.0×10^{-6}	9.9×10^{-6}

RISK COMPARISON
SURRY VS. SEQUOYAH
FIGURE 5-1

Figure 5-1: Risk Comparison Surry vs. Sequoyah

□ WASH-1400 (ADJUSTED)

▨ SEQUOYAH
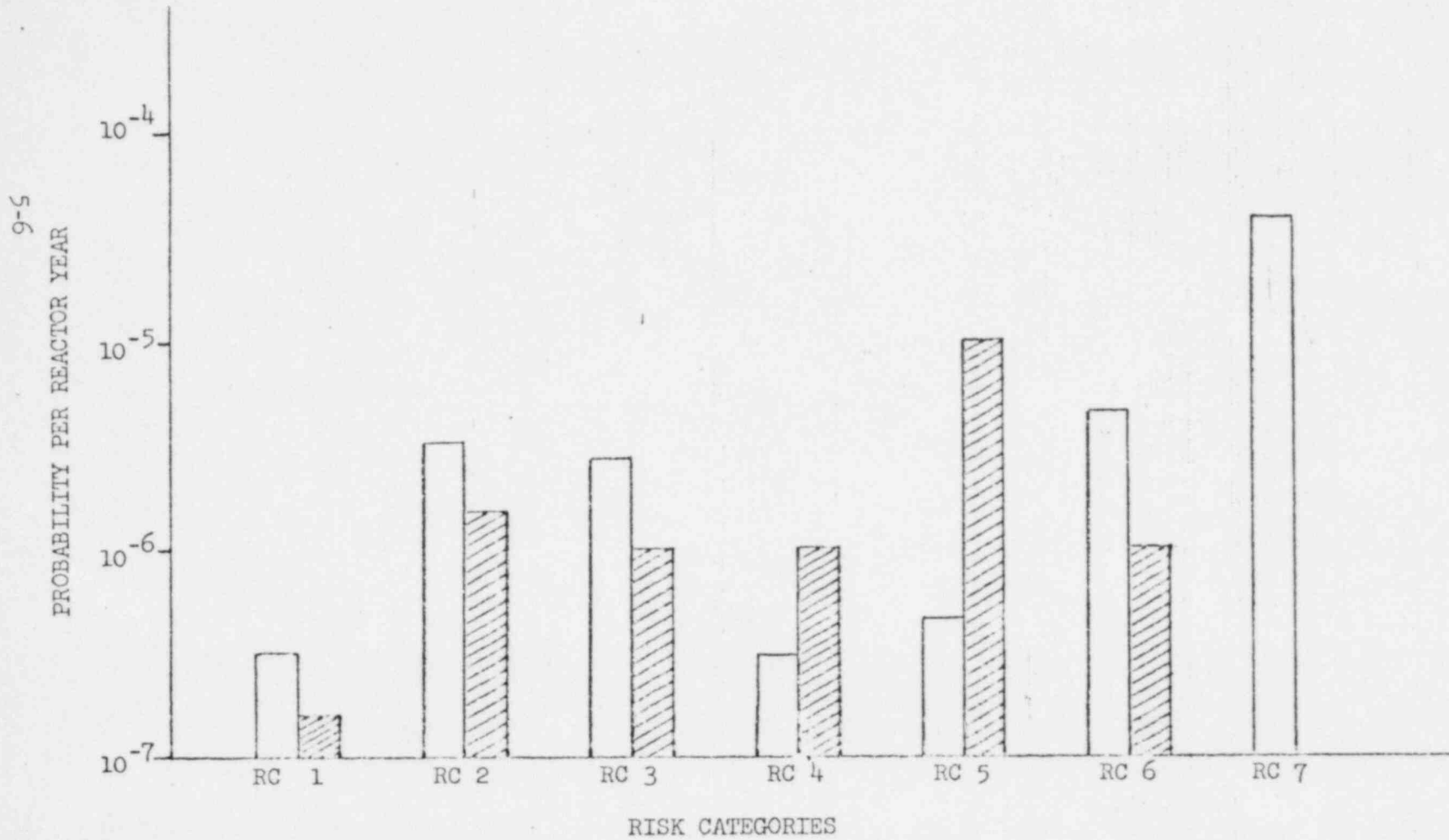
TABLE 5-3

Failure Probabilities Used for the Sequoyah Analysis

	A	S_1	S_2	T_A	T_B
IE	1.0×10^{-4}	3.0×10^{-4}	1.0×10^{-3}	1.0×10^{-6}	3*
B	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}		
B1					1.0×10^{-6}
B2				1.0×10^{-6}	1.0×10^{-5}
C	1.7×10^{-5}	1.7×10^{-5}	1.7×10^{-5}		
D	1.9×10^{-3}	3.5×10^{-3}	3.5×10^{-3}		3.5×10^{-3}
E	1.0×10^{-4}				
F	3.3×10^{-3}	3.3×10^{-3}	3.3×10^{-3}		
G	6.0×10^{-4}	6.0×10^{-4}	6.0×10^{-4}		
H	4.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}		
K		3.5×10^{-5}	3.5×10^{-5}	3.5×10^{-5}	3.5×10^{-5}
L			3.0×10^{-4}	8.0×10^{-3}	1.0×10^{-4}
M				1.0×10^{-6}	1.0×10^{-6}
O				2.3×10^{-3}	2.3×10^{-3}
P				4.3×10^{-5}	4.3×10^{-5}
Q					5.2×10^{-5}
X	1.8×10^{-3}	1.8×10^{-3}	1.8×10^{-3}		
Z	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}		
CF**	1.7×10^{-3}	1.7×10^{-3}	1.7×10^{-3}		
HF**	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}		
CHF**	8.3×10^{-6}	3.2×10^{-6}	3.2×10^{-6}		

TABLE 5-3 (Contin.)

Containment failure modes

α	1.0×10^{-3}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}
β	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}	1.0×10^{-2}
γ, δ	.99	.99	.99	.99	.99

*For the $T_B B_2$ sequences, $T_B = 1.0$

**CF, CHF, and HF are common mode failures whose probabilities of occurrence are unique and are not independent.

TABLE 5-4

Explanation of Symbols

A	-	Large LOCA ($> 6"$)
IE	-	Initiating Event
B	-	Loss of all electric power
B_1	-	Loss of direct current power
B_2	-	Loss of alternating current power
C	-	Containment spray injection
D	-	Emergency core injection
E	-	Emergency cooling functionability
F	-	Containment spray recirculation
G	-	Containment heat removal (ERCW, CCW)
H	-	Emergency core recirculation
K	-	Reactor protection system
L	-	Auxiliary Feedwater System
M	-	Secondary side relief valves
O	-	Containment emergency safety features (C, F, G)
P	-	Primary side relief valves (fail to open)
Q	-	Primary side relief valves (fail to close)
R	-	Reactor pressure vessel rupture
S_1	-	Small LOCA ($6" \leq x < 2"$)
S_2	-	Small LOCA ($\leq 2"$)
TA	-	Transient, loss of all ac power
TB	-	Transient, loss of main feedwater system
V	-	Interfacing check valve failure
X	-	Air return fans
Z	-	Ice condenser
α	-	Containment failure due to steam explosion
β	-	Containment failure due to containment leakage
γ	-	Containment failure due to hydrogen burn
δ	-	Containment failure due to overpressurization
ϵ	-	Containment failure due to meltthrough of the containment basemat

was not assumed to initiate a core melt sequence based on analyses performed by Westinghouse.

5.3 Dominant Sequences

The more important Sequoyah accident sequences from a risk standpoint were found to be:

1. A loss of main feedwater transient (T) followed by failure of the dc power system (B_1) followed by containment failure due to hydrogen burning or overpressurization.

2. Small LOCA's (S_2) followed by failure of the containmant heat removal system (G) followed by containment failure due to hydrogen burning or overpressurization.

3. Small LOCA's (S_2) followed by failure of the emergency core cooling recirculation system (H) and the containment spray recirculation (F) followed by containment failure due to hydrogen burning or overpressurization.

4. Small LOCA's (S_2) followed by failure of the emergency core core injection system (D) followed by containment failure due to hydrogen burning or overpressurization.

5. Small LOCA's (S_2) followed by failure of the emergency core recirculation system (H) followed by containment failure due to hydrogen burning or overpressurization.

5.4 Sequoyah and Surry Comparisons

A comparison of Sequoyah to Surry indicates that the Sequoyah accident sequences are relatively less probable for high consequence categories (risk categories 1, 2, and 3) and relatively more probable for low consequence categories. The major factors in this difference are the containment failure probabilities for hydrogen burn and overpressurization. Sequoyah has an ice condenser and, as such, has a smaller free volume than the Surry dry containment. Similarly, the Surry containment is capable of withstanding a higher internal pressure than is Sequoyah's. The Surry assumption for probability of containment failure given a hydrogen burn or overpressurization varied from 0.1 to 0.9. The Sequoyah evaluation assumed a containment failure probability of 0.99 for sequences including hydrogen burn or overpressurization. The hydrogen burn and the overpressurization are the dominant containment failure modes in release categories 4 and 5 and thus have the largest single influence on the probabilities in this category being higher than for Surry.

The Sequoyah analyses of the dominant sequences and their contribution to the overall risk were performed based on the results of a limited number of detailed evaluations of accident sequences and other qualitative studies.

The analysis indicated that the risk associated with Sequoyah is similar to the risk associated with the Surry plant. The ice condenser containment does not appear in this abbreviated study to be either significantly better or worse than the dry containment for either design basis events or events that result in core degradation.

DA02;CUTCY.01

6.0 Accidents Beyond the Design Bases

The accidents considered by this report for the Sequoyah plant involve changes from design basis considerations in the area of: different phenomenology inherent in the accident; modifications to the analytical tools because of the new phenomena present; analyses of the plant response to more severe events; and an assessment of the risk present during degraded scenarios. Each of these is discussed for the events identified as key during the risk assessment.

6.1 Phenomenological Consequences of Key Accident Scenarios

The key accident sequences that have been identified result in several phenomena which are more severe than previously considered in design basis events. These phenomena include production of noncondensible gas significantly beyond that normally considered in plant design, the possibility of hydrogen deflagration (burning), and the possibility of hydrogen detonation (explosion). In addition, the energy release rates for sequences ending in core melt are drastically different. Large release rates occur late in the transient, and, in some cases, they exceed the rates during the blowdown period at the beginning of the accident. The studies performed for this report consider two types of scenarios: accidents leading to large quantities of hydrogen; and accidents leading to large quantities of hydrogen and followed by total core melt with the subsequent production of concrete decomposition gases.

The basic sequences chosen for additional consideration are: (1) a large pipe break in the primary coolant system with a concurrent loss of all active emergency core cooling systems (ECCS); (2) a small reactor coolant system break with a concurrent loss of high pressure injection; (3) a small reactor coolant system break with the loss of all ECCS due to an inability to recirculate water from the emergency sump after the refueling water storage tank has been emptied; and (4) a transient caused by the loss of the main condenser, loss of all auxiliary feedwater, and loss of high pressure injection. Using the designation provided in the Reactor Safety Study, WASH-1400, Case 1, is designated AD, Case 2 is S_2D, Case 3 is a modification of the S_2D sequence, and Cass 4 was selected by TVA due to the probability of occurrence. The mass and energy releases and hydrogen generation rates for Case 4 are essentially the same as the event WASH-1400 designated TMLB' which is a loss of all ac power. Each of these events, if allowed to proceed unchecked, results in a meltdown of the reactor core. A wide time span is covered by the choice of these events. The large break proceeds to core melt very rapidly while in the Case 4 transient; core melt does not occur for several hours after the initiation of the event.

6.1.1 Hydrogen and Other Noncondensible Gas Production

Hydrogen gas is generated in the core melt sequences from both cladding oxidation and vessel oxidation. In an inadequate core cooling event (AD, S_2D, TMLB', etc.), the cladding oxidizes rapidly due to the high temperatures present until sufficient steam

is no longer available for the reaction. Various researchers estimate as much as 60-80 percent of the core may oxidize before melting occurs. The formation of eutectics between the cladding and the fuel complicates the physics of the process and leads to uncertainties in the total production. Reactions of this magnitude would produce sufficient hydrogen to approach explosive concentrations in the containment at Sequoyah (approximately 1200-1800 pounds hydrogen).

Core melt gases are also generated by the decomposition of limestone in the presence of the molten steel, fuel, cladding, and control rod material from the core. Both carbon dioxide and the flammable carbon monoxide are produced during the decomposition process. Decay heat provides energy to maintain the core debris at elevated temperatures during this period.

6.1.2 Hydrogen Combustion

Hydrogen is a flammable gas that when oxidized can release large amounts of energy. For each pound burned, approximately 61,084 Btu of heat is produced. Hydrogen can burn in two ways, depending on the concentration present. The first is known as deflagration which is a slow burn characterized by low velocities of flame propagation. The deflagration concentration is approximately 4 percent hydrogen by volume. Above this concentration, the hydrogen will burn upward from the point of ignition due to the buoyancy forces present as the hot gases evolve during the burn. Once concentrations of 8 percent and above are reached, the flame will propagate in both upward and downward directions due to the turbulence present. It is estimated that 50 percent of the hydrogen will burn in a 5.6 percent hydrogen mixture, whereas nearly 100 percent of the hydrogen may be consumed in the deflagration of a 10 percent hydrogen mixture. A flammability limit chart for hydrogen, air, and steam mixtures is shown in figure 6-1. It should be noted that the initial temperature and pressure of the mixture affects the flammability limits.

Once a concentration of 18 percent is reached, the potential for detonation or explosion exists. The reaction begins as a deflagration but transitions to a detonation when shock waves, formed during the turbulent burning, reflect from containment walls and reinforce the burning process until the reaction proceeds at the speed of sound or above in the mixture. Substantial energy can exist in the shock wave which impacts surfaces in the containment. Fortunately, large metal-water reactions are required before this volumetric concentration limit is reached at Sequoyah.

Several modes of burning and their impact on the Sequoyah design are discussed below:

1. Slow Burning During Release

    The hydrogen can burn in air if both hydrogen and oxygen have sufficiently high concentrations (greater than 4-6 percent for hydrogen and 5 percent for oxygen). Disregarding the dilution effect of steam, at least 12-18 percent zirconium water reaction is required to reach this high hydrogen concentration.

FIGURE 6-1

Figure 6-1
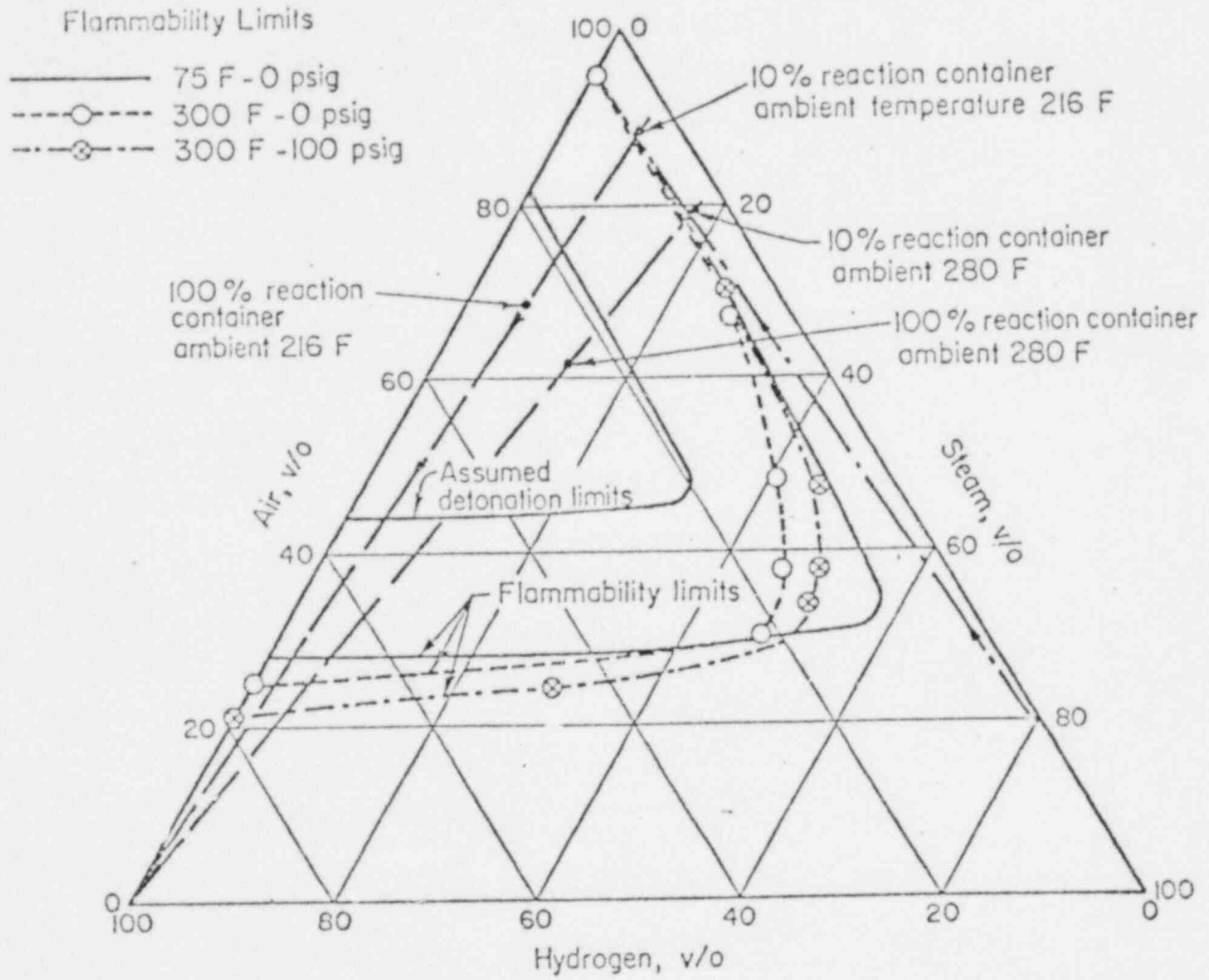
If the hydrogen is reseased slowly from the reactor coolant system (over a period of about 30 minutes for a 25 percent metal-water reaction) and is burned as released or periodically burned as the containment hydrogen concentration exceeds 4 percent, the resulting heat released to the containment could be removed through the containment sprays and heat exchangers and by transfer of heat to the structural heat sinks. Burning the amount of hydrogen estimated to have been burned at TMI would add less than 30 million Btu's to the containment. This amount could be easily handled in a 30-minute period by the containment sprays and heat sinks without a significant increase in containment pressure.

2. Combustion of Hydrogen (Deflagration) After Accumulation

Energy released by burning hydrogen is deposited in the containment atmosphere. If released over a short period of time, the temperature of the air in containment will be greatly increased, resulting in increased containment pressure. The containment temperature and pressure then drop as heat is transferred to metal and concrete in the containment and also removed by the containment spray system.

3. Sporadic, Localized Hydrogen Burning or Detonation

If hydrogen concentrates in localized areas of containment and burns/detonates locally on a periodic basis, a condition somewhere between 2 and 4 results where small pressure/temperature spikes occur, decaying to a steady state until the next event. This condition might occur if there is inadequate mixing, or if a periodic or continuous ignition source exists.

4. Hydrogen Detonation

If hydrogen is available in sufficient concentration (15-20 percent) with air, a detonation may occur. A detonation is characterized by extremely rapid combustion which takes place within a high velocity shock wave. The possibility of occurrence and the nature of a detonation is affected by geometry, presence of ignition sources and diluting media (such as steam). At TMI, at least a rapid combustion (deflagration) occurred. The exact effect of a detonation on the Sequoyah containment is unknown. Since it would depend greatly on the location and total quantity of hydrogen involved. The potential exists for containment overstressing.

6.1.3 Core Melt Energy Consideration Due to Steam

There are three basic periods of steam generation to consider in degraded core accident sequences. These periods are: primary system blowdown; core slump into water remaining in the bottom of the reactor vessel; and reactor vessel melt through into water in the lower reactor cavity. The blowdown period mass and energy releases for the large or small break sequences are essentially the same for both design basis and extended design basis accident sequences. These

releases are controlled by the break size and the initial reactor coolant system pressure and temperature. Active emergency core cooling systems have little impact on this portion of the mass and energy releases. For a transient case such TMLB', the blowdown period occurs in intermittent steps. The reactor coolant system heats up and in the process increases in pressure until the relief valves on the pressurizer open. The loss of mass out the relief lines drops the reactor vessel temperature and pressure until the valves close and then the process begins again. Decay heat is the driving mechanism for this event. The blowdown period for a large break lasts around 30 seconds. Small break blowdowns last longer, but the initial rapid depressurization of the reactor coolant system occurs in a relatively short period of time. The blowdown portion of the TMLB' event lasts several hours.

Based on the MARCH calculations described in section 6.2.2, core slump into the bottom of the vessel occurs in 1200 seconds for a large break LOCA, 7000 seconds for a small break LOCA , and 13,400 seconds for a TMLB' transient. Only small amounts of water remain in the bottom of the reactor vessel at the time of core slump and containment conditions are not appreciably affected due to this part of the transient. Shortly after core slump, reactor vessel meltthrough occurs. When the molten mass of fuel, reactor internals, and reactor vessel metal falls into the pool of water standing in the reactor cavity, large quantities of steam are generated. The energy release rates due to vessel meltthrough are on the same order of magnitude as a large break blowdown. This energy addition was considered in the present study.

Steam explosions, while a potential source of large amounts of energy, were not considered. The physical phenomena associated with such explosions are complex and are not yet completely understood. Minimal experimental data exists, particularly as it relates to the behavior of molten UO_2 (fuel) in water. The WASH-1400 conclusion that the probability of a UO_2-water reaction resulting in a steam explosion causing containment failure was low has been used as a basis for not considering the event until there is more understanding of the processes involved. Recent studies also support this conclusion.

6.2 Analytical Tools

Section 4.2.2 discussed analytical tools used to evaluate current design basis accidents. The following is a discussion of the techniques which are available for analyzing events beyond the design basis.

6.2.1 Limitations of Present Computer Codes

Computational software is not yet available to accurately model the dynamics of severely degraded accidents. For example, primary system codes do not model multicomponent mixtures, nor do containment codes consider concentrations of individual gas species where the species concentration is modified by chemical reaction as in the case of hydrogen burning. Most primary system and containment computer codes have provisions for modeling some of these processes, but not in a

best estimate manner. Some of these limitations are discussed below, since they influenced techniques used in this study to approximate the accident behaviors.

6.2.1.1 Primary System Analysis

The analysis of the key accident sequences selected is considerably more complex than the analysis of present design basis accidents since the computational tools for the severely degraded accidents are not yet fully developed. As mentioned previously, the current primary system thermal hydraulic codes do not model multicomponent two phase flow and therefore cannot track the evolution and distribution of significant quantities of hydrogen and other noncondensibles in the primary system. For the degraded conditions that result in the key sequences described, the hydrogen production could be large, and in the case of small break accidents may remain in the primary system for extended periods of time.

Provided the hydrogen gas does become trapped in the primary system, redistribution of the gas may occur such that heat transfer in the steam generators is influenced. Heat transfer is impaired by mechanisms associated with the presence of the noncondensible gas in the condensation boundary layer located at the steam generator tube surface. For events where steam generator function has already been lost or impaired, this will be of little consequence. Alterations in the primary system void fraction distribution and flow regimes are also possible, depending on the quantities of noncondensible gas present. Both characteristics of the flow have the potential to reduce core heat transfer. Modifications to the advanced thermal hydraulic code TRAC (6-1) are underway to add a noncondensible gas field for the study of these effects. Appropriate changes to the metal-water reaction model to include the production of hydrogen and consumption of steam by the oxidation process will be necessary. TVA has also been encouraging the Electric Power Research Institute to modify their RETRAN (6-2) code to include these effects through our participation in the RETRAN Utility working group. The extension is being considered for a future version.

Unfortunately, a best estimate hydrogen release for containment studies was therefore not available. Hydrogen production rates from the MARCH code were used instead with the assumption that the break size was large enough to transfer all hydrogen from the primary system to the containment as it was produced.

6.2.1.2 Containment Codes

Accidents that involve a degraded core require the consideration of many factors that are also not presently included in the containment codes. These factors include the addition of large quantities of noncondensibles at very high temperatures to the containment atmosphere, the addition of energy due to hydrogen or carbon monoxide burning, and the change in masses of atmospheric constituents due to burning (i.e., $2H_2 + O_2 = 2H_2O$). Experiments which determine ice condenser performance parameters have not been performed for the case of very high temperatures and large quantities of noncondensibles

associated with degraded core scenarios. When burning occurs, containment temperatures exceed the limits of the table functions that existing codes use to evaluate the properties of steam.

Modifications to the codes are also required to allow modeling of features used in various degraded core mitigation concepts. Features such as opening or closing a vent as a function of time, and calculation of pressure drops in gravel bed filtration systems are not presently in the available codes.

6.2.2. The MARCH Computer Code

The Battelle Columbus Laboratories are currently developing a computer tool to study light water reactor meltdown accident response characteristics (MARCH) that overcomes limitations of the primary system and containment codes described above (6-3). Although considerable work has been expended on the understanding of breeder reactor hydrodynamic disassembly and energetic meltdown accident characteristics, as exemplified by the computer codes SAS3A (6-4), and SIMMER (6-5) used in breeder accident modeling; comparatively little effort has been expended on the development of light water reactor (LWR) meltdown codes. Alternatively, light water reactor research has concentrated on prevention and mitigation of accidents before the comparatively benign meltdown occurs, thereby reducing the risk of core melt to an extremely low level. However, the experience gained in the breeder studies may be applied to the LWR meltdown accident. In addition, some LWR specific reseach has been done and several codes have been used to study aspects of the meltdown problem. NURLOC (6-6,6-7) and CHEMLOC (6-8,6-7) are two such codes used to model the fuel rod heatup, oxidation, and melting process. Also the BOIL computer code was written for the reactor safety study (WASH-1400) to determine the time required for fuel failure, melting, and reactor vessel meltthrough. BOIL and a concrete interacton code, INTER, are immediate predecessors to MARCH and are employed as subroutines within the code.

The MARCH computer models contain sufficient detail to follow an accident into the phase where considerable hydrogen is produced by cladding heatup and reaction. Models permit parametric studies of core melt encompassing slumping of the fuel bundles followed by pressure vessel meltthrough. Although the thermal hydraulic models do cover most of the accident physics, they are not as detailed as the models in the RELAP (6-10) blowdown code and other vendor codes used for less severe accidents. No comparisons have been made to BEACON (6-9), the current best estimate containment code, or the RELAP primary system code. MARCH functions in many ways as a simplified coupling of these two codes with the necessary extensions to incorporate core melt. Because of the lack of comparisons for MARCH results and the fact that the code has not been made publicly available by its authors, the conservatism present in the code is not known. Several of the meltdown models may be nonmechanistic and

therefore may not reflect the actual core behavior. Examination of the code details will be required in future efforts before a final design is produced based on MARCH hydrogen production estimates. Areas needing further development or experimentation include: the melting and slumping of the reactor core in the vessel, the degree of metal-water reaction during the core melt process (both invessel and exvessel), the amount of core internal structure and vessel material that participates in the melt, and the formation, quenching, and coolability of the debris bed. The MARCH subroutines are described in Appendix C.

6.2.3 Physical Analysis of Present Sequoyah Capability for Accident Sequences

The accident sequences discussed in section 6.1 have been applied to the Sequoyah design to determine the design margins available in the containment. For events where the capacity of the containment is exceeded, mitigating devices are examined to determine this effectiveness in ensuring containment integrity. The containment analyses have been based on studies done by Battelle Columbus Laboratories, using the MARCH computer code, for a Sandia study of Sequoyah. The noncondensible gas production is discussed followed by the expected containment response in the following section.

6.2.3.1 MARCH Noncondensibles

The MARCH computer code has been applied to several key sequences used in this report. The computer results have been provided by Battelle via Westinghouse and were used in this study because of the limitations of other codes available to TVA at the time of this study. MARCH is not presently released for public use so that data was limited to the studies provided by the indirect channel.

Steam production for these cases is shown for the AD, S_2D, and TMLB' accidents in figures (6-2 through 6-4). The hydrogen generation is shown in figures 6-5 through 6-7. Since the cases all lead to complete core melt, both carbon dioxide and carbon monoxide evolve from concrete interaction with the molten core debris. These gases are plotted in figures 6-8 through 6-10, where the time for their evolution is seen not to begin until late in the transient as expected. The noncondensible gases and steam act as forcing functions for the containment pressure as described below.

6.2.3.2 Containment Response

Analyses of containment pressure and temperature transients from degraded core accidents were performed using hand calculations. Hand calculations were used because of a lack of computer codes in the industry with features that allow detailed consideration of containment phenomena occurring for the events under study. The calculations were conducted on a time history basis using the principle of conservation of mass and energy and the ideal gas law. The release rates of steam, hydrogen, carbon monoxide, and carbon

AD CONDENSIBLE GAS

Sequoyah Hydrogen Study
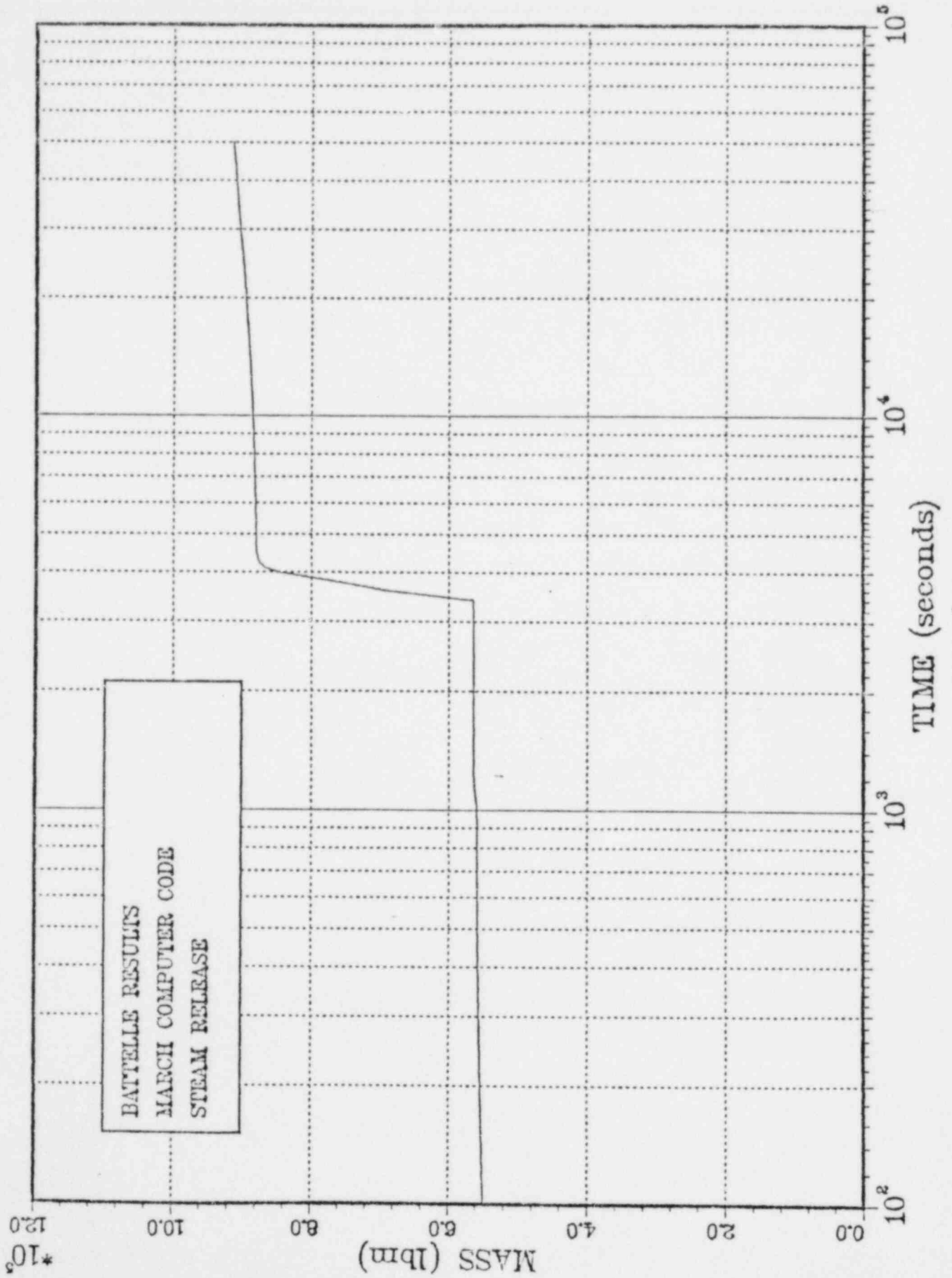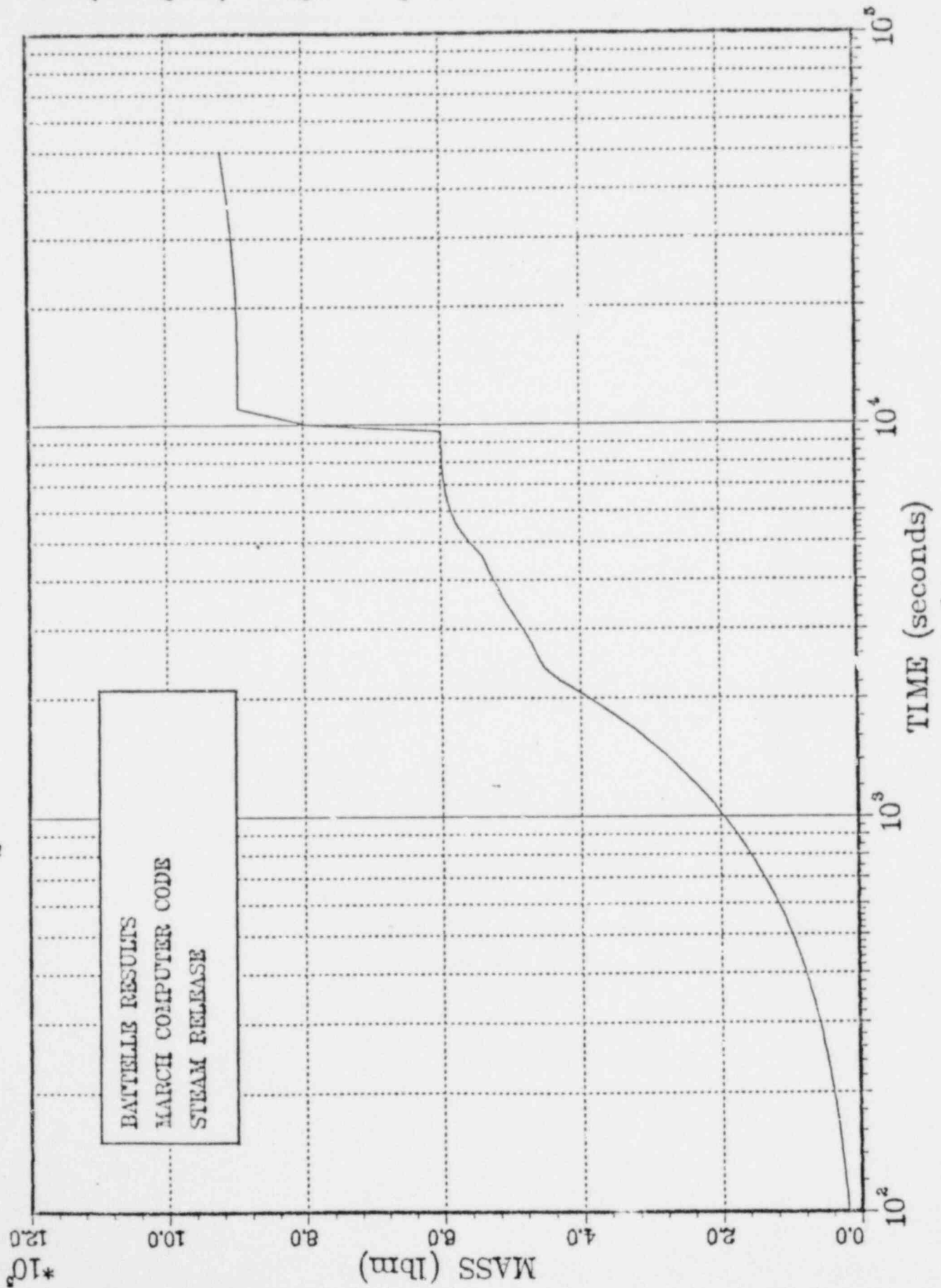
MASS (lbm) *10⁵

TIME (seconds)

FIGURE 6-2

BATTELLE RESULTS
MARCH COMPUTER CODE
STEAM RELEASE

Sequoyah Hydrogen Study

S_2D CONDENSIBLE GAS

BATTELLE RESULTS
MARCH COMPUTER CODE
STEAM RELEASE

MASS (lbm) *10⁶

TIME (seconds)

FIGURE 6-3

6-10

TMLB' CONDENSIBLE GAS
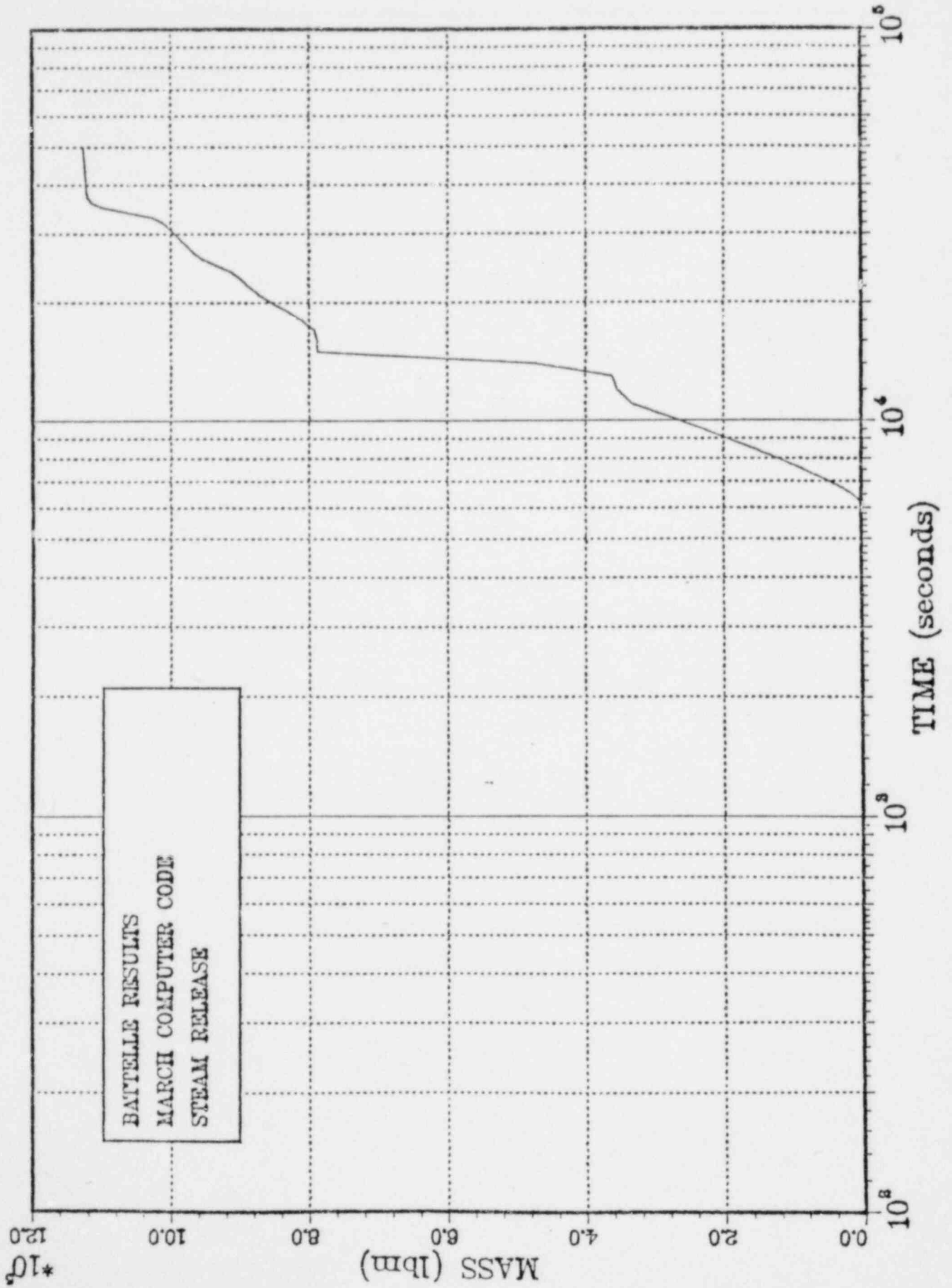
Sequoyah Hydrogen Study

BATTELLE RESULTS
MARCH COMPUTER CODE
STEAM RELEASE

MASS (lbm) *10³

TIME (seconds)

FIGURE 6-4

Sequoyah Hydrogen Study

AD NONCONDENSIBLE GAS

BATTELLE RESULTS
MARCH COMPUTER CODE
HYDROGEN GENERATION

VESSEL MELT

CORE MELT

MASS (lbm)

3800.0 3250.0 2800.0 1950.0 1300.0 650.0 0.0

TIME (seconds)

10^2 10^3 10^4 10^5

FIGURE 6-5

6-12
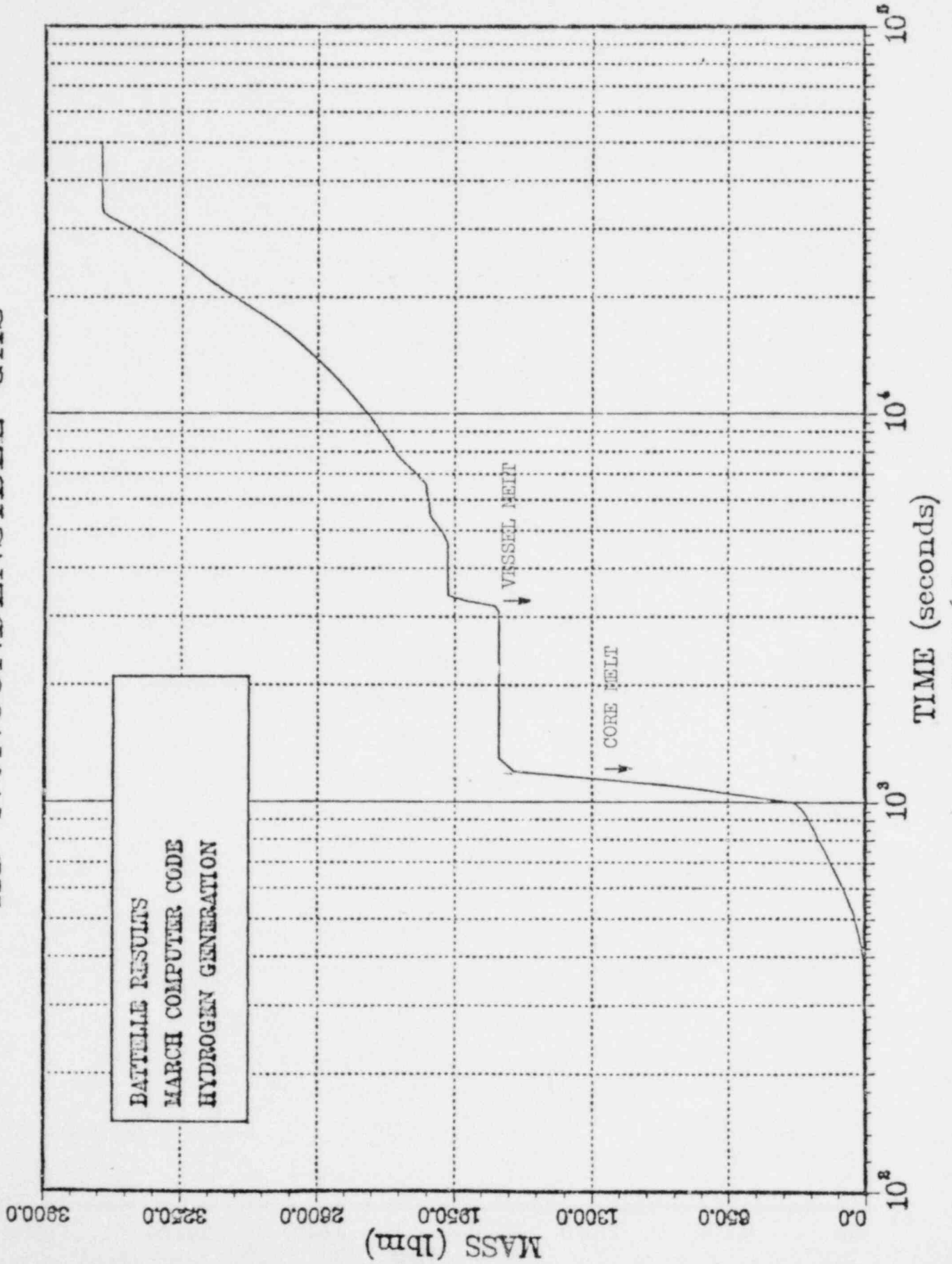
Sequoyah Hydrogen Study

S₂D NONCONDENSIBLE GAS
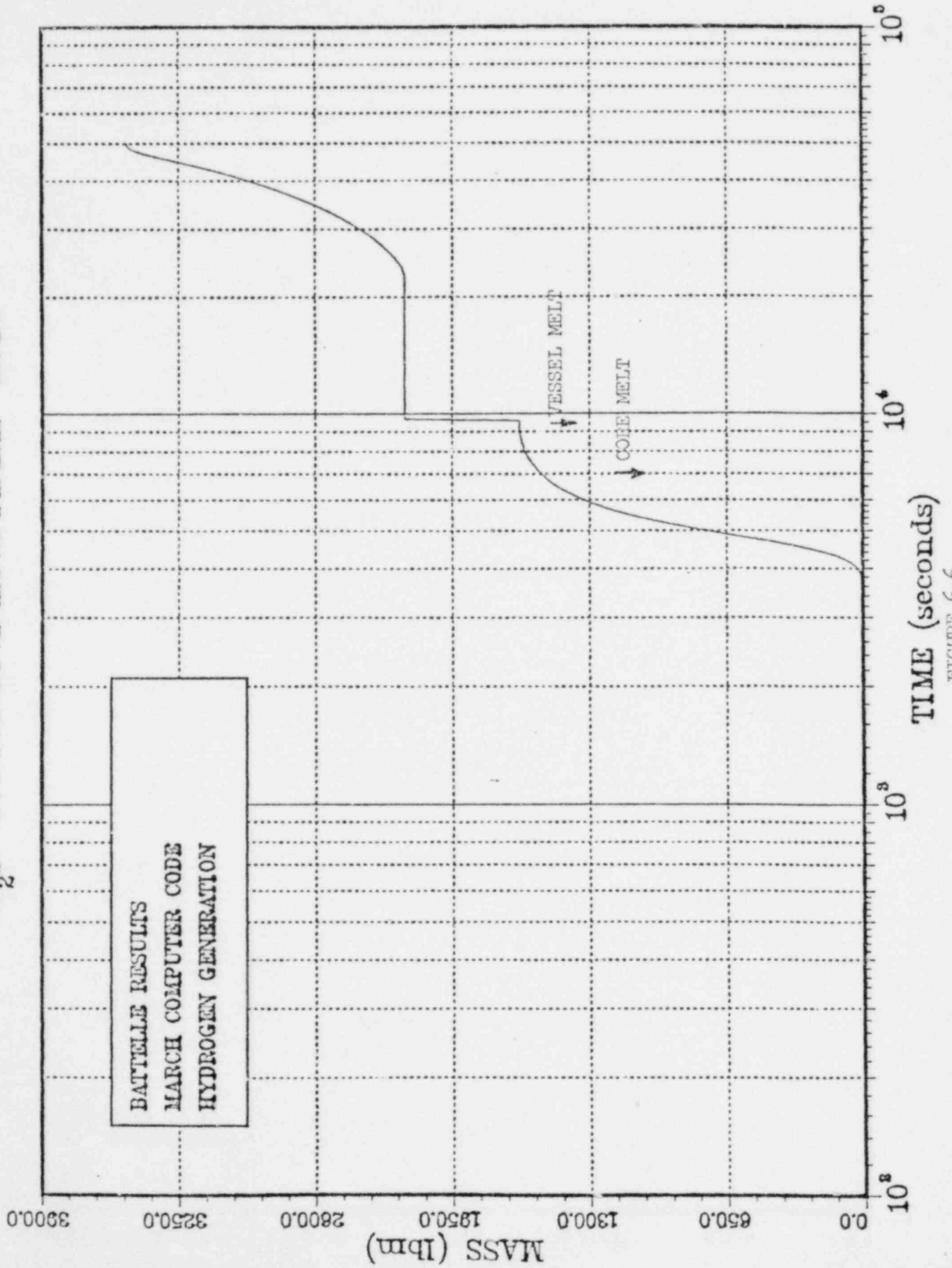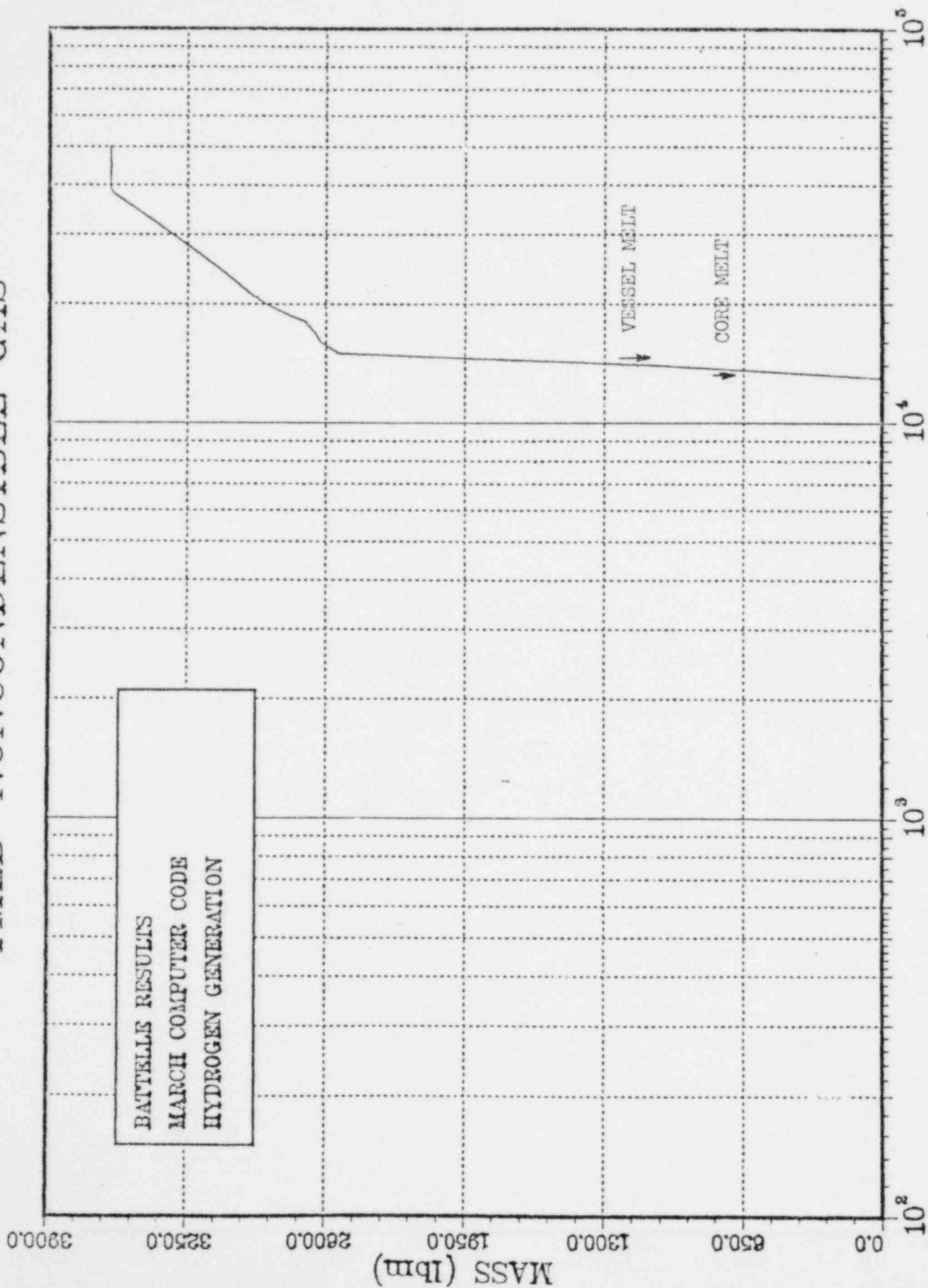
MASS (lbm)

TIME (seconds)

FIGURE 6-6

BATTELLE RESULTS
MARCH COMPUTER CODE
HYDROGEN GENERATION

VESSEL MELT

CORE MELT

Sequoyah Hydrogen Study

TMLB' NONCONDENSIBLE GAS

BATTELLE RESULTS
MARCH COMPUTER CODE
HYDROGEN GENERATION

VESSEL MELT

CORE MELT

TIME (seconds)

FIGURE 6-7

MASS (lbm)

Sequoyah Hydrogen Study

AD NONCONDENSIBLE GAS

CARBON DIOXIDE

CARBON MONOXIDE

BATTELLE RESULTS
MARCH COMPUTER CODE
CARBON MONOXIDE – DIOXIDE

TIME (seconds)

FIGURE 6-8

MASS (lbm) *10⁴

6-15
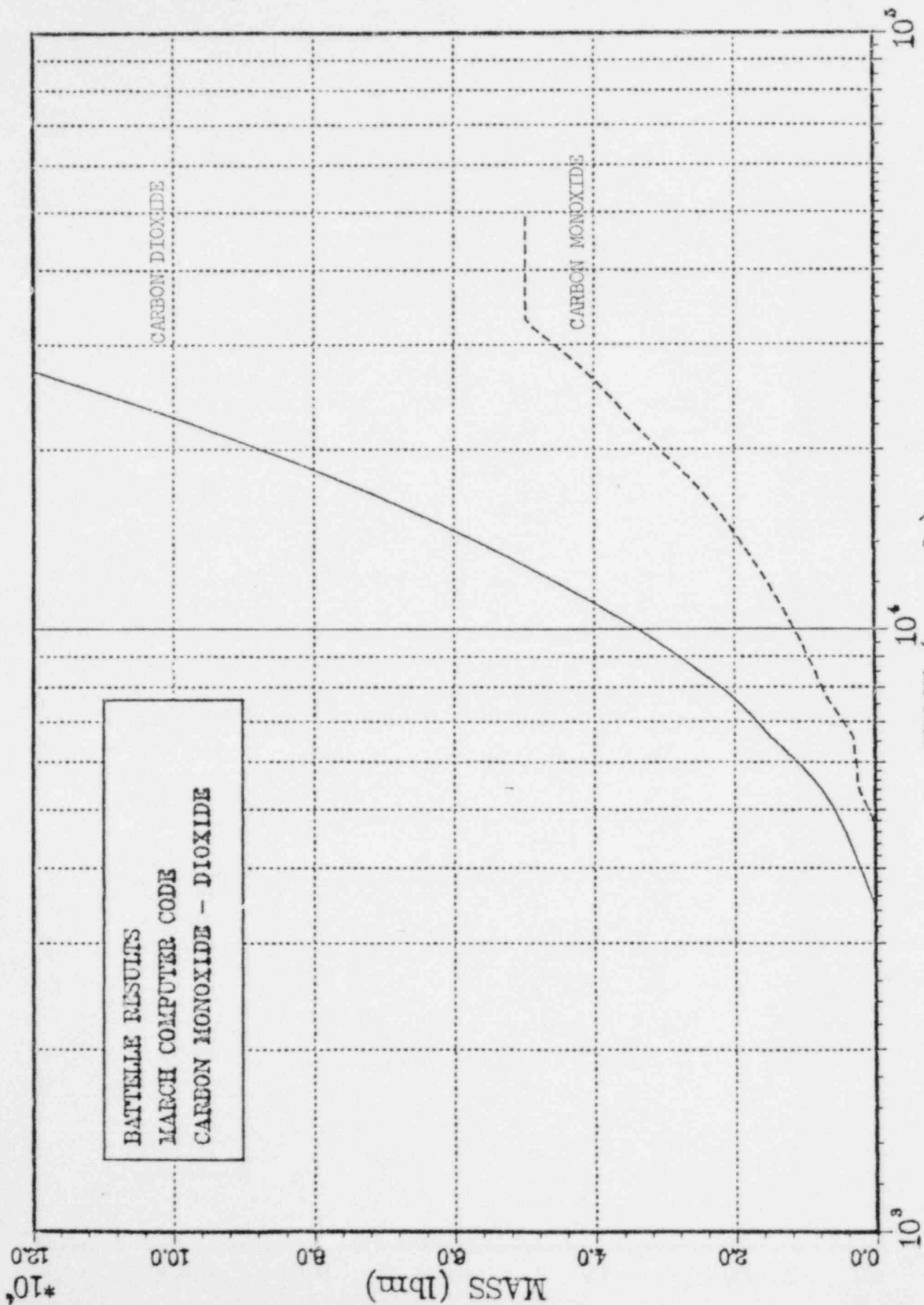
S₂D NONCONDENSIBLE GAS

BATTELLE RESULTS
MARCH COMPUTER CODE
CARBON MONOXIDE – DIOXIDE
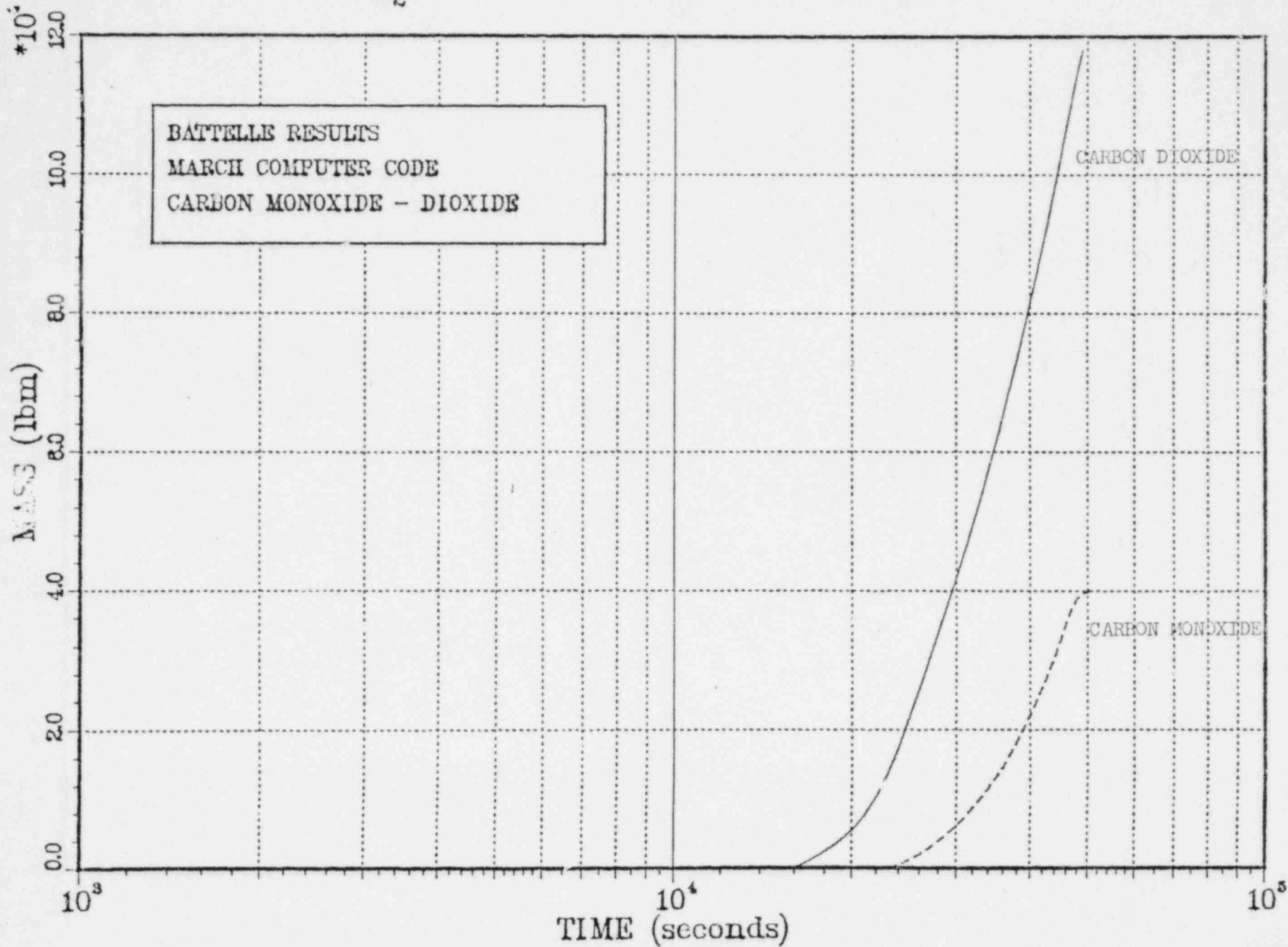
CARBON DIOXIDE

CARBON MONOXIDE

FIGURE 6-9

Sequoyah Hydrogen Study

TMLB' NONCONDENSIBLE GAS
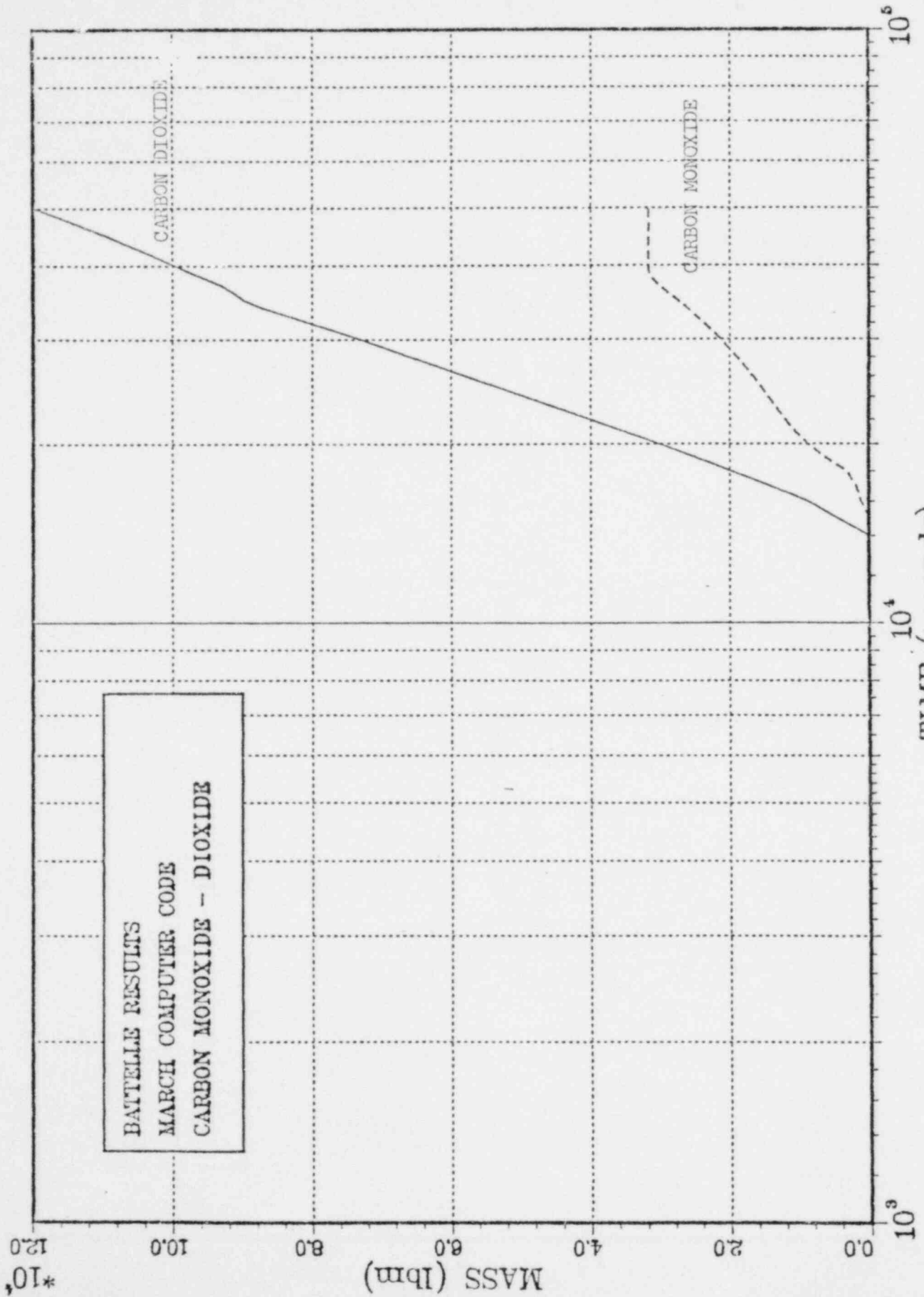
FIGURE 6-10

dioxide for various accidents provided in the MARCH code were reviewed, and the S_2D case was chosen for detailed analysis. The S_2D case is a small break LOCA with loss of high pressure injection. This case was believed to provide a general picture of the conditions that would exist as a result of any accident involving a degraded core. Amounts of hydrogen were allowed to accumulate corresponding to a 20 percent $Zr-H_2O$ reaction and a 60 percent $Zr-H_2O$ reaction. When the hydrogen reaches these limits, it was assumed to burn in five seconds, and the resulting containment temperatures and pressures were calculated. The containment response was also calculated for the S_2D case assuming that the hydrogen was burned as it was produced. This case was analyzed from the initiation of hydrogen production until core melt.

The analyses were performed using the following assumptions:

1. Only the upper compartment was modeled. This assumption was made primarily to simplify the calculations; however, it is believed to be justified. First, for the four major accident sequences studied, large amounts of ice (over one million pounds) remain in the ice condenser at the time of core melt and will be capable of removing significant amounts of energy released by the burning of hydrogen in the lower compartment. The inclusion of the lower compartment and ice condenser in the model would result in lower overall pressures inside containment. However, it is not believed that this represents an undue amount of conservatism.

2. The containment spray system was included (which is consistent with the S_2D sequence). One train of spray was modeled with a flow rate of 4750 gpm. For the rapid burn cases, the entire spray flow was assumed to vaporize due to atmospheric temperatures that were much higher than the saturation temperature of steam associated with the containment pressure at each point in time. For the burn as generated case, the mass of spray water vaporized was determined based on saturation pressure in the containment. The partial pressure of steam produced by vaporizing the spray water was considered.

3. Passive heat sinks were neglected. For periods when rapid burning of large quantities of hydrogen occurs the passive heat sinks have a minimal impact on the peak containment pressure or temperature. The heat sinks will affect the duration of a transient. The impact of the heat sinks on burning small quantities of hydrogen can be ignored due to the effectiveness of the spray system in removing the energy released during these periods.

4. Hydrogen was uniformly distributed throughout the entire containment on a volume fraction basis. A hydrogen collection system is provided at Sequoyah which prevents the accumulation of hydrogen in the steam generator and pressurizer enclosures, the reactor cavity, and the containment dome. This system is

connected to the air return fans which circulate copious
quantities of air throughout the entire containment and provide
for excellent mixing.

5. For cases where hydrogen was allowed to accumulate prior to
    ignition, a flame speed of approximately 10 ft/sec was used during
    the burn period.

6. Initial conditions (prior to the generation of any hydrogen) were
    taken from LOTIC runs performed by Offshore Power Systems. This
    provided initial temperatures, pressures, and distributions of
    steam and air throughout the containment.

6.2.3.2 1 Twenty Percent Zr-H$_2$O Reaction - Rapid Burn

A metal water reaction oxidizing 20 percent of the zirconium cladding produces 420 pounds of hydrogen. In the S$_2$D sequence, this hydrogen is released to the containment as produced. All 420 pounds will be in containment at 4500 seconds after the start of the accident. A rapid burn of accumulated hydrogen results in a containment pressure of 54.3 psia and an atmospheric temperature of 960° F. The containment pressure and temperature profiles for this case are provided in figures 6-11 and 6-12.

6.2.3.2.2 Sixty Percent Zr-H$_2$O Reaction - Rapid Burn

A rapid burn of 1325 pounds of hydrogen produces a peak containment pressure of 126 psia with a concurrent temperature of 2713°F. This quantity of hydrogen, corresponding to approximately 62 percent metal-water reaction, is released at 6970 seconds after the start of the accident in the S$_2$D sequence. Figures 6-13 and 6-14 provide the pressure and temperature profiles for this case.

6.2.3.2.3 Burn Hydrogen as Generated

In this case the containment pressure stayed below the design pressure of 26.4 psia from the start of the accident until 9651 seconds. During this period of time hydrogen is produced as the core is uncovered. At about 7000 seconds, the production of hydrogen drops sharply as the water in the reactor vessel has all boiled off. While significant quantities of hydrogen are generated, they are released steadily over time and the spray system is capable of removing the heat of combustion without exceeding the containment design pressure. For the period of time between 7000 and 9651 seconds, the core continues to heat up due to the decay of fission products and at the end of this period the core melts through the bottom of the reactor vessel and falls into the reactor cavity. The water in the reactor cavity oxidizes the remaining cladding. This occurs with extreme rapidity and the hydrogen associated with the remaining 30 percent of the zirconium cladding is generated in less than 10 seconds. The burning of this release produces a containment pressure of 80 psia (see figures 6-15 and 6-16).

A detailed computer analysis of each transient would be preferable to the hand calculations used; however, these results represent a reasonable and conservative approach to the problem based on the information and methods presently available.

6.3 Plant Capabilities to Withstand Extended Design Basis Events

The Sequoyah containment systems have built-in margins which would allow the plant to withstand events more challenging than those

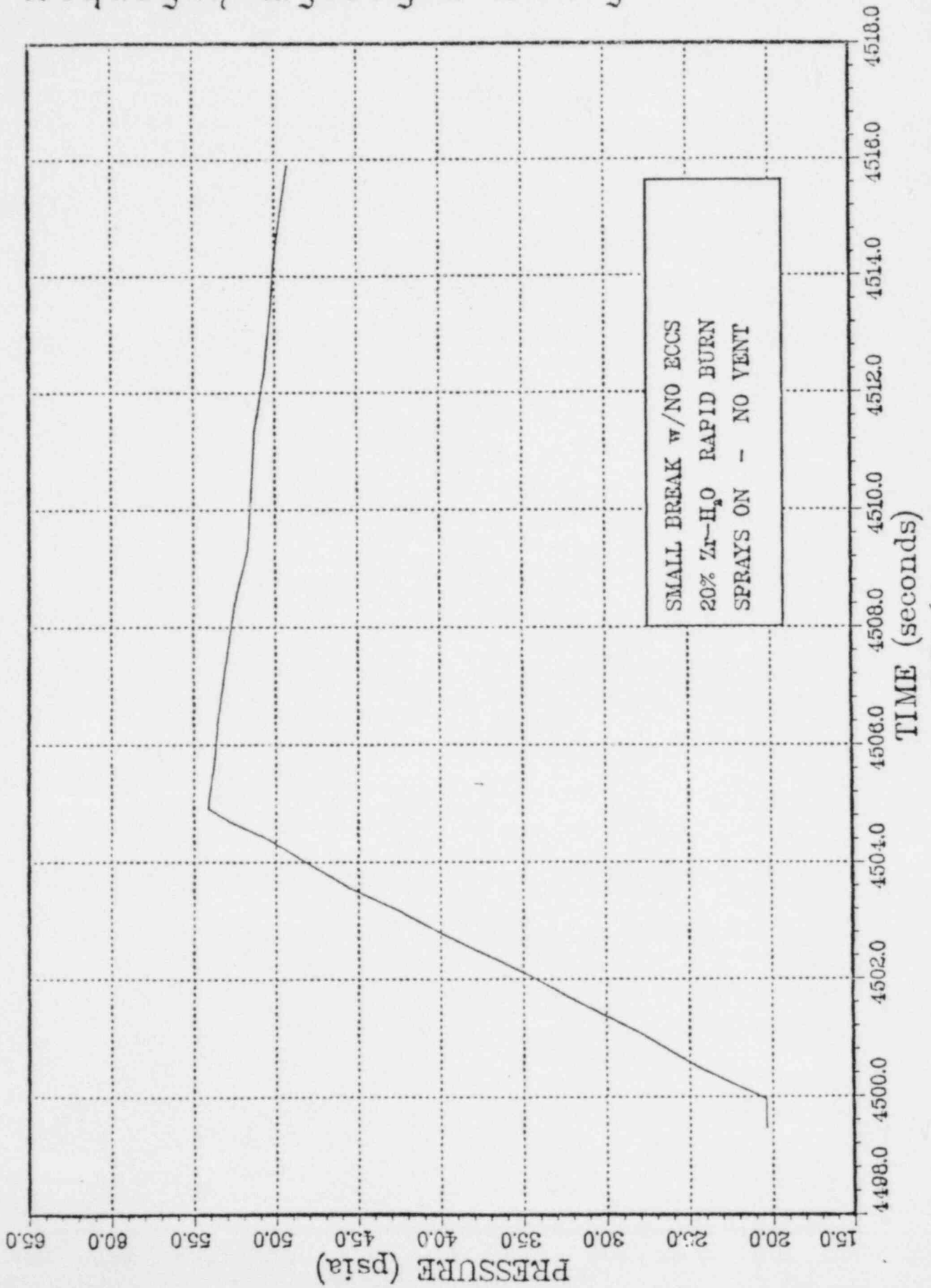# Sequoyah Hydrogen Study

CONTAINMENT PRESSURE RESPONSE

SMALL BREAK w/NO ECCS
20% $Zr-H_2O$ RAPID BURN
SPRAYS ON – NO VENT

TIME (seconds)

PRESSURE (psia)

FIGURE 6-11

6-21

CONTAINMENT TEMPERATURE RESPONSE

Sequoyah Hydrogen Study

SMALL BREAK w/NO ECCS
20% Zr-H₂O RAPID BURN
SPRAYS ON – NO VENT

TIME (seconds)

TEMPERATURE (°F)

FIGURE 6-12

CONTAINMENT PRESSURE RESPONSE

Sequoyah Hydrogen Study

SMALL BREAK w/NO ECCS
82% Zr—H_2O RAPID BURN
SPRAYS ON - NO VENT

TIME (seconds)

PRESSURE (psia)

FIGURE 6-13

CONTAINMENT TEMPERATURE RESPONSE

Sequoyah Hydrogen Study

SMALL BREAK w/NO ECCS
82% Zr-H₂O RAPID BURN
SPRAYS ON – NO VENT

TEMPERATURE (°F)

TIME (seconds)

FIGURE 6-14

6-24

CONTAINMENT PRESSURE RESPONSE

SMALL BREAK
100% Zr-H₂O REACTION
BURN AS PRODUCED

PRESSURE (psia)

TIME (seconds)

FIGURE 6-15

CONTAINMENT TEMPERATURE RESPONSE

Sequoyah Hydrogen Study

SMALL BREAK
100% Zr–H₂O REACTION
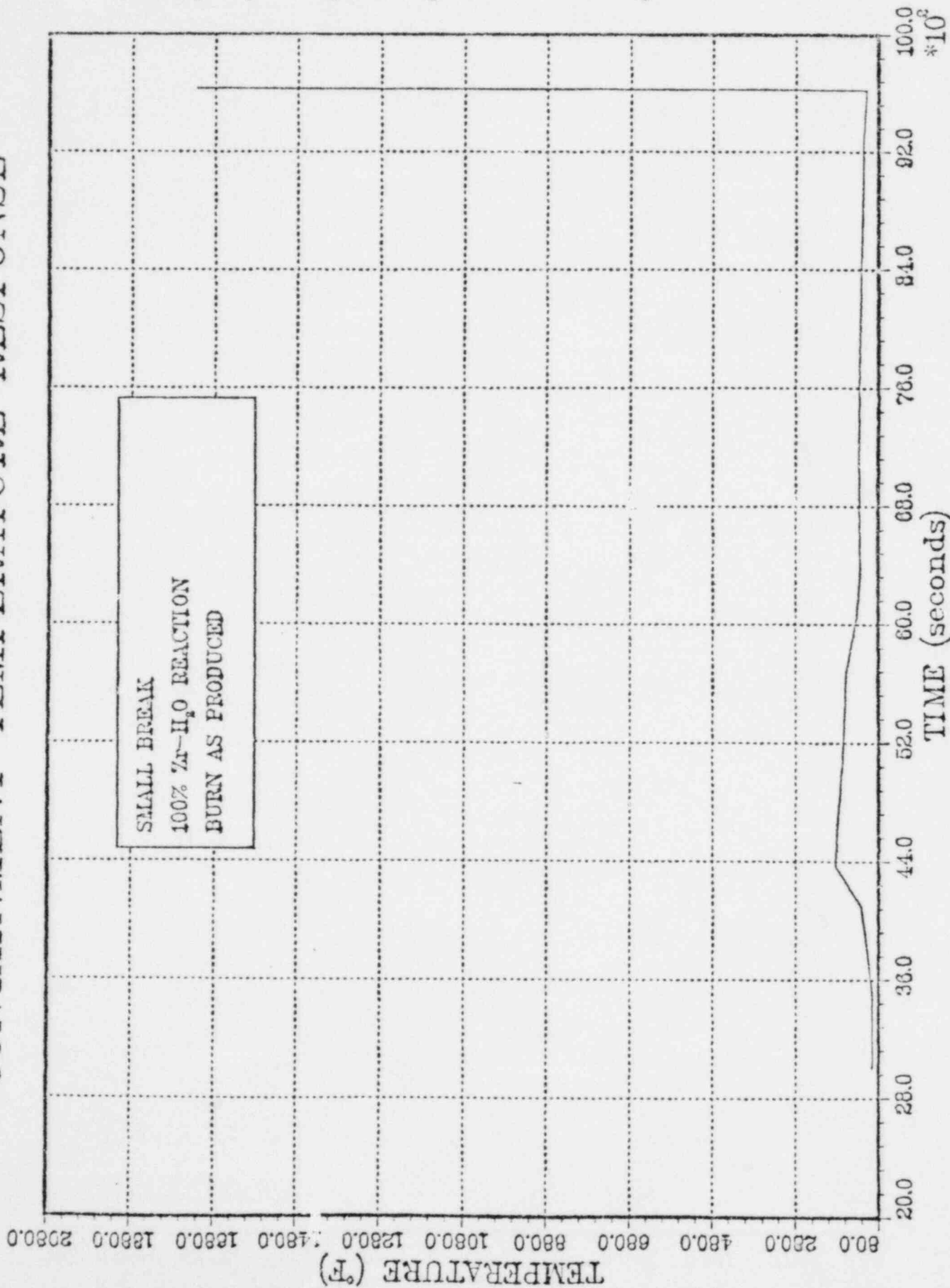BURN AS PRODUCED

TEMPERATURE (°F)

TIME (seconds)
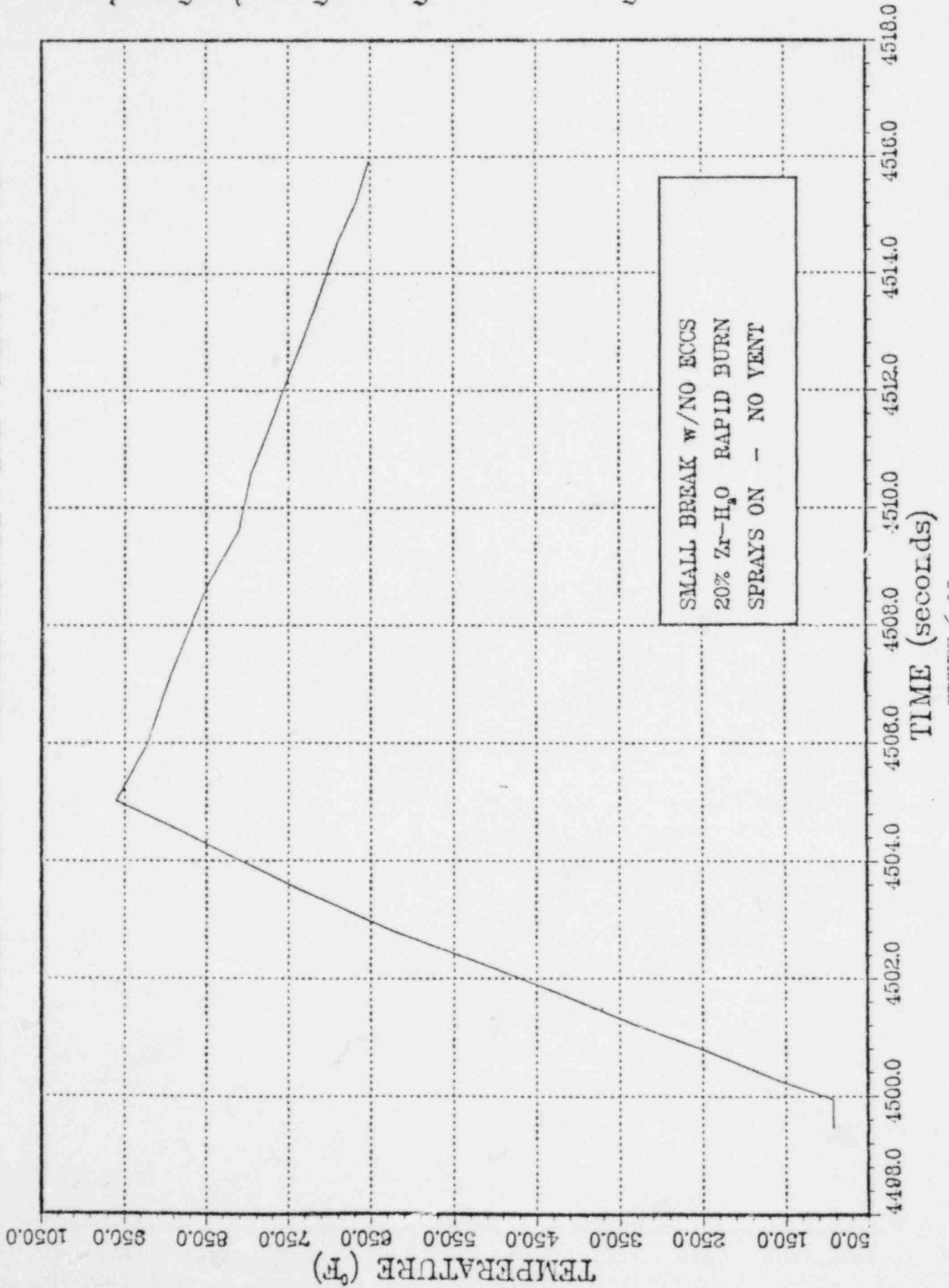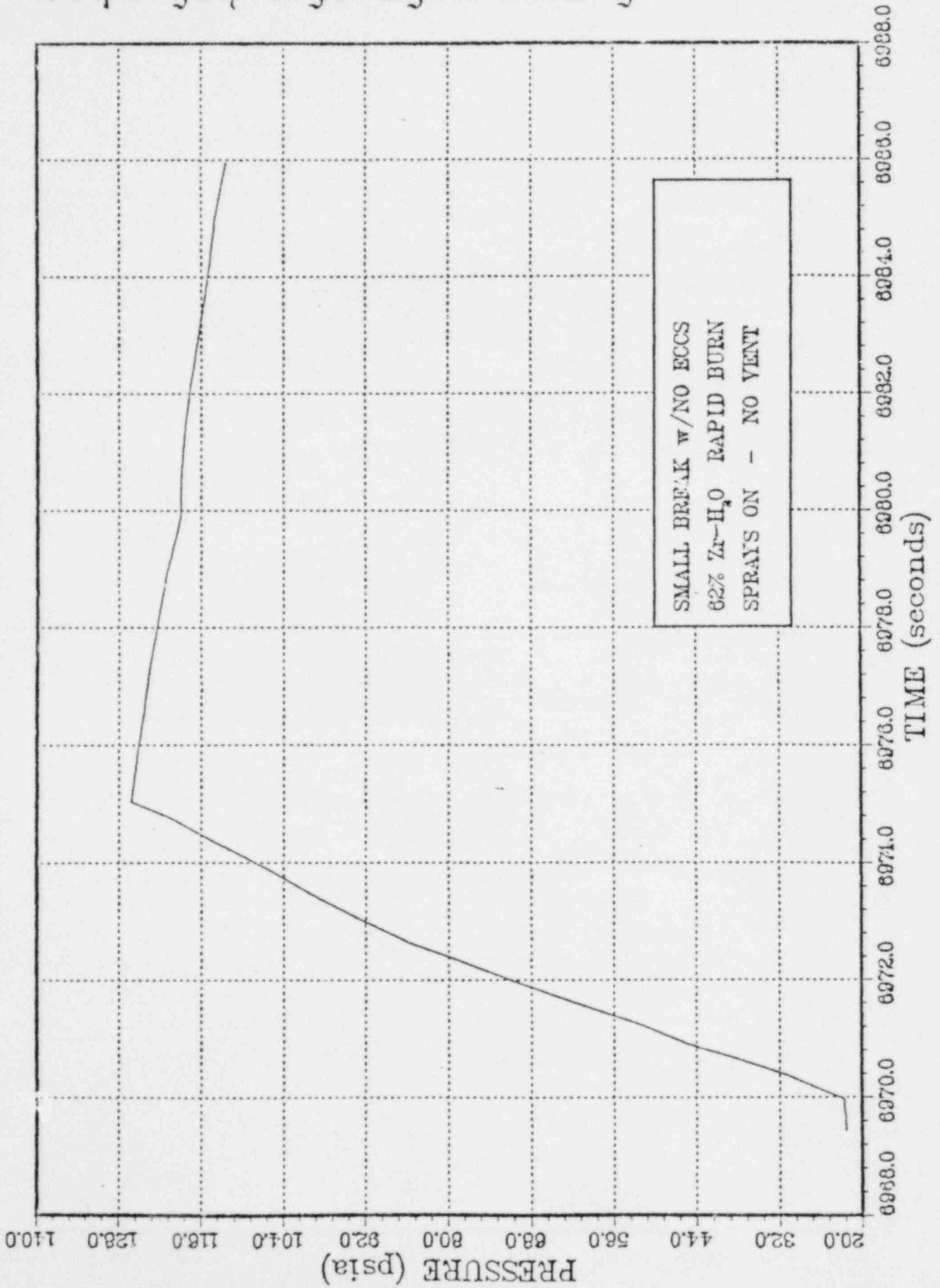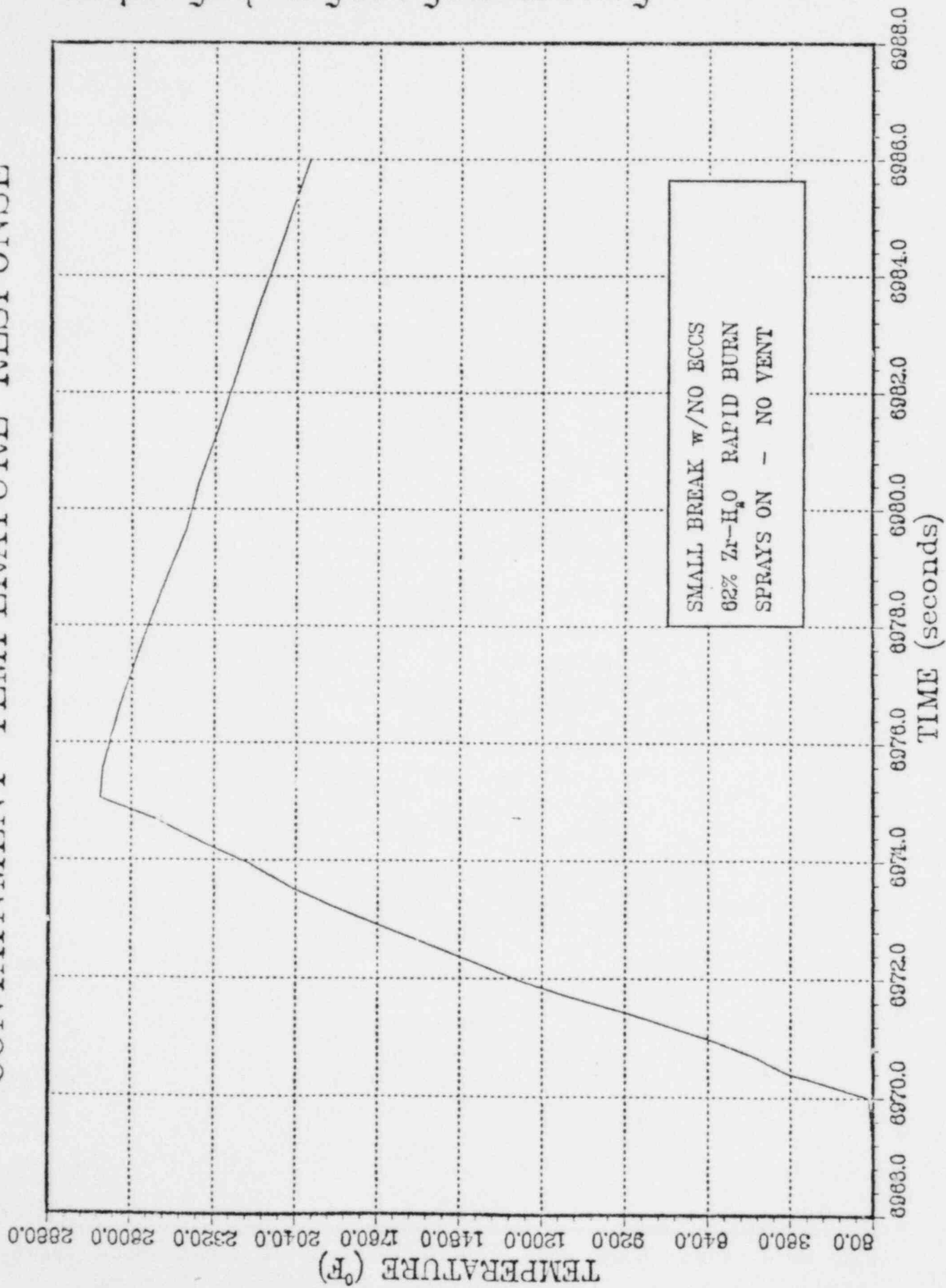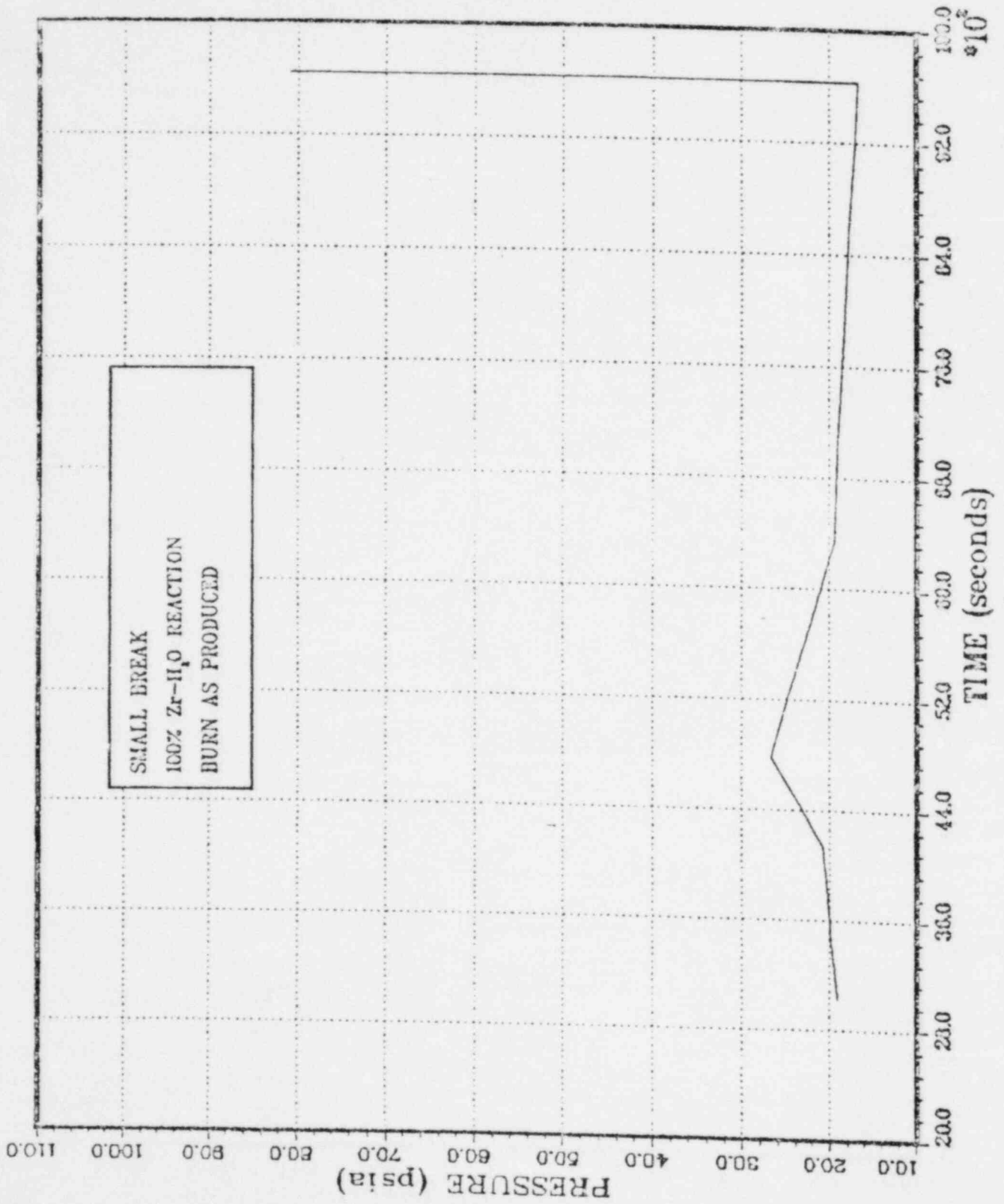
FIGURE 6-16

6-26

considered in the original design. Some of the more useful for
dealing with hydrogen are discussed below:

6.3.1 Containment Structure

The containment design pressure is 26.4 psia. However, there
are very large margins in this value. The containment steel is not
expected to yield until a pressure greater than 37.4 psia
is reached, and the containment ultimate strength corresponds to a
pressure of 57.9 psia. The containment can withstand pressure
more than three times design without a significant probability of
failure.

6.3.2 Ice Condenser

The actual ice loading at Sequoyah was 2.97×10^6 pounds compared
with a value of 2.45×10^6 pounds used in the design analyses. This
represents an added heat sink of 130×10^6 Btu's available to
accommodate heat from hydrogen burn effects.

6.3.3 Containment Spray Systems

The Sequoyah containment is equipped with two spray systems composed
of four independent subsystems. Each subsystem has its own piping,
pumps, and headers. The spray system is capable of injection 13,500
gpm. The analysis of the current design basis events used a spray
flow rate of 6750 gpm. The evaluations of the events studied
used a spray rate of 4750 gpm.

DA03:NSAHY1.PB

7.0 Mitigation Concepts

Based on our knowledge of hydrogen and other core damage-related
phenomena and concepts proposed by TVA and others, the following
concepts were identified as those having the most promise and being
worthy of study for mitigating the effects of hydrogen and containment
overpressurization:

1. Inerted containment
2. Halon injection
3. Ignition sources
4. Augmented cooling
5. Filtered vented containment
6. Additional containment
7. Coupled containment
8. Oxygen removal by chemical reactant
9. Passive enclosure
10. Structural reinforcement
11. Combinations of the above

Concepts 1, 2, 3, and 8 provide some protection from the effects of
hydrogen combustion. Concepts 4, 5, 6, 7, 9, and 10 provide
some protection of the containment due to the effects of overpressure.
These concepts were given a screening evaluation in terms of
adaptability to existing plants and plants under construction, safety,
and feasibility. This preliminary screening resulted in the
elimination of oxygen removal by chemical means, passive enclosure
and structural reinforcement concepts for being infeasible,
impractical, or ineffective for Sequoyah. The preliminary study also
concluded that for a number of accident sequences, many of the
concepts under consideration may not be sufficient to cope with all of
the effects of core degradation and would have to be considered in
combination with other mitigation techniques. Such combinations would
consist of one of the concepts which provide protection from
hydrogen combustion and one of the concepts which provide protection
from containment overpressure.

7.1 Inerted Containment

To prevent hydrogen combustion, prior to reactor startup and
operation, containment atmosphere would be diluted with nitrogen until
the oxygen concentration is reduced to less than 4 percent by volume.
The nitrogen could be introduced into containment through existing
ventilation and purging ducts. Nitrogen would either be supplied
by an outside vendor or a moderate sized manufacturing facility
onsite. All areas in containment must be provided with life support
equipment and gas monitors. Modifications such as replacing air with
nitrogen in the compressed air system will be required to eliminate
oxygen sources in the containment. The system should be designed
to rapidly reduce the containment oxygen concentration to minimize
plant downtime.

7.2 Halon Injection

Halon is a fire suppressant that in sufficient quantities may prevent burning and detonation of hydrogen in the containment. This concept consists of spray headers at various locations in containment fed by pressurized tanks located outside containment. Halon would be injected into the containment over a short time interval upon indication of a failure which may lead to significant hydrogen generation or upon detection of a significant concentration of hydrogen in containment. The effect would be to create an inerted atmosphere postaccident. In contrast to oxygen depletion, such as by dilution with nitrogen, the halon apparently absorbs the burn energy rapidly by chemical processes, thus preventing the flame front from propagating.

7.3 Ignition Sources

This concept i   aimed at controlling hydrogen buildup within containment.   hydrogen is released to the containment slowly and burned at rela vely low concentrations, it can be controlled without severl verpressurizing the containment (which could occur if it is allowed to build up in large amounts and is ignited at random). The concept consists of ignition sources located throughout containment in areas where hydrogen would potentially accumulate and additional hydrogen monitors to provide indication of the buildup. Upon indication, the igniters would be activated. Continued monitoring would provide indication of the need to deactivate the igniters (i.e., if hydrogen buildup is too rapid). Since the ignition system will be designed to burn hydrogen in low concentrations, this will require an open flame or its equivalent.

7.4 Augmented Cooling

Additional containment cooling would help remove heat from the containment, thus reducing the containment pressure. There are several potential ways to increase the cooling capacity of the containment such as additional containment sprays, additional air coolers, and other heat sinks. Because of containment space limitations, passive heat sinks could be considered only in conjunction with other concepts such as an additional containment volume. While additional active cooling may not be sufficient for mitigation of large energy releases of short duration, it may be satisfactory for energy removal under slow controlled hydrogen burning.

7.5 Filtered Vented Containment

This concept provides containment pressure relief through a vent by flow of noncondensibles, steam, and energy. The flow is vented through a mechanism such as a suppression pool or gravel bed to remove heat and steam. The flow proceeds through a sand filter for additional removal of moisture and radioactive materials and through additional filters such as particulate and/or charcoal for high efficiency removal of radionuclides. Remaining noncondensible gases and radioactive noble gases are discharged to the atmosphere via a stack to provide elevated release and dispersion. The design must

provide for capture and postaccident cleanup of radioactive liquids and filters and for removal of fission product decay heat. Hydrogen must be prevented from uncontrolled burning within the vent system. This concept allows controlled, reduced releases to the environment to prevent an uncontrolled release that could occur following containment failure.

7.6 Additional Containment

This concept provides containment overpressure protection with additional volume and heat sinks to hold some of the noncondensible gases, steam, and energy generated from an event involving core damage. The additional containment volume would be coupled to the existing containments via a large vent which could be opened during an accident if conditions warranted. One additional containment would serve all units at a plant site. A preliminary estimate of the size required is between 1.5×10^6 ft^3 and 3.5×10^6 free volume.

7.7 Coupled Containment

The coupled containment concept is identical to the proceding concept except that it makes use of the second nuclear unit's containment to provide the additional volume for overpressure protection (all TVA nuclear plants have two or more units per site). For pressure suppression containments, the second unit may provide a large amount of additional cooling.

7.8 Oxygen Removal by Reaction

While in theory this concept has merit, our preliminary investigation found no chemical which would permanently remove the oxygen or remove it quickly and reliably. It appears that extensive research is needed to advance this concept to a stage permitting a conceptual design. Thus, this concept was not considered further in this study.

7.9 Passive Enclosure

The passive enclosure concept has previously been considered as an alternate containment concept in which the entire reactor coolant system is encased in a guard pipe/chamber arrangement. For use as a mitigation technique of degraded core events, the device would be designed to withstand extremely high pressures. It would provide a mini-containment around the entire primary system and be designed to retard melt through. Cooling would be incorporated. This concept; however, is impractical for current TVA plants and was eliminated from further consideration.

7.10 Passive Structural

This concept may be used in limited cases. The purpose is to enhance the containment load-bearing capability by structural reinforcement.

A review of TVA plants indicated that significant strengthening cannot be achieved within practical means. No further consideration of reinforcement was made in this study.

7.11 Other Concepts

Other concepts may exist in addition to those listed above. It is believed that others worthy of consideration would employ one or more of the principles involved in the concepts identified abo e; therefore, those described are representative of the proposed mitigative concepts.

8.0 Risk Assessment of Mitigation Concepts

A preliminary risk assessment of selected events was performed to provide an approximate quantification of the reduction in risk associated with the addition of various mitigation concepts for accidents resulting in reactor core degradation. This assessment does not quantify the change in total risk to the public, only the change in the probability of containment failure for the selected events. Also, the assessment did not take into account that the addition of mitigation concepts could potentially increase the risk to the public for design basis events (e.g., if the operator inadvertently opened valves in the line to the filter bed used in the filtered vented containment concept, the dose to the public would be increased).

Based on analyses of selected events beyond the design basis LOCA that threaten containment integrity, event trees were prepared using methods similar to those used in WASH-1400. The event trees were prepared for the present plant design and then redone to account for the benefits of each concept or combination of concepts. These were used to estimate how frequently and by what mode the containment would fail (leakage, hydrogen burn, steam explosion, noncondensible gas production). Table 8-1 summarizes the reduction in containment failure due to various mitigative strategies for a particular small LOCA event (S_2D). The numbers in this table are not representative of large LOCA's and transients and may not be representative of other small LOCA's. For these other events, the numbers may change in absolute value and/or relative to one another. This is because the consequences of different accidents vary in both rate and magnitude (as described in section 6.2.3.1 for the sequences used in this study) such that the different mitigative strategies have varying degrees of benefit. In addition, there are other important accident sequences that may lead to containment failure (e.g., failure to close all isolation valves or interfacing check valve failure) that do not involve either hydrogen burning or overpressure. None of the mitigations studied here could reduce the risk from those sequences. Under no circumstances should the containment failure probability reductions presented in Table 8-1 be generalized to imply any similar overall risk reduction for Sequoyah.

Concepts which employ containment inerting involve the additional risk of having an accident because of reduced access to containment for inspection, maintenance, and repair. While this risk is difficult to quantify, a small increase in the probability of failure of a safety system may outweigh the potential benefits from including hydrogen mitigation in the plant design. Effects of this type are not reflected in the results in Table 8-1.

TABLE 8-1

REDUCTION IN PROBABILITY OF CONTAINMENT FAILURE BY
VARIOUS MITIGATION STRATEGIES FOR A SMALL LOCA
EVENT RESULTING IN CORE DAMAGE (S_2D)

	Reduction Effectiveness for the Probability of Containment Failure[2]			
	Partial Core Damage[3]		Core Melt	
Mitigation Strategy	Hydrogen Burn Generated Failure	All Failures[4]	Hydrogen Burn Generated Failure	All Failures
Ignition Sources	70	70	7	7
Halon Injection	70	70	75	11
Nitrogen Inerting	100	100	100	20
Filtered Vented Containment	15	15	9	8
Additional Large Containment	34	34	16	15
Coupled Containment	30	30	5	5
Filtered Vented Containment and Inerting	100	100	100	87
Large Containment and Inerting	100	100	100	96

1. These reductions are for the S_2D case only and cannot be applied to any other event.

2. The numbers represent the percentage of times, given that the event occurs, that the particular mitigation strategy would prevent containment failure.

3. Partial core damage represents a level of damage where it is possible to terminate the event if cooling water can be restored to the core. For the S_2D case, this damage level corresponds to 60-65 percent clad oxidation; for other events, this level may be much lower. Once the partial core damage upper bound is reached, the event proceeds to core melt with extreme rapidity.

4. All failures occurring as a result of the S_2D event (e.g., overpressure due to noncondensible gas production).

9.0 Mitigative Conceptual Designs

Each of the seven mitigations described in section 7.0 was implemented
in a conceptual design by various architect-engineering and TVA design
organizations based on the preliminary design criteria summarized
in Appendix D. These criteria include performance criteria,
environmental conditions, operating modes, seismic criteria, passive
vs. active components, and safety-related versus nonsafety-related
qualification. It should be emphasized that none of the designs were
implemented as safety-grade for this study except where required to
prevent degradation of current safety systems (e.g., contianment
isolation valves). In the absence of any clear regulatory directives
and due to the brief duration of this study, this decision was made
purposefully to simplify the mitigation designs and not escalate their
cost. More rigorous critiera would have a major impact on cost.
The designs based on these criteria included enough detail to evaluate
feasibility, safety, scheduling, and cost. The results of these
evaluations are summarized in this section.

9.1 Filtered Vented Containment

9.1.1 General Description

The filtered vented containment (FVC) system will allow high
temperature and pressure gases to be vented from either containment
after an accident through iodine and heat removal subsystems and
a particulate removal subsystem before being released to the
atmosphere.

The system, as shown in figures 9-1 and 9-2, will penetrate each
containment structure with two 36-inch lines. Each line has three
principal isolation valves: one inside the primary containment, one
in the annulus, and one outside the concrete secondary containment.

Each line also has a rupture disk which opens at 35 psia. The two
lines from each contianment tie into a 54-inch header in an
underground valve pit. Each line will have an isolation valve to
isolate the containments from each other and to prevent contamination
of the lines not in use.

A common 54-inch line will be routed to a second underground valve
pit where it will split into two 54-inch discharge headers, each with
its own isolation valve. One of these 54-inch headers, the pool vent
gas discharge header, is employed when the affected containment
presure is at or above 5 psig, and releases the gas under a pool of
chemically treated water through a distribution network.

The other 54-inch header, or the pool bypass vent gas header,
discharges the gas into the air space over the pool and is used when
the affected containment pressure is approximately 5 psig or below.

The pool of chemically treated water, located in the lower portion
of the filtration building, will serve as the primary steam
condensing, heat removal, and iodine removal mechanism during use
of the system. It is also the source of water for the iodine removal
spray subsystem, which is located at the top of the air space above
the pool. When the vented filtered cotnainment system is required
to operate, a vertical turbine pump will discharge the chemically
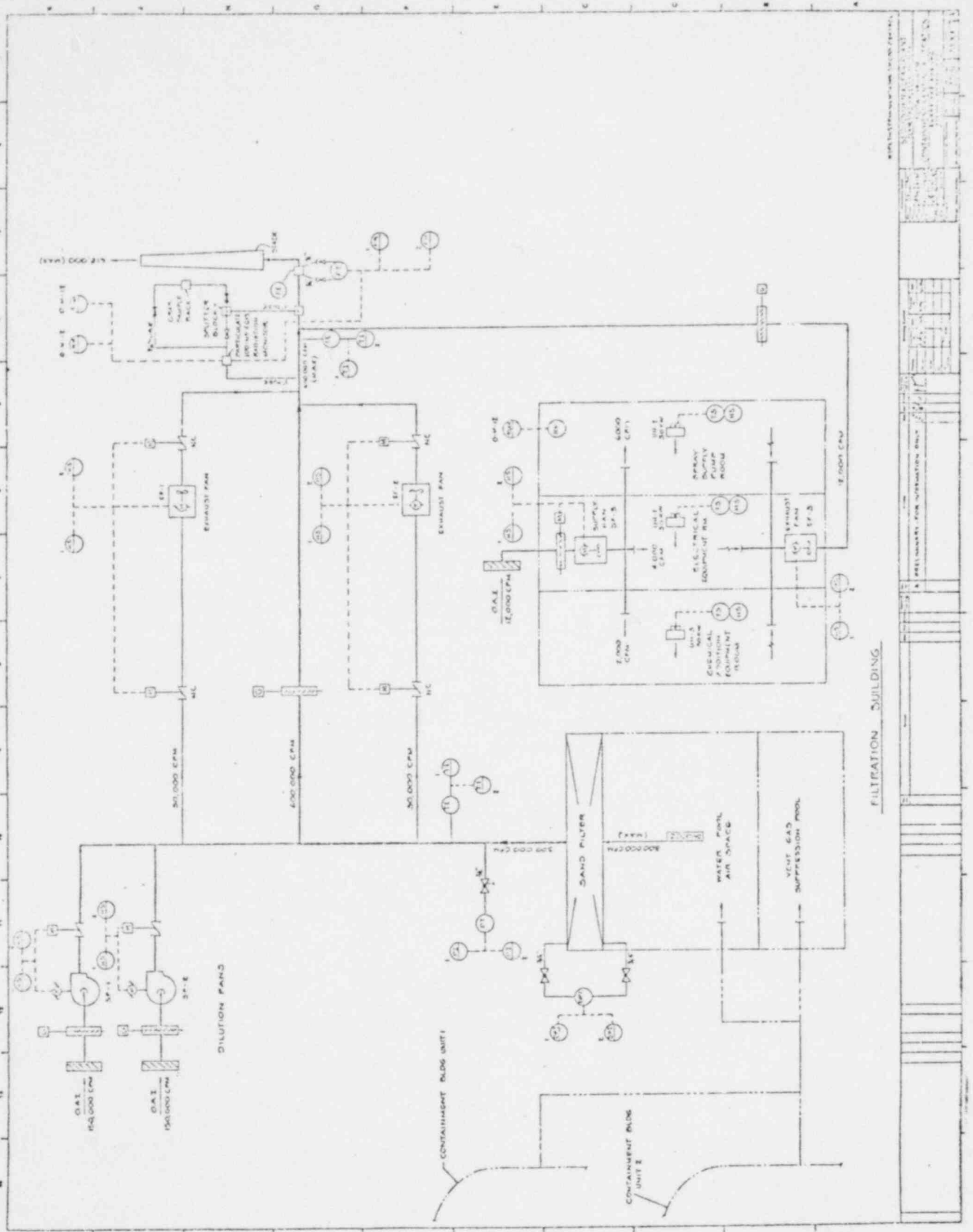
FIGURE 9-1

FIGURE 9-2

treated water to the spray headers.

The vented containment gas mixture will rise through the iodine removal sprays above the pool and flow upwards through a sand filter. The sand filter will act as a high efficiency particulate and heat absorbing filter which has the ability to withstand the high heat and pressure expected after an accident.

The sand filter is composed of layers of progressively smaller gravel and sand covered by two layers of larger gravel and sand. The overburden of larger sand and gravel is needed to maintain the sand filter's integrity and to prevent dusting.

The vented containment gas will exit the sand filter into a plenum above, where it will be mixed with outside dilution air. The quantity of dilution air is sufficient to keep the hydrogen concentration of the vented gas below the explosive threshold.

When the accident occurs and core damage becomes likely, the appropriate containment isolation valves would be opened. This action would not initiate flow to the FVC system until containment pressure reached the setpoint of the inline rupture disks (35 psia). The appropriate flow path isolation valve would also be opened. The other valves in the system should all have been prealigned so that no other valve operation would be required. Once the containment isolation valves have been opened, creating the possibility of FVC system operation, the iodine removal spray supply pump should be started.

When the rupture disk finally blows, the containment vent gas will travel down the 54-inch gas supply piping, into the sparger piping, through the pool and iodine removal sprays and into the particulate sand filter. This path of gas venting could continue until an isolation valve is closed or until the pressure in containment falls below a pressure of about 5 psia (the nominal height of the pool water plus system losses). When the vent gas supply line flow indicator indicates little or no flow, or when the containment pressure indicators indicate a pressure of about 5 psig, the pool discharge vent gas header isolation valve should be closed and the pool bypass vent gas header isolation valve should be opened. This action would transfer the system to the low capacity/pressure mode of operation.

The combined containment vent gases and dilution air exit the building through the backdraft dampers and flows through an underground discharge tunnel where it is monitored for radiation levels, temperature and flow. The vented gases are released to the atmosphere via a 400-foot freestanding vent stack.

Two large exhaust fans are also provided to maintain either primary containment at a slightly negative pressure if the containment isolation function is degraded. Exhausted containment atmosphere would bypass the suppression pool via the pool bypass vent gas header. It would pass through the iodine removal sprays above the pool, through the sand filter, and be exhausted through the vent stack. Auxiliary subsystems are provided to maintain proper chemistry in the pool, supply building heating and ventilation, and supply

IMAGE EVALUATION
TEST TARGET (MT-3)

6"
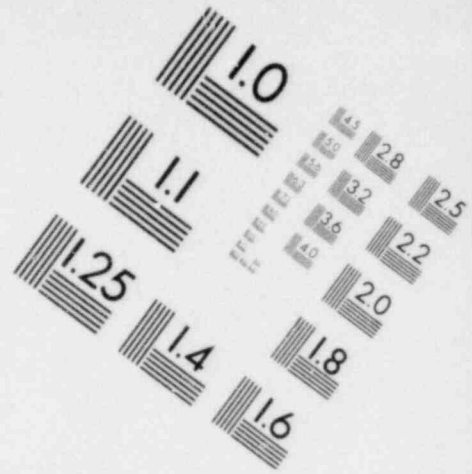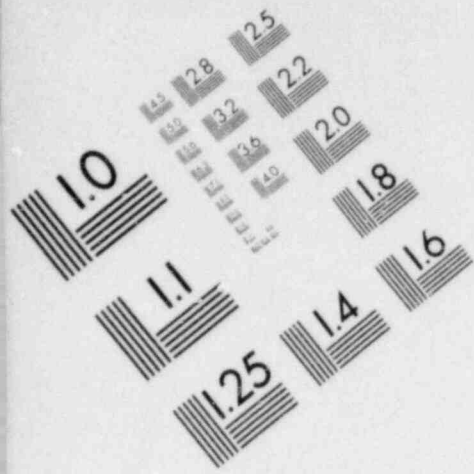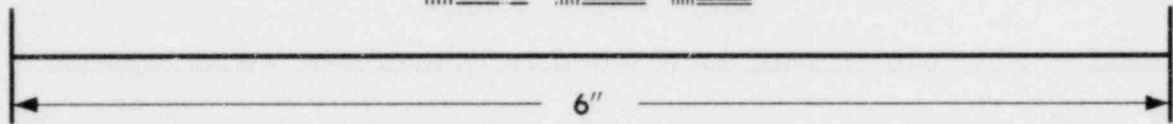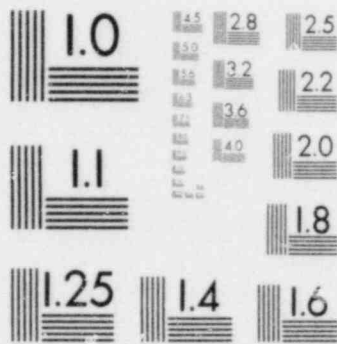
MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6"
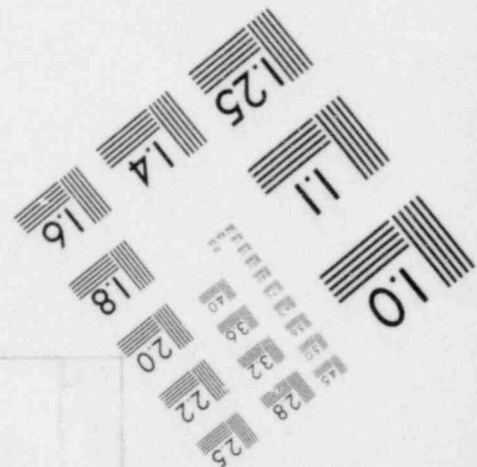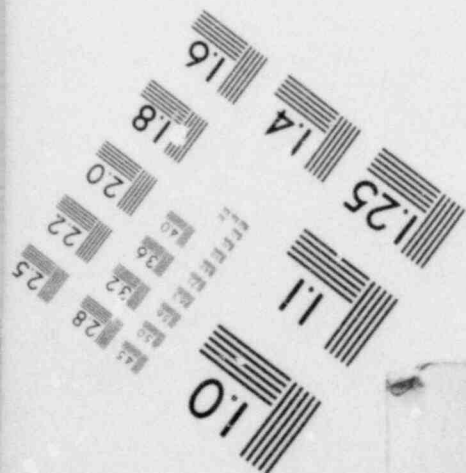
MICROCOPY RESOLUTION TEST CHART

decontamination availability.
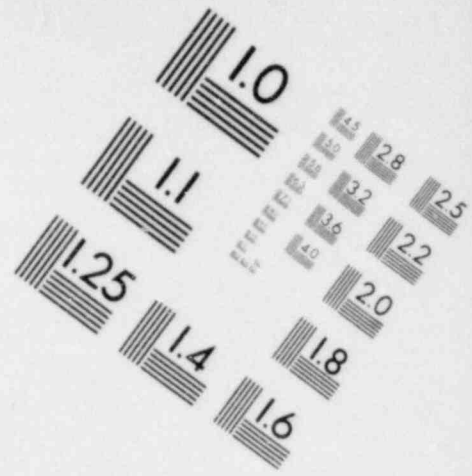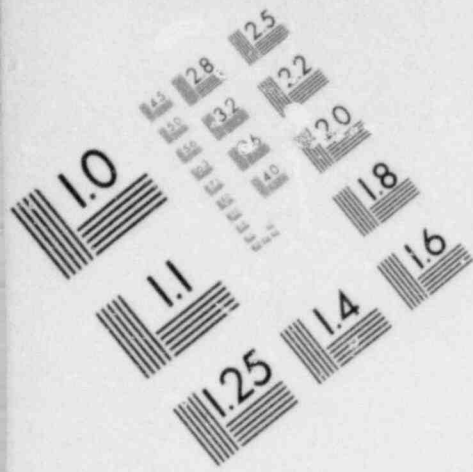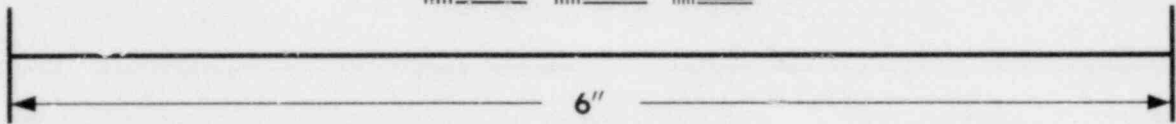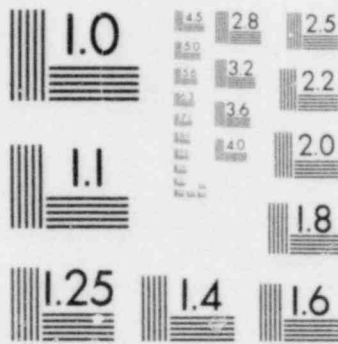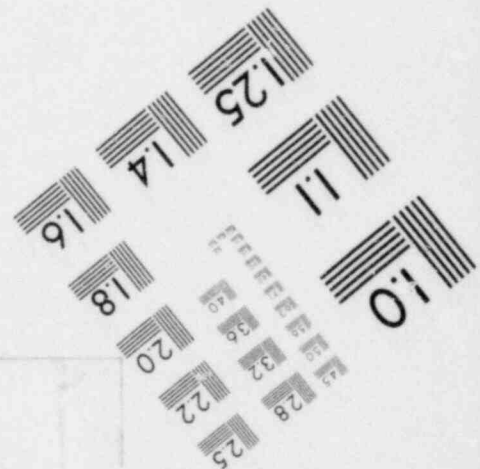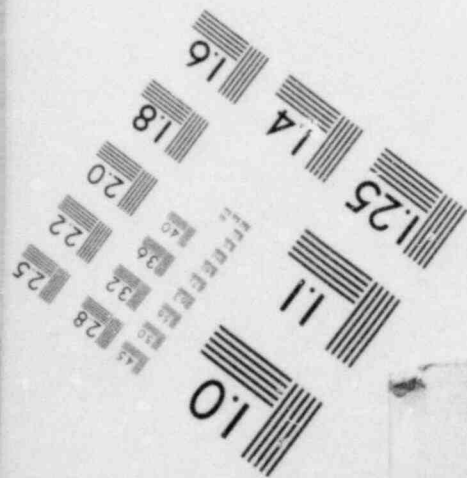
9.1.2 System Operating Modes

9.1.2.1 High Capacity/Pressure Venting

The accidents that the filtered vented containment system is designed
to mitigate will generate pressures of 35 psia or greater. At this
pressure, a maximum flow rate of 467,000 cfm could be sent to the
FVC system. There is a distinct flow path in the FVC system for
handling the combination of high pressure and high flow capacity.
The flow path which is utilized for cases where high capacity and
pressure are not present is described in section 9.1.2.2.

After the spray system has condensed the steam, approximately 300,000
cfm of gas will leave the pool area and be forced under pressure
through the sand filter directly above the pool. The intermediate
sand layers will capture and retain any particulate matter in the
gas stream. Upon passing through the sand and gravel layers, the
gas will enter the dilution plenum. In this mode of operation,
dilution fans will supply 300,000 cfm of outside air to lower the
hydrogen concentration.

The gas is forced through a bank of gravity dampers, into the discharge
tunnel. From there, the gas will be discharged through the discharge
vent stack.

9.1.2.2 Low Capacity/Pressure Venting

When the pressure in the postaccident containment has receded to
approximately 5 psig, it will no longer be possible or necessary
to pass the gas through the pool of water. By closing the pool
discharge vent gas header isolation valve and opening the pool bypass
vent gas header isolation valve, the system will be able to continue
venting the containment. This valve line-up puts the system in the
low capacity/pressure venting mode. In this mode of operation, the
vent gas discharges across the surface of the pool's water and travels
through the iodine removal sprays and on to the particulate sand
filter. The spray supply pump should be in operation in this mode
in order to cool the gas and remove iodine. After leaving the water
spray system, the air flow path is identical to the high pressure
mode of operation. However, the flow rate of gas from the containment
is substantially reduced.

9.1.2.3 Partial Vacuum Venting

If a containment isolation function has been degraded, the partial
vacuum venting mode of operation may be used.

In this mode, the containment vent gas flow path to the filtration
building is identical to the low capacity/pressure venting mode with
respect to the valve line-up. Operation of the iodine removal spray
subsystem will probably be required, depending upon conditions in
the affected containment.

The flow of gas will be from low flow to 60,000 cfm, depending upon the extent of degradation of the containment isolation. The gas flow will be drawn through the sand filter by use of the vacuum exhaust fans. The intermediate sand layers will capture and retain any particulate matter in the gas stream.

Upon passing through the sand and gravel layers, the gas will enter the dilution plenum. The dilution fans are not operating in this mode, and their discharge valves are shut. The gas is then drawn through the two vacuum exhaust fans and exhausted through the discharge tunnel to the discharge vent stack.

9.1.3 Evaluation

9.1.3.1 Physical Effectiveness

The sand filter used in the system is of proven design and is inherently sturdy. It is relatively self-sealing and has the ability to withstand shock loadings and large changes in upstream pressure without becoming inoperative. It has a high heat retention capacity, is fire resistant, inert to chemical attack, and its efficiency improves with life (to the point of breakthrough). However, sand filters require large area, have little design flexibility within cost and efficiency limits, and require higher operating pressures than other particulate filters. In addition, decontamination and disposal of the spent media would be difficult and expensive.

The mechanical ventilation system was selected to dilute the gases leaving the sand filter with outside air to keep the hydrogen concentration below the detonation threshold, since detonations within the filtration system are unacceptable. Dilution was selected instead of a controlled combustion because it is a more reliable method even though it increases the size of the system components. Other disadvantages of this active system are that it needs electrical power and periodic maintenance. In addition, dilution downstream of the sand filter does not prevent detonable mixtures from being formed in the air space between the water pool and the sand filter. The damage potential of a detonation in this region may be significant even though the filtration building is a relatively massive structure. Disadvantages of dilution upstream of the sand filter would include requiring a larger filter area and higher fan static pressure.

The iodine removal subsystem includes a water pool and a spray system which have the advantages of proven effectiveness for iodine capture by a buffered sodium thiosulfate spray, a proven effectiveness for steam condensation and heat removal by a water pool, gas stream prefiltering, and simple operation and control. The disadvantages include the need to periodically test and maintain the chemical solution and the need for an active, very reliable spray pump. The only other viable iodine removal system would require charcoal filters which have the relative advantage of longer maintenance or replacement intervals and the relative disadvantage of higher initial and replacement cost, the need for cooling both the gas upstream of the filters and the filters themselves and the need for filter fire

protection.

If a large fraction of the noncondensible gases in the containment were vented, excessive vacuum could result as the atmosphere is cooled and the steam condenses. No provisions are included in the design for adequate backflow through the vent or for any other replacement of noncondensible gases.

9.1.3.2 Schedule

At least 42 months are estimated for filtered vented containment engineering design and construction time, accounting for maximum potential benefit from overlap. This estimate includes no time for licensing or safety evaluations which could have a very significant impact on the schedule due to the controversial nature of this mitigation.

9.1.3.3 Cost

A total cost of   is estimated for design and construction of the filtered vented containment system for Sequoyah.

9.2 Additional Containment Volume

9.2.1 General Description

A vent system to an additional containment volume will also allow high temperature and pressure gases to be vented from either containment after an accident. However, the gas stream will not be filtered and released from a stack as in the FVC system described in section 9.1 but will be contained within the added building as it is vented.

The system as shown in figure 9-3 will penetrate each containment structure with one 48-inch line. Each line has three isolation valves and a rupture disk as in the FVC system. The lines from each containment are routed underground and tie into a common 48-inch line which then continues underground to the 1.5 million ft^3 added containment structure. This additional volume will serve as a mass and energy sink to reduce the pressure loading on the original, vented containment structure.

9.2.2 System Operating Mode

The vent to the additional containment allows both high capacity/pressure venting and low capacity/pressure venting through the same flow path since no major pressure drops are encountered. No valve realignment is necessary since the original containment will vent until the flow stops when the pressure are equalized. At this time, some necessary backflow will probably occur as the original containment cools further. No provision for partial vacuum venting has been included in this design.
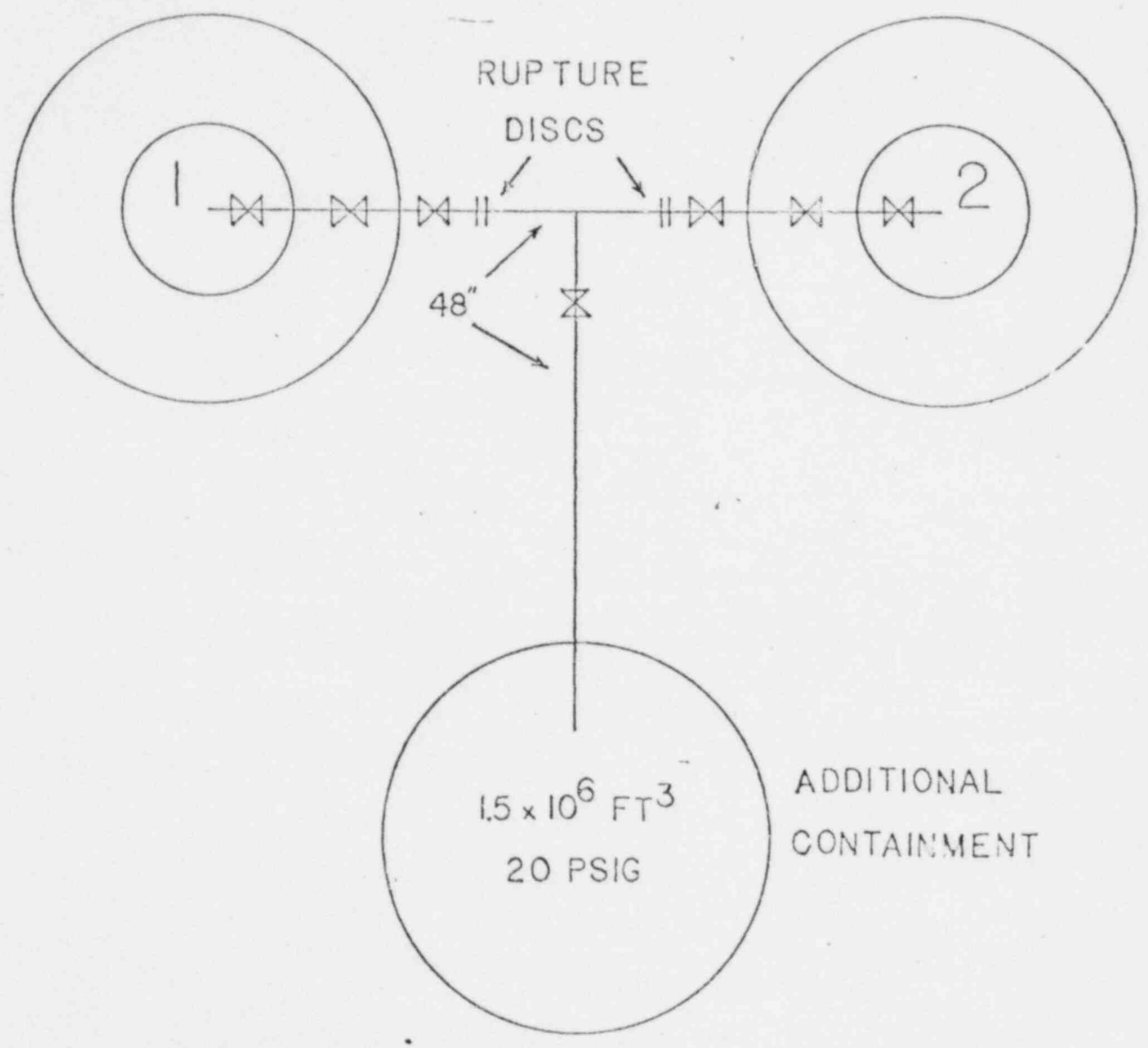
ADDITIONAL
CONTAINMENT

RUPTURE
DISCS

48"

1.5 × 10^6 FT^3
20 PSIG

ADDITIONAL
CONTAINMENT

FIGURE 9-3

9.2.3 Evaluation

9.2.3.1 Physical Effectiveness

Venting to an additional containment volume should be very reliable using this design. Other than prealigning the containment isolation valves, no active components are required for the vent system to function during the entire course of the event. Increasing the volume of the additional containment to three or four million ft^3 should be considered since the pressure reduction with the 1.5 million ft^3 volume was not adequate for all the accident sequences evaluated.

9.2.3.2 Schedule

At least 36 months are estimated for the additional containment vent engineering design and construction time accounting for maximum potential benefit from overlap. This estimate includes no time for licensing or safety evaluations.

9.2.3.3 Cost

A total cost of   is estimated for design and construction of the additional containment vent for Sequoyah.

9.3 Coupled Containment

9.3.1 General Description

The coupled containment vent system will allow high temperature and pressure gases to be vented from either containment to the other after an accident. The gas stream will not be filtered and released as described in section 9.1 but will be contained as it is vented similar to the additional containment as described in section 9.2.

The system as shown in figure 9-4 will penetrate each containment structure with one 24-inch line. Each line has three isolation valves and a rupture disk as in the FVC system. The line between the containments is routed underground. The coupled containment will serve as a mass and energy sink with both active and passive heat removal to reduce the pressure loading on the original containment experiencing the accident.

9.3.2 System Operating Mode

Similar to the vent to the additional containment, the vent to the coupled containment allows both high capacity/pressure venting and low capacity/pressure venting, venting until the containment pressures are equalized, and necessary backflow. Again, no provision for partial vacuum venting has been included in this design.

9.3.3 Evaluation

9.3.3.1 Physical Effectiveness

Venting to the coupled containment would not be as reliable as venting to an additional containment since more isolation valves must be opened in the vent path. In addition, operator action from both units
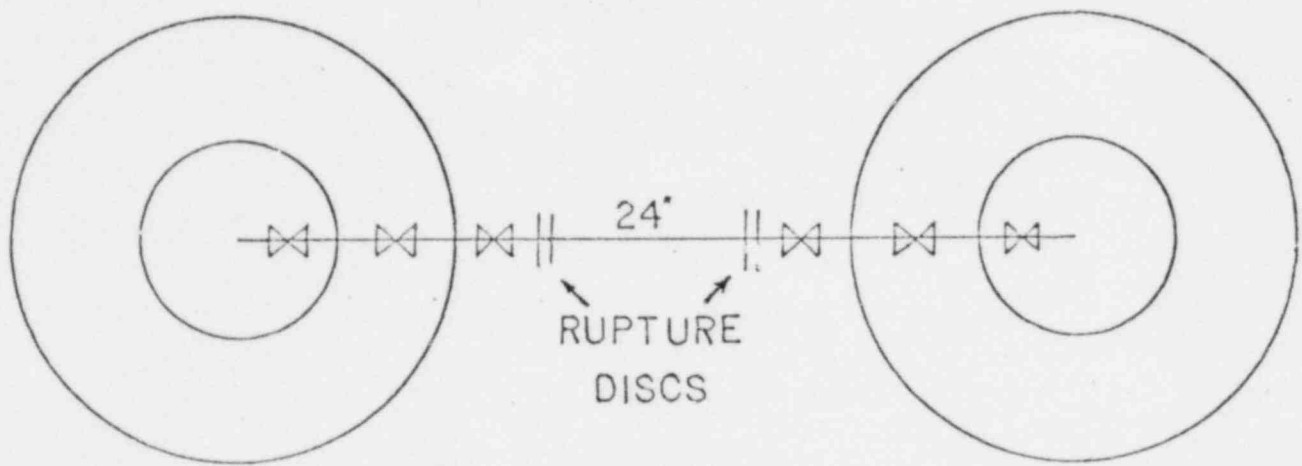
9-9

COUPLED
CONTAINMENT

FIGURE 9-4

is required to prealign both sets of isolation valves. However, no other active components are required for venting. Increasing the vent size to 48 inches should be considered to improve its effectiveness.

9.3.3.2 Schedule

At least 15 months are estimated for the coupled containment engineering design and construction time accounting for maximum potential benefit from overlap. This estimate includes no time for licensing or safety evaluations which could have a significant impact on the schedule.

9.3.3.3 Cost

A total cost of   is estimated for design and construction of the coupled vent for Sequoyah.

9.4 Augmented Containment Cooling

9.4.1 General Description

Augmented containment cooling was accomplished in this conceptual design by containment air cooling. It was proposed in the design that the existing upper and lower compartment air coolers (see figure 9-5) be modified so that they would be environmentally qualified for the accident conditions. None of the air coolers are currently claimed to serve for postaccident heat removal. To qualify the coolers, new fan motors and controls would be required along with some revisions to the present ERCW control logic.

9.4.2 System Operating Mode

The augmented cooling from the containment air coolers will be automatically initiated upon signals of high containment pressure and temperature after a LOCA, without waiting for indication of hydrogen generation. The coolers will constantly be removing heat at their maximum rated capacity and will be available to mitigate heat added during potential hydrogen combustion.

9.4.3 Evaluation

9.4.3.1 Physical Effectiveness

The cooling from the four upper and four lower compartment air coolers should be reliable since they receive trained power and are seismically qualified. However, the active components (motors, controls) may not survive in the elevated temperature following a postulated hydrogen burn. In addition, the coolers' limited short term heat removal capacity is not effective for the rapid energy addition rates that could result from some hydrogen burns.

9.4.3.2 Schedule

At least 18 months are estimated for the augmented cooling engineering

FIGURE 9-5

POWERCODE
REACTOR BUILDING
FLOW DIAGRAM
HEATING AND VENTILATING
AIR FLOW

SEQUOYAH NUCLEAR PLANT
TENNESSEE VALLEY AUTHORITY
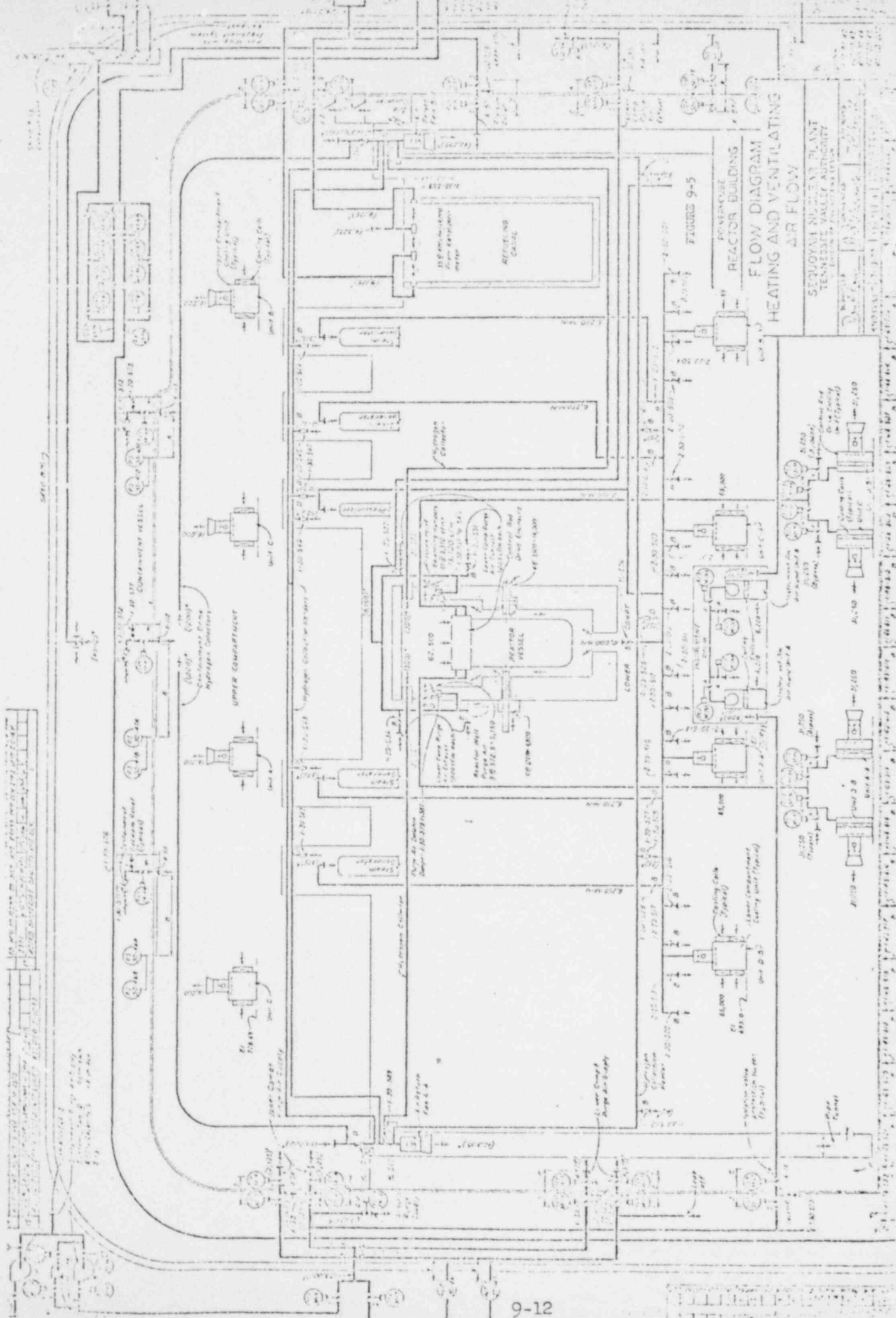
design, procurement, and construction time accounting for maximum potential benefit from overlap.

9.4.3.3 Cost

A total cost of _____ is estimated for design and construction of the augmented cooling at Sequoyah.

9.5 Containment Inerting

9.5.1 General Description

The containment inerting system designed for this study prevents combustible hydrogen-air mixtures from being formed after a LOCA by replacing the air in the containment with nitrogen when the reactor is brought to power and by maintaining this inert atmosphere during normal operation. Liquid nitrogen, stored in tanks onsite, is vaporized and blown into the containment using the present containment purge air supply system, which can be modified for more efficient supply and exhaust.

The system, as shown in figure 9-6, takes advantage of the fans and supply and exhaust ducts of the present 28,000 cfm capacity purge system. Supplying nitrogen at this rate will replace the 1.2 million ft^3 of air in the containment building every 45 minutes. It was estimated that eight to ten volume changes would be necessary to reduce the remaining oxygen concentration to the design level of 4 percent. This would require about 140,000 gallons of liquid nitrogen for each period of inerting so tank storage capacity of 160,000 gallons per unit was included in the design. Vaporizing nitrogen for the 28,000 cfm flow rate requires a heating unit of 8.5 MW capacity.

After vaporization, the nitrogen enters the purge air system upstream of the pair of supply fans which discharge into the containment. The present distribution system could be modified by routing ring headers around the upper compartment and around the lower compartment inside and outside the crane wall with branch lines into compartments with poor circulation. Nitrogen to the ice condenser compartment must be supplied at $15^\circ F$ in a separate header to prevent ice sublimation as much as possible. Provisions were made in the design to deinert the instrument room separately to allow freer access while the rest of the containment is still inerted. The present exhaust system could be modified by taking suction from the present hydrogen collection system which draws air from all the major compartments. Some mixing fans would probably be required, and any additional modifications could be made after an onsite survey of the containment.

Since automatic control of the inerting system would be very difficult to design and operate, the design specifies manual control by the operator. Extensive information on local gas concentrations would be provided so the operator could adjust the flow rates as necessary to ensure good mixing.
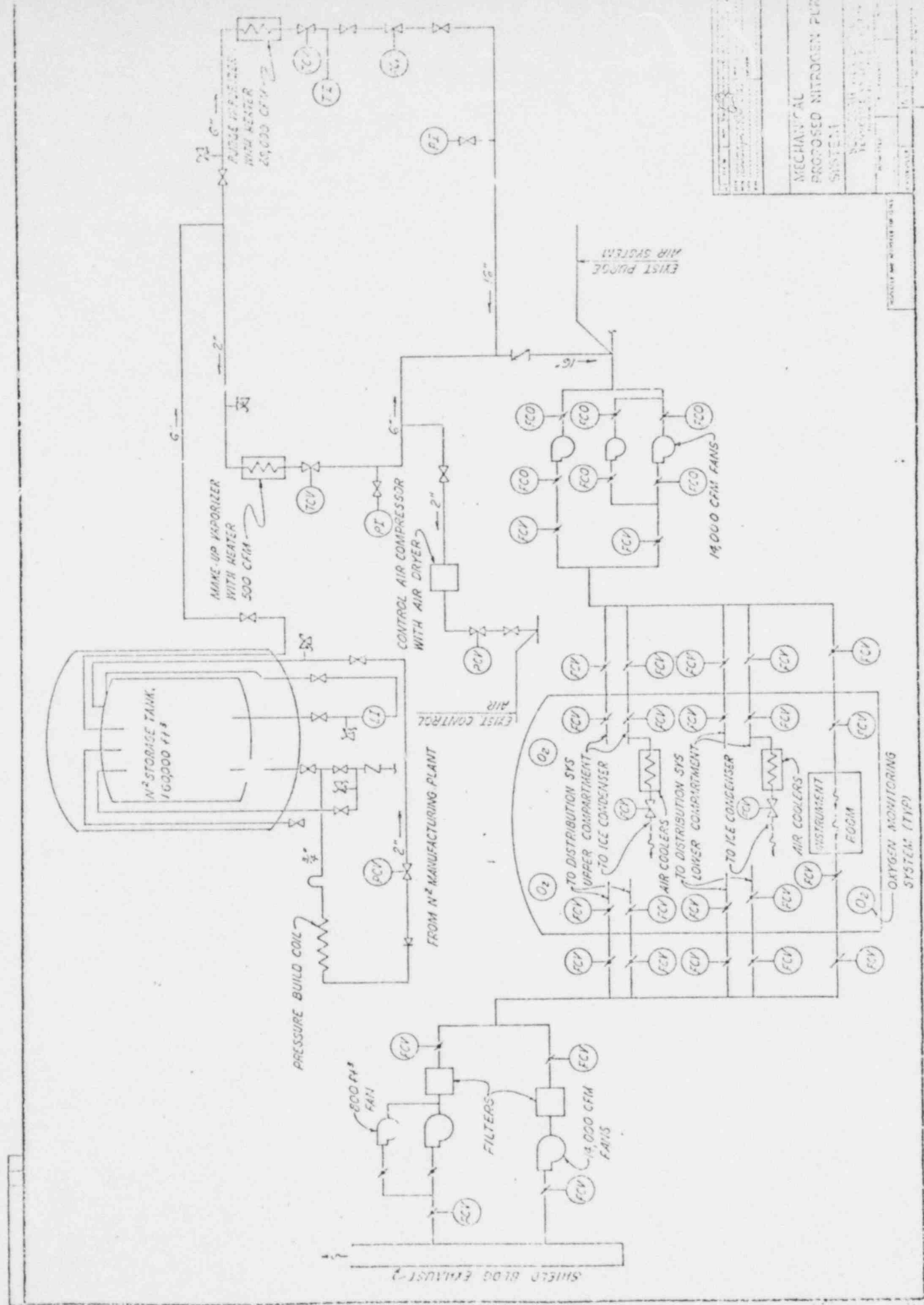
MECHANICAL
PROPOSED NITROGEN PUR
SYSTEM

9-14

Other modifications to allow effective inerting would change the present nonessential control and service air system to nitrogen inside containment. This would require a separate nitrogen makeup supply with its own 150 kW vaporizer. The four pressurizer safety and pressurizer spray valves that require essential air could be replaced with nonair-operated valves.

Personnel access for ice condenser equipment inspection and maintenance (which occurs daily) and other purposes while the containment is inerted would require portable air packs. Access even when the containment is deinerted would require portable oxygen monitors to avoid local pocketing of nitrogen.

The design included provisions for maintaining an inert nitrogen atmosphere in an additional containment building of 1.5 million ft^3 as described in section 9.2. The amount of stored nitrogen would be similar, but the design flow rate would be much less since no forced downtime is involved. The supply, exhaust, mixing, and control systems would be simpler since the added containment building would not be as compartmentalized.

9.5.2 System Operating Modes

Nitrogen to inert the containment, supplied at the maximum flow rate of 28,000 cfm, will require six to eight hours to reduce the oxygen concentration below 4 percent. Deinerting with air to reach oxygen levels of 16-18 percent may take slightly less time at that flow rate. If supply and exhaust at these large flow rates is not allowed due to licensing restrictions (e.g., if the reactor is at power) correspondingly longer times would be required. The containment instrument room is capable of being deinerted separately in this design. A constant makeup flow of nitrogen will be required for the valve control system inside containment.

9.5.3 Evaluation

9.5.3.1 Physical Effectiveness

This design has an elaborate supply, exhaust, and monitoring system and should be able to effectively inert and deinert the containment. The high flow rate of 28,000 cfm would allow fairly rapid inerting, but may not be practical to achieve in a nitrogen supply system. Oxygen pockets may still exist after inerting in compartments with poor circulation but should not be significant in the event of a hydrogen release. Nitrogen pockets remaining after deinerting could be significant to worker life safety. Deinerting the instrument room separately would not be very effective without further modifications to the room to maintain a breathable atmosphere. Any inerting system that disturbs the ice condenser atmosphere will increase the sublimation rate, but the cold air supplied by this system should have as little impact as possible. Any inerting system that maintains the inert atmosphere during personnel access is very hazardous to life as well as causing inefficient working conditions and limited access durations (see Section 10.3.1).

9.5.3.2 Schedule

At least 30 months are estimated for the containment inerting system engineering design and construction time accounting for maximum potential benefit for overlap. This estimate includes no time for safety evaluations.

9.5.3.3 Cost

A total cost of   is estimated for the design and construction of the containment inerting system for Sequoyah and an additi   cost of   is estimated for the design and construction of the inerting system for an additional containment. These costs include   and   , respectively, for the cost of enough nitrogen to inert the two present containments and the additional containment once.

9.6 Halon Injection

9.6.1 General Description

The containment halon injection system prevents combustible hydrogen air mixtures from being formed after a LOCA by adding enough Halon 1301 (C Br F$_3$) postaccident to occupy at least 31.4 percent of the containment volume. Liquid halon, stored in tanks onsite, is supplied to four containment halon ring headers and injected and vaporized through nozzles.

To postinert the containment would require 11,000 gallons of halon which is divided among four 750 ft^3 tanks, one per header. Each halon tank has its own 750 ft^3 nitrogen accumulator at 650 psig to to supply injection pressure. The halon tank level is monitored to automatically shut the isolation valves upon low tank level to reduce the amount of nitrogen injected into the containment.

The halon is supplied to the containment through three individual 6-inch ring headers in the upper compartment and one in the lower compartment. Smaller branch lines will be provided into the various enclosed areas of the lower compartment. At least one nozzle will be provided for about every 2000 ft^3 of containment free volume.

9.6.2 System Operating Mode

The containment isolation valves betwen the halon tanks and the ring headers would be automatically opened upon a high containment pressure signal followed by a low reactor vessel water level signal. However, manual operation of another isolation valve at each tank would be required before halon could begin to be injected. This is to reduce the potential for inadvertent actuation. The halon would be injected in a maximum of 30 minutes after an accident, assumming it is all vaporized in the line before it passes through the nozzles. The flow would probably be somewhat faster in practice.

9.6.3 Evaluation

9.6.3.1 Physical Effectiveness

The postaccident injection rate in this design should be fast enough to effectively inert the containment before significant amounts of hydrogen are released if the manual actuation by the operator follows the automatic trip fairly quickly. Operation of the air return fans should promote adequate mixing so that the hazards from any remaining noninerted areas are insignificant. Inadvertent actuation of the halon injection system as designed could be catastrophic. Unless sufficient heat is being added to the containment atmosphere (such as occurs following a LOCA), the heat removed to vaporize the injected halon may result in a drastic depressurization of the containment. In addition, the halon injection occurs at a higher rate when sufficient heat of vaporization is not readily available. Further analysis would be required to size any additional vacuum breakers or halon heaters that might be necessary. Inadvertent actuation when personnel are present could also be hazardous if portable air packs were not immediately available. This design also suffers from the environmental and cleanup problems that would result from any massive injection of halon.

9.6.3.2 Schedule

At least 30 months are estimated for the halon injection system engineering design and construction time accounting for maximum potential benefit from overlap. This estimate includes no time for licensing or safety evaluations.

9.6.3.3 Cost

A total cost of   is estimated for design and construction of the halon injection system for Sequoyah.

9.7 Ignition Sources

9.7.1 General Description

The design for controlled burning of any hydrogen released after a LOCA, before a detonable concentration can be reached, includes a system of ignition sources and hydrogen monitors located throughout the containment. The potential hydrogen release points were located and the ensuing circulation paths were analyzed. Ignition sources were then located to efficiently burn the hydrogen.

Since mixing helps prevent locally high concentrations of hydrogen, aids in completing the combustion process, and allows the maximum use of the containment heat sinks, the design for controlled burning specified continuous operation of the air return fans with the hydrogen collection system after an accident. Containment sprays would also promote good mixing as well as aiding in removal of heat from the burning.

The ignition system consists of 12 igniters placed in various relatively open areas within the containment as shown in figures 9-7 through 9-10. Any hydrogen release from a rupture in the reactor

PLAN-LOWER COMPARTMENT

IGNITER LOCATIONS

(1) REACTOR COMPARTMENT	(6) STEAM GEN. CELL	(11) UPPER CAMPARTMENT
(2) LOWER COMPARTMENT	(7) STEAM GEN. CELL	(12) UPPER COMPARTMENT
(3) LOWER COMPARTMENT	(8) STEAM GEN. CELL	(13) REACTOR HEAD VENT
(4) LOWER COMPARTMENT	(9) STEAM GEN. CELL	(14) PRESSURIZER RELIEF TANK VENT
(5) PRESSURIZER CELL	(10) UPPER COMPARTMENT	

FIGURE 9-7. CONTAINMENT PLAN AT RPV ELEVATION - IGNITER LOCATIONS

IGNITER LOCATIONS   SECTION D12-D12

(1) REACTOR COMPARTMENT	(6) STEAM GEN. CELL	(11) UPPER COMPARTMENT
(2) LOWER COMPARTMENT	(7) STEAM GEN. CELL	(12) UPPER COMPARTMENT
(3) LOWER COMPARTMENT	(8) STEAM GEN. CELL	(13) REACTOR HEAD VENT
(4) LOWER COMPARTMENT	(9) STEAM GEN. CELL	(14) PRESSURIZER RELIEF
(5) PRESSURIZER CELL	(10) UPPER COMPARTMENT	TANK VENT

FIGURE 9-8 CONTAINMENT ELEVATION WITH PRESSURIZER - IGNITER LOCATION
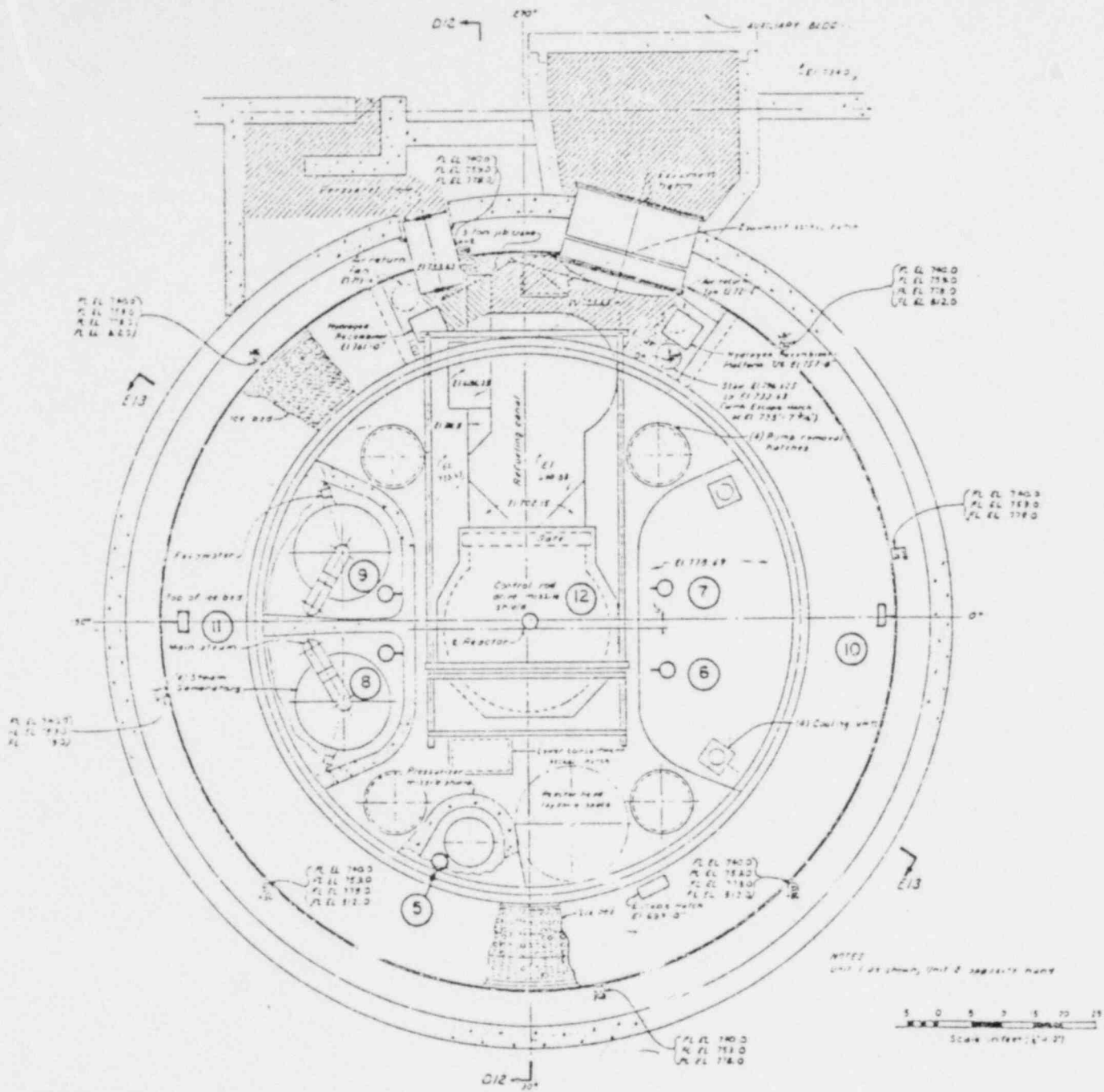
PLAN-UPPER COMPARTMENT

IGNITER LOCATIONS

①	REACTOR COMPARTMENT	⑥	STEAM GEN. CELL	⑪	UPPER CAMPARTMENT
②	LOWER COMPARTMENT	⑦	STEAM GEN. CELL	⑫	UPPER COMPARTMENT
③	LOWER COMPARTMENT	⑧	STEAM GEN. CELL	⑬	REACTOR HEAD VENT
④	LOWER COMPARTMENT	⑨	STEAM GEN. CELL	⑭	PRESSURIZER RELIEF
⑤	PRESSURIZER CELL	⑩	UPPER COMPARTMENT		TANK VENT

FIGURE 9-9 CONTAINMENT PLAN AT ICE CONDENSER ELEVATION - IGNITER LOCATION
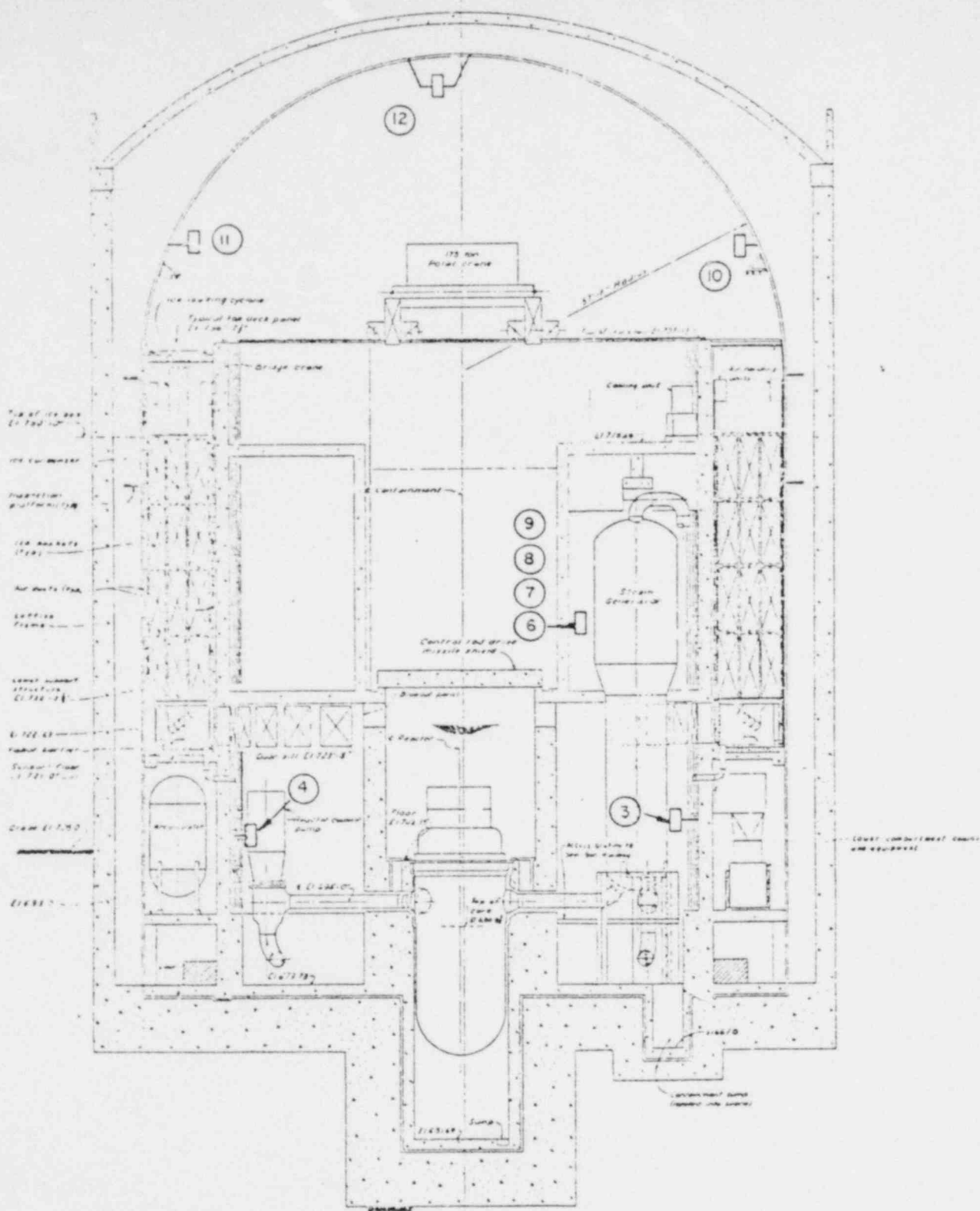
IGNITER LOCATIONS

SECTION E13-E13

1	REACTOR COMPARTMENT	6	STEAM GEN. CELL	11	UPPER COMPARTMENT
2	LOWER COMPARTMENT	7	STEAM GEN. CELL	12	UPPER COMPARTMENT
3	LOWER COMPARTMENT	8	STEAM GEN. CELL	13	REACTOR HEAD VENT
4	LOWER COMPARTMENT	9	STEAM GEN. CELL	14	PRESSURIZER RELIEF TANK VENT
5	PRESSURIZER CELL	10	UPPER COMPARTMENT		

FIGURE 9-10. CONTAINMENT ELEVATION WITH STEAM GENERATOR - IGNITER LOCATION

primary loop or from pressure-relieving devices will occur in the lower compartment of the containment. Nine of the igniters are located in this portion of the containment to burn the hydrogen as soon as it reaches a combustible level. If the hydrogen is swept into the upper compartment, three igniters in this area will burn the hydrogen in the same manner. In addition to the 12 igniters, two waste gas burners with integral igniters are located in the lower compartment above the pressurizer relief tank (see figure 9-11), where they will burn hydrogen from the reactor head vent or the relief tank vent.

Igniting hydrogen-air-steam mixtures at low concentrations of hydrogen (4-8 percent) requires a uniform, well controlled, hot ignition source. The design specified an option of continuously operated glow plugs or cyclically operated arc igniters. These should be equivalent to flame igniters and easily meet the minimum flash point temperature of hydrogen-air mixtures. The igniters will be shielded to prevent water sprays from impinging on them.

9.7.2 System Operating Modes

The igniters should be started automatically upon an accident signal or manually if the hydrogen analyzers indicate the presence of hydrogen below detonable concentrations. Hydrogen levels should be individually monitored at each igniter location with indication in the main control room. Great care must be taken to ensure that igniters are not operated when the potential for hydrogen detonation exists.

9.7.3 Evaluation

9.7.3.1 Physical Effectiveness

The air return fans should provide mixing between the upper and lower compartments that is adequate for dispersion and combustion of the hydrogen. Some additional mixing fans may be required in localized dead-ended volumes. With adequate mixing, the igniters as located in this design should ensure that combustion would be fairly complete for low hydrogen release rates and should burn most of the hydrogen for larger releases. The intermittent energy addition that would result from these stepwise burns as the hydrogen reaches combustible concentrations should be within the containment heat removal capacity. Careful hydrogen level monitoring must be performed for proper igniter operation, and careful control of the flow to the waste gas burners would also be required. Local flame and temperature effects should be considered for each of the selected igniter locations. Inadvertent actuation when excessive hydrogen or other flammable materials (solvents, paints, etc.) are present could be hazardous.

9.7.3.2 Schedule

At least 18 months are estimated for the ignition system engineering design and construction time accounting for maximum potential benefit from overlap.

ELECTRIC
LEAD

IGNITOR

IRON / CONSTANTAN
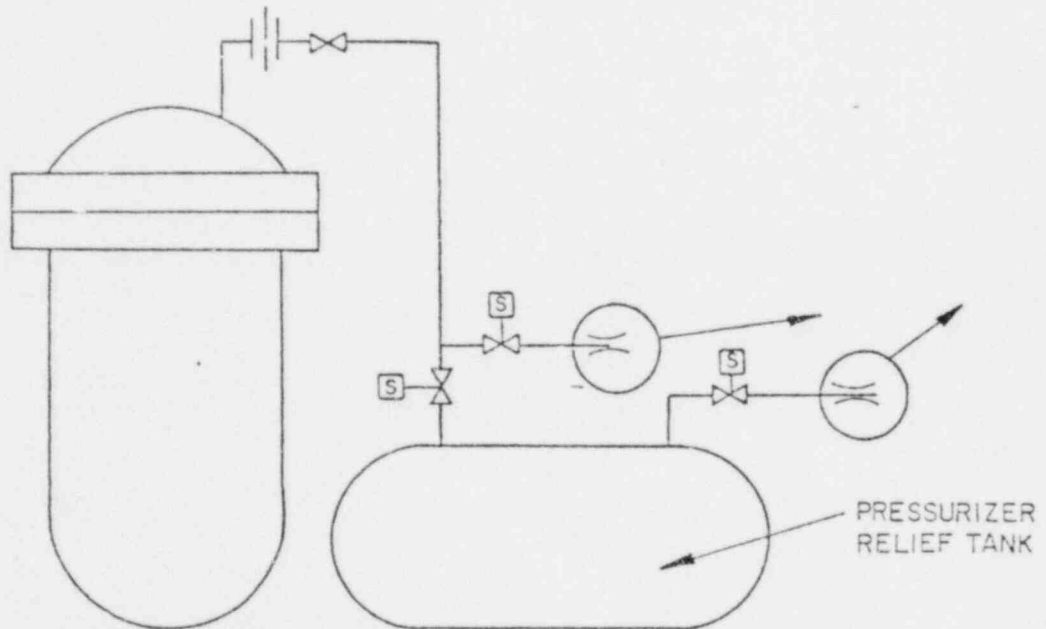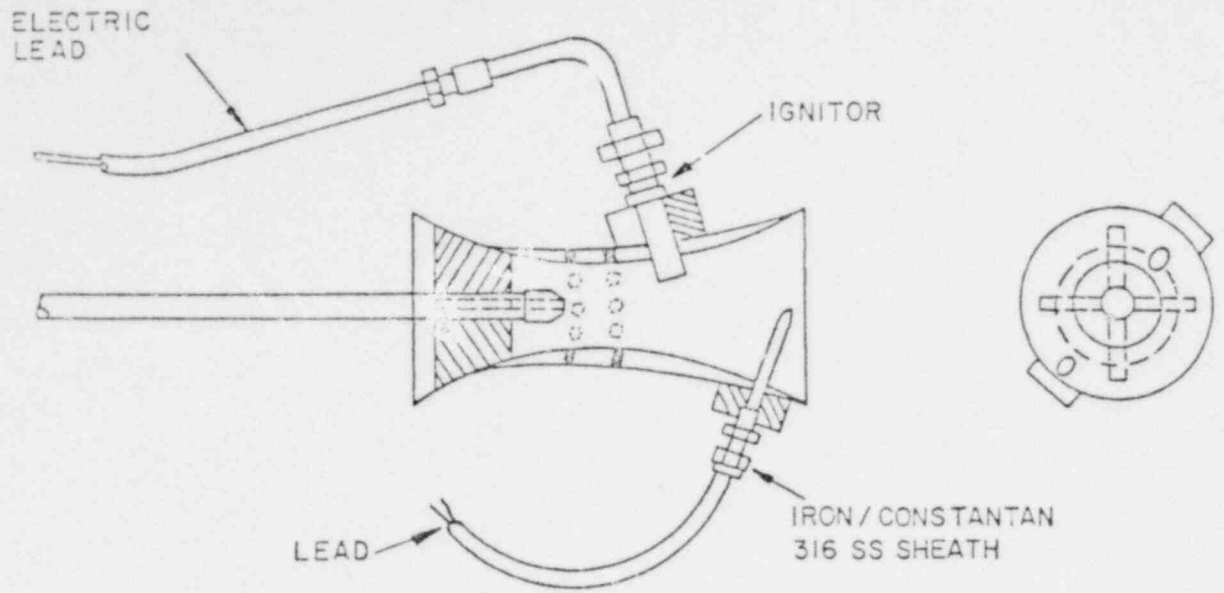316 SS SHEATH

LEAD

PRESSURIZER
RELIEF TANK

FIGURE 9-11. HYDROGEN IGNITER FOR REACTOR HEAD VENT

9-23

9.7.3.3 Cost

A total cost of   is estimated for design and constructio
of the ignition system at Sequoyah.

DA02:NSAHYD.M5

10.0 Evaluation of the Alternative Concepts

The evaluation of the alternative concepts included the potential benefits, problems, and costs. This evaluation stressed the relative merits of each concept; definitive and quantitative evaluations would require additional design detail, analyses, and research. Nonetheless, this evaluation does provide a good comparison of relative merit. Limitations of this study are discussed in section 12.

10.1 Overview

No individual concept or combination of concepts is without significant drawbacks or uncertainties. However, several appear technically feasible and relatively effective at mitigating the effects of core damage.

The concepts for hydrogen burn suppression (nitrogen inerting, halon, and ignition sources) provide varying degrees of protection from the effects of hydrogen combustion, but provide essentially no protection from overpressure due to noncondensible gases. These concepts are discussed in section 10.3.

The vented containment concepts (filtered venting, additional containment, and coupled containment) provide less protection from the effects of hydrogen but provide substantial protection from overpressure due to noncondensible gases. These concepts are discussed in section 10.4.

10.2 Evaluation Basis

The following factors were considered in judging the relative merits of each concept:

Effectiveness - The ability of the concept to accommodate the effects of core degradation, including hydrogen generation.

Technical Feasibility - The present capability to implement each concept, emphasizing any serious technical problems.

Additional Risk - Hazards to nuclear safety that may accompany each concept.

Reliability - The degree of assurance that the system would be available and used when needed. This includes the inherent reliability of the system and the need (and time and information available) for operator action.

Cost - Includes capital costs, reductions in plant availability, and operating cost. Special research and analysis costs have not been included. These may be large and cannot be estimated at this time.

10.3 Concepts to Prevent or Minimize Hydrogen Combustion

The three concepts evaluated are inerting the containment (with nitrogen during operation), halon (postaccident inerting), and

ignition sources (controlled burning). The first two prevent
combustion, while the latter attempts to remove hydrogen to prevent
the rapid combustion of a large accumulation. These three concepts
do not require major new structures, but do require a significant
amount of equipment. Since our evaluation indicates that (for events
where containment cooling is available) the plant can withstand the
releases of energy and noncondensibles (except the energy of hydrogen
burning rapidly) for events up to core melt, these concepts can
provide substantial protection for noncore melt accidents. These
concepts do not provide any protection against overpressure failure
due to the steam and gas releases during a core melt.

10.3.1 Containment Inerting with Nitrogen

Effectiveness - No significant sources of oxygen into containment
during operation have been identified (except the compressed air
system which would be modified to eliminate this potential source).
Therefore, inerting should be fully effective at preventing
postaccident hydrogen combustion. The additional partial pressure
of hydrogen could lead to containment pressures above design, but
still well within the capability of the containment. The increased
containment pressure could lead to some increase in leakage, but the
secondary containment features at Sequoyah would minimize the impact.

Technical Feasibility - Based on our experience at Browns Ferry,
this concept is feasible, but technical problems exist. The
compartmentalization at Sequoyah will require extensive measures
to eliminate trapped pockets of oxygen. Of more concern is the
problem of reduced access for inspection and maintenance.

Additional Risk - An inerted ice condenser containment causes a major
additional risk to the operator. Ice condenser containments require
a significant increase in containment entries over other types of
containments. This increase is due to both additional surveillance
required by technical specifications and additional inspection and
maintenance on the ice condenser system components. Technical
specifications require weekly visual inspections, while current
experience at Sequoyah and other ice condenser plants indicates that
containment entries are required every two days to perform maintenance
on the ice condenser systems, particularly the air handling units.
Based on this high rate of maintenance, entries are required daily
to inspect these units. In addition, ice weighing activities are
expected to require several weeks per year with a crew of people
inside all day. Even assuming that all possible plant modifications
are made to reduce access requirements for maintenance, it is
estimated that personnel entries would still be required at least
16 times per year with an associated forced downtime of 15 percent.
Industrial experience has shown that personnel occupancy in inerted
enclosures can be extremely hazardous. Even while deinerted, there
is a risk of a pocket of nitrogen existing that could threaten
workers.

In addition, the reduced access for inspection and maintenance will
reduce the reliability of other plant systems. Experience has
indicated that other (nonice condenser) maintenance will require

containment entry on a weekly basis. This includes instrument, pump, and valve repairs. Reduced maintenance could increase the possibility or severity of accidents. Often, it is desirable to enter the containment for manual valve operations in the event remote operation fails. Reduced access will inhibit such repair activities in the event of nonpipe break accidents. This additional risk appears significant.

Reliability - Since inerting would take place prior to going to power, after which the system is passive and the status of containment can be monitored, this concept is judged to be very reliable.

Cost - While the capital costs are moderate, the operational costs are very high due to the cost of large quantities of nitrogen, the additional downtime for inerting/deinerting, and the forced outages for increased maintenance in containment. All costs are in 1980 dollars and for a 2-unit plant.

    Capital Cost   -
    Nitrogen Cost   -   /yr.
    Inert/Deinert Loss of Availability -   /yr. (minimum)

Additional downtime would be required to perform maintenance that cannot be done in an inerted containment. Also, the periodic inerting and deinerting operations would increase the rate of ice loss in the ice condenser and force its earlier replacement with concurrent downtime of several months. Neither of these operating costs have been included in the estimate above but could be substantial.

10.3.2   Halon

Effectiveness - There is fairly good evidence that Halon 1301 can suppress hydrogen combustion. However, there is some question about its long term behavior in containment. If it decomposes or is removed by the containment sprays or ice, combustion could still take place, probably at a high hydrogen concentration. These questions must be answered before it can be concluded that halon would be effective. (Also, see Reliability below).

Technical Feasibility - The system is very similar to installed fire suppression systems and should not cause significant technical problems except in scaling up to such a large volume with the need for uniform injection and good mixing. Since steps have to be taken to prevent inadvertent operation (which could pose a life safety and containment pressure problem), and since the operator must have sufficient warning of a potential hydrogen release to actuate the system manually, the controls may be complicated. It is thought to be feasible.

Additional Risk - Halon can be a hazard to personnel. We judge this hazard to be small since the probability of inadvertent actuation can be kept low.

10-3

Halon itself can cause some risk of containment failure. We believe that the risk of containment overcooling during normal operation can be minimized. If halon were injected after a hydrogen burn or when high pressure exists in containment, the additional partial pressure (even accounting for the cooling effects of halon) could cause failure due to overpressure

In addition, the decomposition products of halon include certain acids, halogens, and other substances which have the potential for inducing cracking in stainless steel, degrading other equipment materials, and complicating any cleanup operations. This may ultimately prove to be so severe as to preclude its use.

Reliability - The system is very similar to installed fire suppression systems and should be reliable mechanically. However, since the system must be actuated by the operator, there is some risk that the operator will fail to recognize the need to actuate the system.

Cost - The principal costs are due to the halon, mechanical equipment, controls, and instrumentation.

Capital Cost -

10.3.3   Ignition Sources

Effectiveness - Ignition sources are effective only for those events where the hydrogen evolution and rate of burn will be slow (in the order of the heat removal capability of the containment sprays). Such conditions occur when the hydrogen concentration in containment is between 4 and 8 percent. Ignition would be most effective in the lower compartment because of the availability of heat removal in the ice condenser. However, it is not certain that reliable burning can be achieved in the lower compartment due to the presence of steam in large quantities and the relative lack of air. Above a hydrogen concentration of 8 percent, burn occurs very rapidly, producing energy at a rate which cannot be removed quickly by the containment cooling systems if a large quantity of hydrogen is available.

Technical Feasibility - We believe that the igniters can be developed although considerable research and testing will be needed. Further analysis is required to establish how many are needed and where to locate them. Controlled ignition of the effluent from the reactor vessel head vent prior to mixing in the containment may require a more sophisticated system than in open areas, but may be feasible.

Additional Risk - The igniters could cause combustion that otherwise might take place. We believe this risk is small since there are already mechanisms in containment which will probably cause ignition sooner or later. Local effects of controlled burning could be severe.

Reliability - At least some degree of operator action is
required. It appears that complex guidelines will be required
for which igniters to use, when to use them, and when not to use
them. The chance of operator error is therefore significant.

Cost - The major cost items are hardware, controls, and
instrumentation.

Capital Cost -

10.4   Concepts to Increase Containment Capacity for Overpressure Events

The four concepts evaluated are filtered vented containment (relief
through filters to atmosphere), additional containment (an added
containment building), coupled containment (connecting both
containments at a 2-unit site), and additional containment cooling
(heat exchangers to cool the containment air). The first three
provide protection against containment overpressure by
relieving some of the gases to the environment or another structure.
The last provide, protection by serving as a backup to the containmer
sprays and by removing steam to allow for more noncondensible gases.
All involve major construction and design efforts. The first three
provide limited protection from the effects of hydrogen by reducing
the hydrogen in containment and providing relief for slow to moderate
pressure increases.

10.4.1   Filtered Vented Containment

Effectiveness - Since venting is to the environment, this concept
provides unlimited total relief capacity. However, its effectiveness
is limited for rapid pressure transients because of practical vent
sizing which limits the rate of relief. The concept does not provide
complete mitigation since deliberate releases of radionuclides are
used to prevent against accidental releases due to containment
failure. In fact, the release of noble gases could be nearly as
severe as for containment failure; the release of halogens and
particulates can be reduced by about a factor of 100.

Effectiveness as a mitigator of hydrogen is limited to: those
events that are relatively slow such that operator action can take
place before any major pressure increase (to reduce the amount
of gases inside containment); and those events that result in slow
to moderate pressure increases.

It is believed the effectiveness in practice would be reduced due
to a natural reluctance by the operator to use a system which involves
a deliberate radioactive release of large magnitude. (A relief
duration of only a few seconds could lead to releases many times that
from the worst event presently analyzed for the design basis.)

Technical Feasibility - Overall, it is felt that the system could
be designed and constructed to work. There are several areas
where research and/or development are required, including:

1. The connection to containment is a difficult design problem.
    The principal concerns are modifying an existing design to

minimize loads on containment and ensuring the integrity of the relief line.

2. The choice and design of the heat removal facility (pool or sand/gravel bed) will require extensive analysis and possibly some research. Concerns include structural provisions for dynamic loads, draining of the bed, and long term behavior.

3. If hydrogen and oxygen mixtures must be handled, a hydrogen burning chamber or dilution system must be provided. It is not known at this time if such systems can be designed to function reliably.

4. The effectiveness of various types of particulate and iodine removal filters when subjected to large, high temperature, high moisture content flows should be demonstrated.

Additional Risks - This concept involves deliberate releases; therefore, it only reduces rather than eliminates risk. It is estimated that the whole body dose would be in excess of 900 REM in the low population zone compared to the Sequoyah FSAR LOCA whole body dose of 2.4 REM in the LPZ. In addition, venting could result in undesirable doses to the operators in the control room. Also, the system can lead to violation of containment by inadvertent operation when not needed. There is some possibility that if the system is used for less severe events in conjunction with containment sprays, the containment pressure could go sufficiently subatmospheric to threaten the containment structurally or the core cooling systems due to inadequate NPSH.

Reliability - The mechanical reliability is high because the system is essentially passive. Some designs employ active filtration, heat removal, or hydrogen control components, though. However, operator action is required to control the system. Considering the impact of using the system (deliberate releases) and the short amount of time in some postulated events for making the decisions, it is believed that the chance of misoperation or delayed operation could be significant.

Cost - The cost is largely in constructing the facility with some maintenance cost (not estimated).

Capital Cost -

10.4.2 Additional Containment

Effectivess - Since venting is to an additional containment, this concept provides a large but finite total relief capacity. Its effectiveness is limited for rapid pressure transients because of practical vent limits (but larger venting rates than for filtered vented containment can be handled). The concept does provide complete mitigation for those events within its capacity since no deliberate release to the environment occurs.

Effectiveness as a hydrogen mitigator is limited similarly as the

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filtered vented system to those events allowing operator prealignment
of isolation valves and involving slow to moderate pressure increases.
(It may be possible, however, to make the isolation valves automatic.)

Technical Feasibility - This concept is also believed to be feasible
and does not involve the technical problems of filtration. However,
the problems of containment/vent structural attachment and the
adequate treatment of the hydrogen in the effluent remain.

Additional Risks - No additional risks were identified.

Reliability - The mechanical reliability is high because the
isolation valves are the only active components required. However,
operator action is required to actuate the valves which introduces
some chance of delayed operation.

Cost - The cost is in constructing the vent system and the additional
containment structure.

        Capital Cost -

This cost is for a 1.5 million cubic foot containment. A containment
with more capacity (3 0 million cubic feet) may be required for
mitigating all core melt pressure effects. It would cost
approximately   .

10.4.3   Coupled Containment

Effectiveness - Since venting is to the other reactor containment
building, this concept provides a finite total relief capacity, even
though some benefit may be obtained from the other unit's heat removal
and pressure suppression systems. Again, the effectiveness is limited
for rapid pressure transients because of vent size limitations. The
coupled concept does provide complete mitigation for those events
within its capacity since no deliberate environmental release occurs.

Effectiveness for hydrogen mitigation is also limited to those events
allowing anticipatory operator actuation of isolation valves and
involving slow to moderate pressure increases.

Technical Feasibility - The coupled concept is believed to be
feasible, but still faces the structural problem of containment/vent
attachment without allowing excessive accident loads to be transmitted
between units.

Additional Risks - In principle, venting between containments should
not cause problems in the second unit that do not potentially already
exist in the first one. However, some equipment failure or
misoperation in the second unit could result due to the environmental
effects of the effluent from the first. This is especially likely
due to the high temperatures that could result from hydrogen
combustion. In any case, the detrimental effects of radioactive
contamination and reduction in accessibility, need for cleanup, etc.,
would be doubled.

Reliability - The mechanical reliability should be high since the isolation valves are the only active components. However, more valves are involved than for venting to an additional containment and anticipatory operator action from both units is required to actuate the valves. In addition, the significant impact of using the system (contaminating the other unit) suggests that the chance of misoperation or delayed operation could be significant.

Cost - While the cost in constructing the coupled vent system is moderate, the operational cost would be very high due to the additional downtime that would be forced. The units would have to be operated in conjunction since one could not be at power while the containment of the other was open because of maintenance, refueling, or initial construction. This could almost double the expected unit downtime as well as greatly complicating outage planning and personnel requirements.

Capital Cost -

10.4.4   Augmented Cooling

Effectiveness - The practical short term heat removal capacity of air coolers or additional containment sprays is limited to the moderate energy addition rates from partial or intermittent hydrogen burns. The air coolers or current sprays are ineffective in removing energy from rapid combustion.

Technical Feasibility - This concept is technically feasible since air cooling and sprays are currently used for containment heat removal, either normally or postaccident.

Additional Risks - No additional risks were identified.

Reliability - The mechanical reliability is fairly high. Operator actuation would not be required because the systems could initiate automatically on accident signals since no additional risks are involved.

Cost - The cost is in the initial installation of the equipment.

Capital Cost -

10.5   Concepts to Mitigate the Combined Effects of Hydrogen Combustion and Containment Overpressure

Concepts have been discussed in this section which could alleviate some aspects of core damage. Some concepts would prevent the failure of containment by eliminating the rapid energy released with hydrogen burns. Others attempt to prevent the failure of containment due to overpressure by removing mass and energy in large amounts. No single concept studied was found capable of preventing containment failure due to both mechanisms. Thus, combinations of the concepts were evaluated for mitigating the main effects of core melt.

A review of the individual concepts show that venting to an additional containment in combination with either halon injection or containment inerting with nitrogen may provide complete protection. Each of these combinations involves major construction, extremely high cost, and may drastically affect operations.

Effectiveness - When combined with venting to an additional containment, either halon injection or inerting with nitrogen appears to be capable of preventing containment failure due to either hydrogen combustion or overpressure. However, the additional containment would also have to be inerted or provided with a halon injection system, depending on the option selected for the original containment. This is because venting from a halon-injected or an inerted containment into an air-filled additional containment could alter the overall gas concentrations enough to make combustible mixtures possible.

Technical Feasibility - There is nothing inherent in either of the combinations which would preclude the use of the concepts in a combination. However, each of the concepts has serious technical problems which must be solved before they could be used, individually or in combination.

Additional Risks - There are no additional risks by using these concepts in combination beyond those identified for each separately.

Reliability - The combination with nitrogen inerting should be as reliable as the individual venting to an additional containment since the inerting is judged to be very reliable. However, the combination of halon injection and venting would not be as reliable since two separate active systems, each actuated manually by the operator, would be required to operate.

Cost - While the capital costs for both options are similar (for inerting/venting and   for halon/venting), the operational costs of inerting are very high as described in section 10.3.1.5.

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11.0 Conclusions and Recommendations

11.1 Conclusions

11.1.1   These studies have shown that the overall risk to the public from the Sequoyah plant is about the same as the NRC reference plant which has a dry containment and, thus, Sequoyah presents no special risk to the public.

11.1.2   The Sequoyah plant containment was found capable of withstanding a hydrogen burn from an amount of hydrogen generated in the oxidation of approximately 20 percent of the zircaloy fuel cladding without any additional mitigation.

11.1.3   All concepts studied in this report were found to have serious technical and/or operational problems which need extensive research and development before design work could proceed.

11.1.4   Containment inerting was found to be the most reliable of those concepts which control or prevent hydrogen, but it is extremely expensive, would likely cause personnel deaths, and may lead to reduced maintenance and safety. The halon suppression concept has few of these problems, but has serious technical concerns with its actual performance in a postaccident containment environment. An ignition system also has some technical questions requiring resolution and is not effective for complete reaction of the zircaloy fuel cladding.

11.1.5   Of those concepts which prevent containment overpressure, venting to an additional containment appears to be the best. Although it is more expensive than filtered venting, it is simpler and releases no radioactivity to the public. The major drawback of the filtered vented containment is a large dose to the public. The coupled containment concept is too restrictive on operation.

11.1.6   To mitigate the conseqences of an accident resulting in 100 percent zircaloy clad reaction with water or one that proceeds to complete core damage, two or more of the concepts described in this study must be used, at least one each from hydrogen control and one from overpressure prevention.

11.2 Recommendations

11.2.1   These studies have shown that (1) Sequoyah is comparable to the best plants operating in the U.S. in terms of risk to the public, (2) Sequoyah containment can withstand a burn of hydrogen equivalent to 20-percent metal-water reaction, or about the same as the NRC estimate of the hydrogen burned at Three Mile Island, and (3) all concepts of mitigation of hydrogen effects studies clearly need substantially more research and development. Based on these conclusions, we recommend that no commitments to specific mitigation devices be made at this time. Further investigation and development of the most promising concepts identified in this study or which may be proposed are recommended.

11.2.2   In the event that a commitment to incorporate some
mitigation of hydrogen effects becomes necessary, the following
approach is recommended:

Commit to an ignition system - Although this system is not
effective for 100-percent reaction of the zircaloy fuel cladding
and does not reduce the overall risk significantly, it may handle
hydrogen releases up to 60 percent metal-water reaction, which
is the upper range of the estimates of hydrogen generated at
Three Mile Island (TMI). This system has the advantages of short
implementation time following further developmental work, low
risk of jeopardizing current safety, relatively low cost, no
effect on operation, and can be treated as a temporary measure
until the NRC has gone to rulemaking on hydrogen.

11.2.3   In the event that a commitment to incorporate mitigation
of hydrogen effects for 100-percent zircaloy cladding-water reaction
and/or complete core degradation becomes necessary, the following
approach is recommended:

The best mitigation requires the combination of venting with
prevention of the hydrogen burn. We recommend that this option
be committed to, only if no choice is given. The best concepts
when considering effectiveness, life safety, operational
problems, costs, and improvement of risk to the public is the
halon suppression system plus venting to an additional
containment; however, the halon system has serious technical
problems which may not be resolvable before design must begin.
Therefore, at this time, nitrogen inerting plus venting to an
additional containment must be recommended. It is reliable,
but has very serious operational problems, is extremely
expensive, and poses some safety problems.

11.2.4   As a result of our findings, we recommend that TVA
discourage NRC from use of the filtered venting concept as the
requirement of the rulemaking process because of the large radiation
dose to the public, which may be in excess of 900 REM body dose in
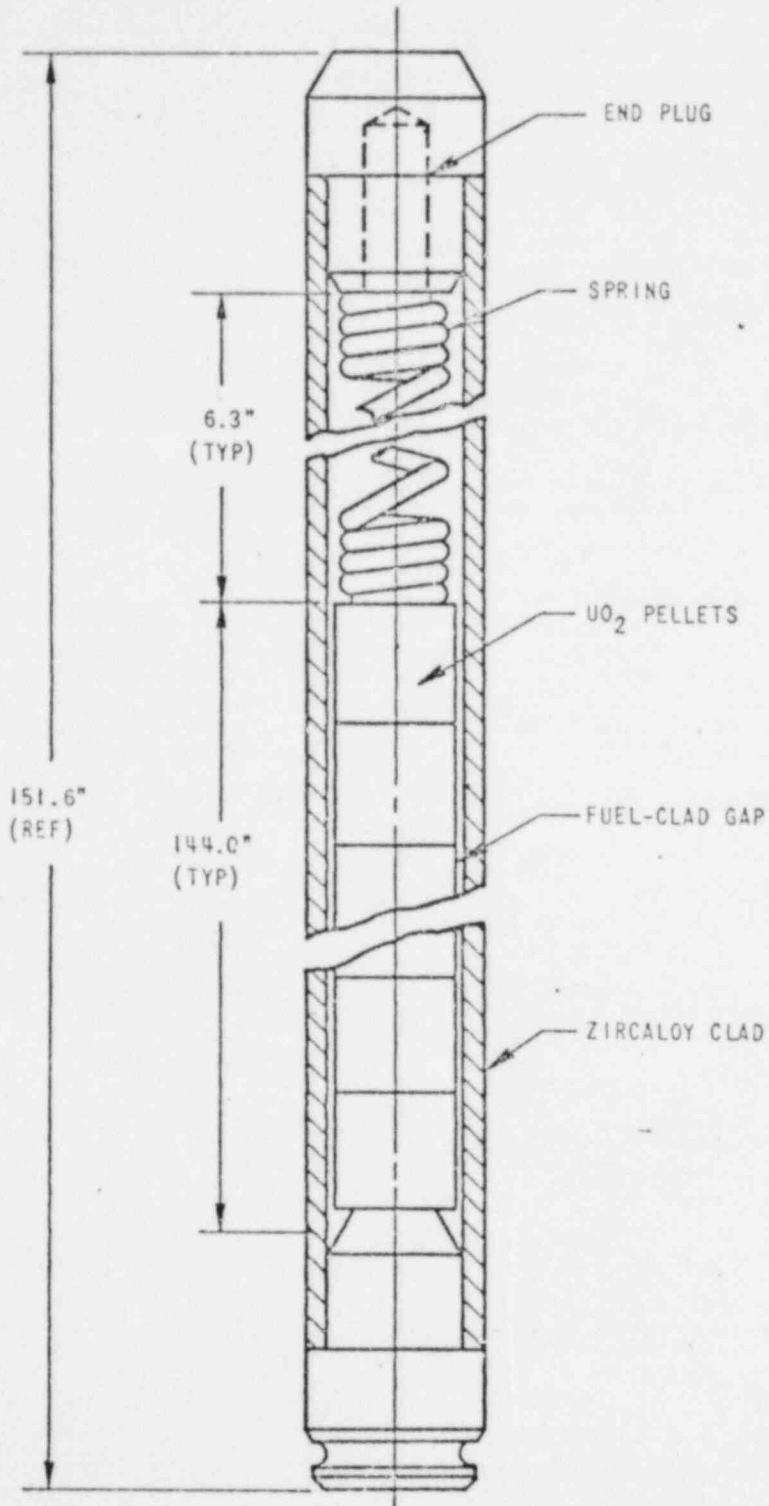the low population zone a Sequoyah.

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APPENDIX A

CURRENT DESIGN AND BASES

A.1   The Reactor Core, Fuel, and Coolant System

The reactor core is comprised of an array of fuel assemblies which
are identical in mechanical design, but different in fuel enrichment.
The core is cooled and moderated by light water at a pressure of 2250
psia in the reactor coolant system. The moderator coolant contains
boron which is varied as required to control relatively slow
reactivity changes including the effects of fuel burnup. The
primary barrier to radiation release is composed of the individual
power producing fuel rods in the core. Two hundred and sixty-four
fuel rods are mechanically joined in a square array to form a fuel
assembly. The fuel rods are supported in intervals along with their
length by grid assemblies whi' maintain the lateral spacing between
the rods throughout the design. fe of the assembly. The fuel rods
consist of uranium dioxide cera pellets contained in slightly cold
worked Zircaloy-4 tubing which i. 'igged and seal welded at the ends
to enapsulate the fuel. The Zircai ,-4 is primarily composed of
zirconium metal. A schematic of the fuel rod is shown in figure A-1.
The fuel pellets are right circular cylinders consisting of slightly
enriched uranium dioxide powder which has been compacted by cold
pressing and then sintered to the required density. The ends of each
pellet are dished slightly to allow greater axial expansion at the
center of the pellets. Power is produced volumetrically by the
fission reaction within individual uranium pellets. Thermal energy
is transferred across the gap to the cladding that serves as the heat
transfer surface to the primary coolant. To avoid overstressing of
the clad or seal welds, void volume and clearances are provided
within the rods to accommodate fission gases released from the fuel,
differential thermal expansion between the clad and the fuel, and
fuel density changes during burnup. At assembly, the pellets are
stacked in the clad to the required fuel height, a spring is then
inserted into the top end of the fuel tube, and the end plugs pressed
into the ends of the tube and welded. All fuel rods are internally
pressurized with helium during the welding process in order to
minimize compressive clad stresses and creep due to coolant operating
pressures. The fuel rods retain all fission gases and radioactive
fission products unless the cladding is breached by mechanisms present
during an accident. One function of the reactor coolant system is
to ensure the cladding remains covered and cooled to prevent clad
melt during normal and accident events.

The reactor coolant system shown in figure A-2 consists of four
similar heat transfer loops connected in parallel to the reactor
pressure vessel. Each loop contains a reactor coolant pump, steam
generator, and associated piping and valves. In addition, the system
includes a pressurizer, a pressurizer relief tank, interconnecting
piping, and instrumentation necessary for operational control. All
the above components are located in the containment building.

FIGURE A-1 — Cutaway diagram of a fuel rod with labels: END PLUG, SPRING, UO₂ PELLETS, FUEL-CLAD GAP, ZIRCALOY CLAD. Dimensions: 151.6" (REF), 144.0" (TYP), 6.3" (TYP).

SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES SUCH AS
PRE-PRESSURIZATION, POWER HISTORY, AND DISCHARGE BURNUP

SEQUOYAH NUCLEAR PLANT
REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP

695' 3"

27-1/2" ID

REACTOR VESSEL

UHI

706'-8"

EL 600'-9"
TOP OF CORE

673'-11"
BOTTOM OF CORE

RELIEF VALVE

NORMAL WATER LEVEL 739'-0"

PRESSURIZER

706'-6"

29" ID

14" ID

NORM WATER LEVEL 717'-9"

720'-4"

FEEDWATER INLET

NORMAL WATER LEVEL
745'-11"

733'-0"
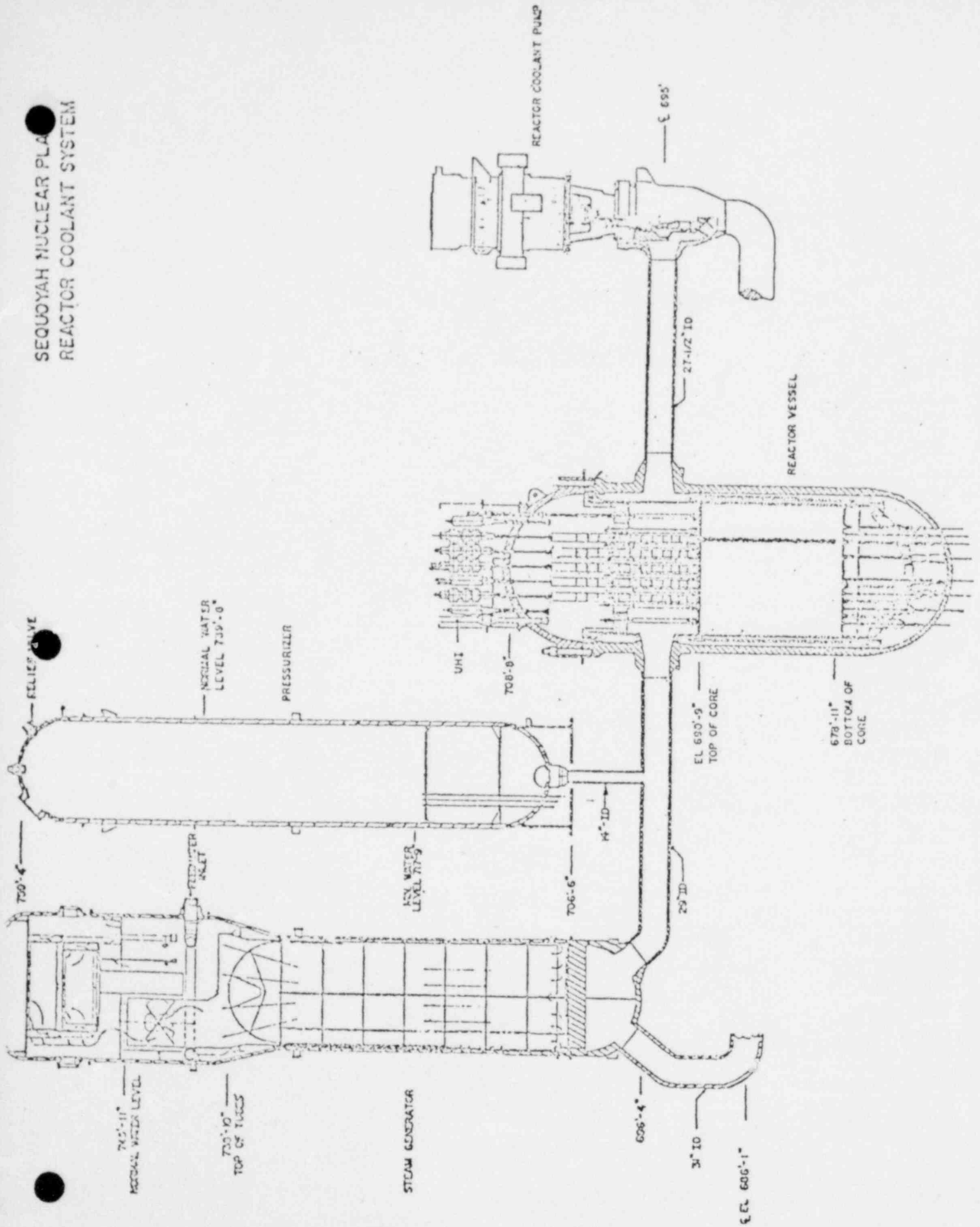TOP OF TUBES

STEAM GENERATOR

606'-4"

31" ID

EL 606'-1"

FIGURE A-2

During operation, the RCS transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator. Borated demineralized light water is circulated in the RCS at a flow rate and temperature consistent with achieving the design reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector and as a solvent for the neutron absorber used in chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant. Its extent is defined as:

1. The reactor vessel including control rod drive mechanism housings.

2. The reactor coolant side of the steam generators.

3. Reactor coolant pumps.

4. A pressurizer attached to one of the reactor coolant loops.

5. Safety and relief valves.

6. The interconnecting piping, valves, and fittings between the principle components listed above.

7. The piping, fittings, and valves leading to connecting auxiliary or support systems up to and including the containment isolation valves.

RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations due to contraction and expansion of the reactor coolant. Three spring-loaded safety valves and two power-operated relief valves are mounted on the pressurizer and discharge to a relief tank, where the steam is condensed and cooled by mixing with water. These valves ensure the integrity of the RCS during overpressure events. For accidents where the pressure boundary integrity is lost, emergency core cooling systems are used to maintain core cooling within the RCS. This entails removal of both stored and fission product decay heat from the reactor core. Systems employed in this function include: cold leg injection accumulators; upper head injection accumulators; residual heat removal pumps and heat exchangers; centrifugal charging pumps; safety injection pumps; and indirectly the auxiliary feedwater system which maintains steam generator secondary inventory.

A.2 Containment

The containment is divided into three compartments: The lower compartment; the upper compartment; and the ice condenser compartment. The lower compartment completely encloses the reactor

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coolant system equipment. The upper compartment contains the refueling canal, refueling equipment, and the polar crane used during refueling and maintenance operations. The upper and lower compartments are separated by the operating deck, which provides a low-leakage barrier between these two compartments. The ice condenser connects the lower compartment to the upper.

The ice condenser concept utilizes a large mass of ice to condense escaping high-energy steam from postulated loss-of-coolant accidents (LOCA) or steam line break accidents. The rapid condensation of steam in the ice bed keeps the maximum containment pressure relatively low while maintaining the capacity to absorb a continuing high energy input from the reactor core and reactor coolant systems. The ice condenser is made up of 24 individual bays which form a 300° arc inside containment. Each bay consists of three major sections: A lower plenum: an ice bed; and an upper plenum (figure A-3). The lower plenum is isolated from the lower compartment by doors in each bay that open at a differential pressure of 0.007 psi. The ice bed contains a minimum of 2.45 million pounds of ice. The ice is stacked in columns one foot in diameter and 48 feet high. The upper plenum contains cooling units used to maintain the low ice bed temperature during normal plant operations. The upper plenum is separated from the ice bed and the upper compartment by two sets of doors that will open with a differential pressure of 0.028 psi.

In the event of a LOCA, steam pressurizes the lower compartment which opens the lower inlet doors. An air-steam mixture enters the ice bed where all the steam is condensed. The rising air then causes the top two sets of doors to open and then flows into the upper compartment. To provide maximum use of the ice bed, air return fans are provided which circulate containment atmosphere through the ice bed condensing steam released after the initial pressurization. When all the ice has melted, the spray system located in the upper compartment removes the remaining energy released to the containment.

A.3 Design Bases of Containment

The ice condenser is designed to limit the containment pressure below the design pressure for all reactor coolant pipe break sizes up to and including a double-ended pipe severance of the largest main reactor coolant pipe. Characterising performance of the ice condenser requires consideration of the rate of addition of mass and energy to the containment as well as the total amounts of mass and energy added. Analyses have shown that the accident which produces the highest blowdown rate into an ice condenser containment will result in the maximum containment pressure rise; that accident currently is the double-ended severance of one of the main reactor coolant pipes. The design basis accident is therefore defined to be the double-ended reactor coolant loop pipe break. Postblowdown energy releases are also accommodated without exceeding containment design pressure.

A.3.1 Mass and Energy Releases
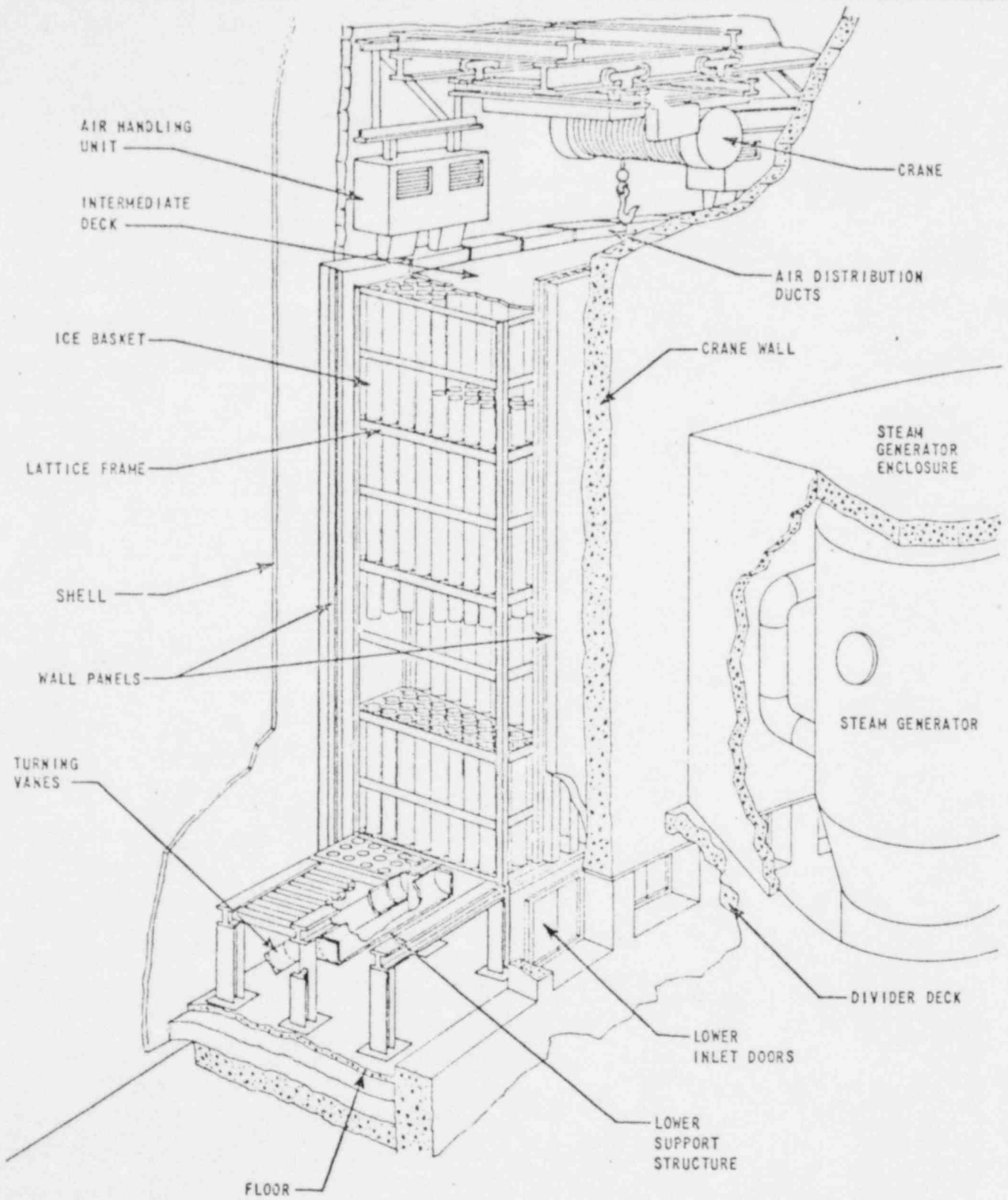
AIR HANDLING UNIT

INTERMEDIATE DECK

ICE BASKET

LATTICE FRAME

SHELL

WALL PANELS

TURNING VANES

FLOOR

CRANE

AIR DISTRIBUTION DUCTS

CRANE WALL

STEAM GENERATOR ENCLOSURE

STEAM GENERATOR

DIVIDER DECK

LOWER INLET DOORS

LOWER SUPPORT STRUCTURE

Following a postulated rupture of the Reactor Coolant System (RCS), steam and water are released into the containment. Initially, the water in the RCS is subcooled at a high pressure. When the break occurs, the water pas͟ ͟ through the break where a portion flashes to steam at the lowe͟ ͟p͟essure of the containment. These releases continue until the RCS depressurizes to the pressure in the containment (end of blowdown). At that time, the vessel is refilled by water from the accumulators and Safety Injection (SI) pumps. The analysis assu͟ ͟ ͟͟at the lower plenum is filled with saturated water at the end of blowdown to maximize steam releases to the containment. Therefore, the water flowing from the accumulators and SI pumps start to fill the downcomer causing a driving head across the vessel which forces water into the hot core.

During the reflood phase of the accident, water enters the core where a portion is converted to steam which entrains an amount of water into the hot legs at a high velocity. Water continues to enter the core and release the stored energy of the fuel and clad as the mixture height in the core increases. When the level, two feet below the top of the core, is reached, the core is assumed to be totally quenched, which leaves only decay heat to generate steam. This type of break is analyzed at three locations.

The location of the break can significantly change the reflood transient. It is for this reason that the hot leg, pump suction, and cold leg break locations are analyzed. For a cold leg break, all of the fluid which leaves the core must vent through a steam generator and becomes superheated. However, relative to breaks at other locations, the core flooding rate (and therefore the rate of fluid leaving the core) is low because all the core vent paths include the resistance of the reactor coolant pump. For a hot leg pipe break, the vent path resistance is relatively low, which results in a high core flooding rate, but the majority of the fluid which exits the core bypasses the steam generators in venting to the containment. The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and steam generator heat addition as in the cold leg break. As a result, the pump suction breaks yield the highest energy flow rates during the postblowdown period. The spectrum of breaks analyzed includes the largest cold and hot leg breaks, reactor inlet and outlet, respectively, and range of pump suction breaks. Because of the phenomena of reflood as discussed above, the pump suction break location is the worst case. This conclusion is supported by studies of smaller hot leg breaks which have been shown, on similar plants, to be less severe than the double ended hot leg. Cold leg breaks, however, are lower both in the blowdown peak and in the reflood pressure rise. Thus, an analysis of smaller pump suction breaks is representative of the spectrum of break sizes.

The LOCA analysis calculational model is typically divided into three phases: (1) blowdown, which includes the period from accident occurrence (when the reactor is at steady state full power operation) to the time when zero break flow is first calculated; (2) refill, which is from the end of blowdown to the time the ECCS fills the vessel lower plenum; and (3) reflood, which begins when water starts

moving into the core and continues until the end of the transient.
For the pump suction break, consideration is given to a possible
fourth phase; that is, froth boiling in the steam generator tubes
after the core has been quenched. (See reference A-1 for a
description of the calculational model used for the mass and energy
release analysis.)

A.3.2   Noncondensible Gas Production

Another source of mass that must be considered in containment design
is the production of noncondensible flammable gases. The more severe
loss-of-coolant accidents postulated for plant design may result in
the production of excess hydrogen which ultimately is released to
the containment air space. Depending on the final volumetric
concentration of the hydrogen and the quantity of oxygen present,
the gas can be a fire hazard, an explosion hazard, or too diffuse
to chemically react.

The corrosion of metals in the presence of water at temperatures
characteristic of light water reactors has been recognized for many
years and many studies of the phenomena have been conducted.
Fortunatley, the oxidation rates at operating temperatures are
extremely low and do not pose a problem for the design life of the
fuel assemblies or the reactor vessel. At elevated temperatures
encountered during mitigated accidents, the corrosion rates become
moderate and some damage to the fuel cladding is expected. Baker
and Just defined an upper bound on the oxidation expected for the
Zircaloy fuel cladding used in light water reactor systems (A-2).
At temperatures below 2200° F, the oxidation does not degrade the
cladding fission product barrier. For this reason and others, all
power reactor designs use this limit as one of the criteria for the
effectiveness of the emergency core cooling systems. For excessively
high temperatures expected at clad melt (4000° F and above) the
reaction rates for Zircaloy and water become large and, in addition,
oxidation of other metals becomes important. Since the vessel is
comprised of stainless and carbon steels, at melting temperatures
these can account for thousands of pounds of hydrogen. Reaction
equations for various metals (in the Arrhenius equation form) are
shown in Table A-1. At Sequoyah, the total zirconium present for
oxidation is approximately 43,204 pounds which, if oxidized,
completely produces about 2000 pounds of hydrogen (figure A-4).
The concentration by volume percent in the Sequoyah containment based
on various clad reaction percentages is shown in figure A-5. The
exothermic oxidation reaction alone accounts for 2805 Btu's per pound
of zirconium oxidized. Only 18 percent reaction can be accommodated
by dilution in the containment before the flammable limit of 4 percent
by volume is reached. However, considerable oxidation must occur
before the detonation limit of 18 percent is reached. For each pound
of zircaloy oxidized, 7.88 cubic feet of hydrogen at STP is released
(32° F, 14.7 psia).

Another source of hydrogen is the corrosion of zinc primers and
aluminum metal inside the containment. These sources along with sump
radiolysis are slow in evolution and are important in design for long

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TABLE A-1

(See references 6-7 and A-4)

METAL-WATER CORROSION RATE PARAMETERS

Metal	A	B
Zirconium*	33.3×10^{6}	-45,500
Stainless Steel**	2.4×10^{12}	-84,300
UO_2***	1.65×10^{5}	-50,800

$W^2 = A\ t\ \exp(B/RT)$

where t = time (sec)

$\quad$ R = gas constant 1.987 cal/mole $^{\circ}$K

$\quad$ T = temperature $^{\circ}$K

*W = weight (mg) zirconium reacted per cm^2 area

**W = oxygen uptake (mg) per cm^2 area

***W = oxygen absorbed (mg) per cm^2 area

Sequoyah Hydrogen Study

HYDROGEN PRODUCTION

ZIRCONIUM OXIDATION

NO RADIOLYSIS

NO SPRAY CORROSION

FIGURE A-4

METAL/WATER REACTION (%)

HYDROGEN (lb)

CONTAINMENT HYDROGEN

Sequoyah Hydrogen Study

ZIRCONIUM OXIDATION

NO RADIOLYSIS

NO SPRAY CORROSION

H₂ CONCENTRATION (VOL %)
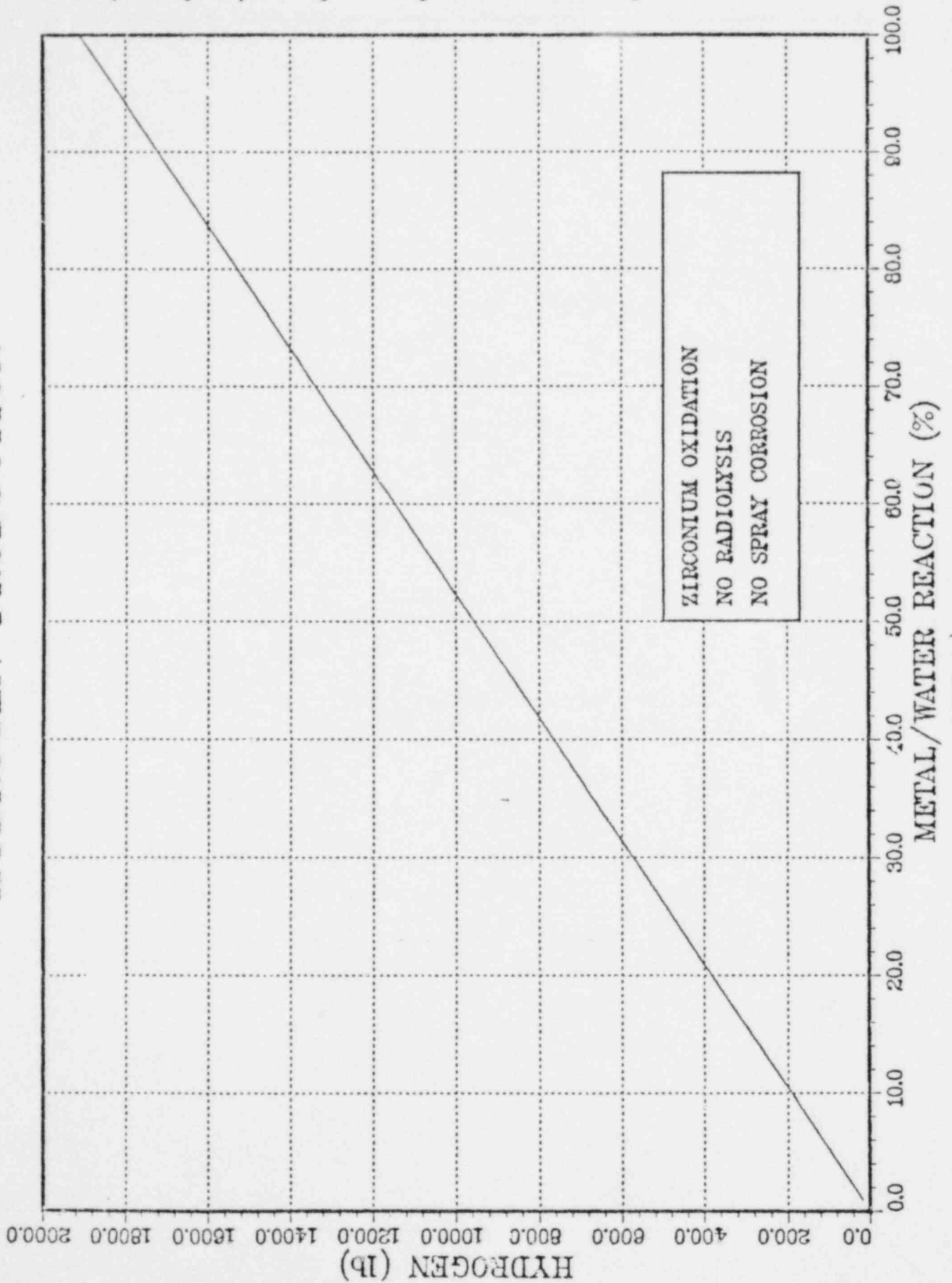
METAL/WATER REACTION (%)
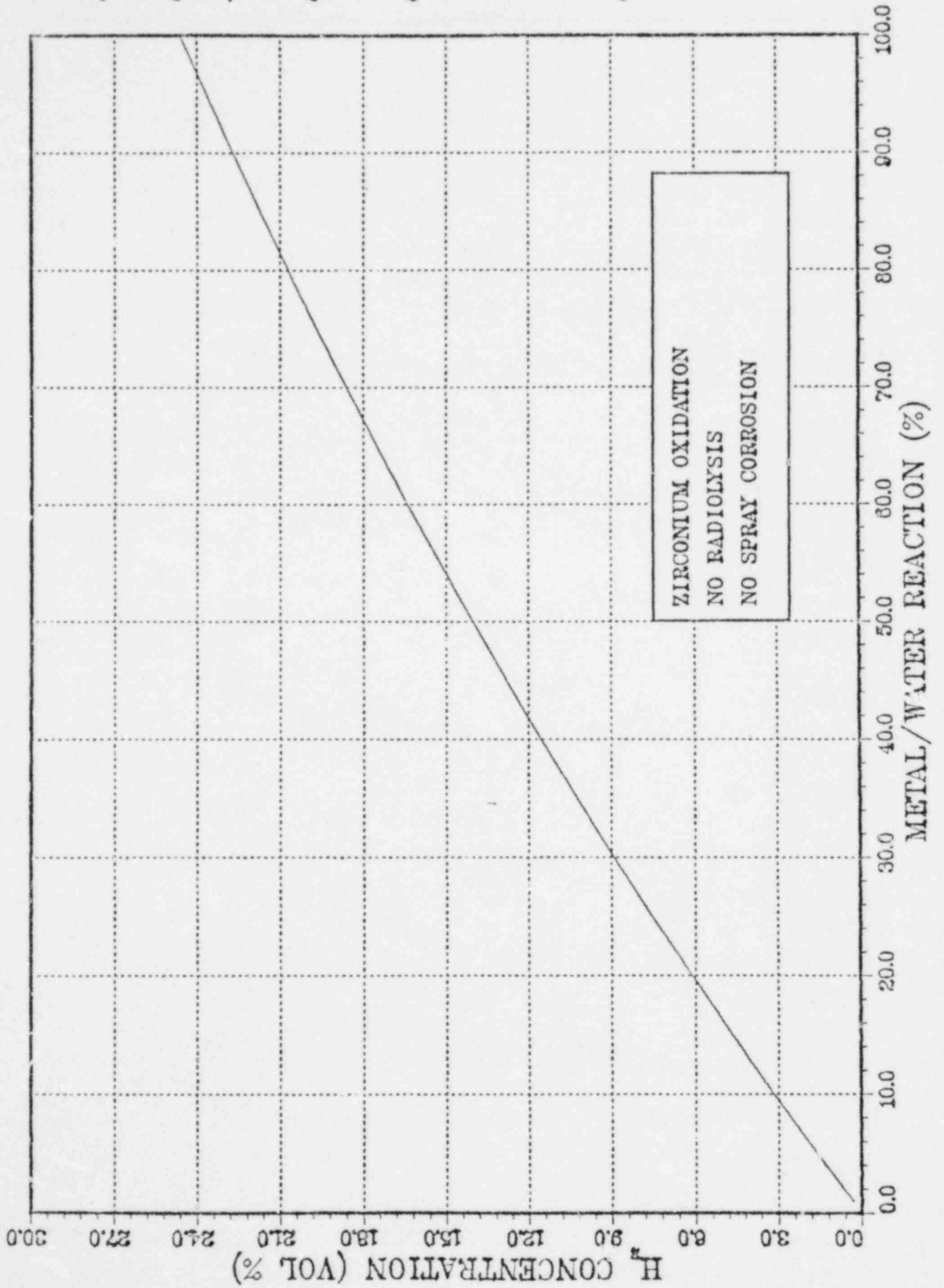
FIGURE A-5

term control. The corrosion reaction is similar to that discussed above; however, it occurs at a much lower temperature characteric of the post-LOCA containment environment. Normally, this reaction would not be rapid enough for considertaion; however, very extensive surface areas render this source important.

The production of noncondensible gas inside a pressurized water reactor (PWR) primary system occurs during normal plant operation. Hydrogen and oxygen gas are simultaneously generated by the dissociation of core coolant via ionizing radiation in a PWR, Fortunately, a radiolysis back reaction operates concurrently which recombines the oxygen and hydrogen present to again form water. Provided excess hydrogen is placed in solution in the primary system coolant, free oxygen generated will not remain because of this recombination reaction, thereby preventing any flammable mi ture formation within the reactor coolant loops. The plant chemical and volume control system ensures excess hydrogen is present in solution. As long as the hydrogen remains in solution and inside the reactor coolant pressure boundary, it represents no hazard to the containment. However, when the primary fluid is discharged to the containment, the dissolved hydrogen gas may come out of solution at the lower pressure of the containment. Solubility of hydrogen in the water at the temperature and pressure characteristic of the reactor coolant system before the accident determines the maximum hydrogen that may be released. In addition, radiolytic decomposition of the water in the core and the sump is assumed to favor the dissociation reaction, producing hydrogen which separates from the coolant. The production rate depends on the quantity of fission products released to the coolant and the core power history. Appropriate pessimistic assumptions are made during analysis to bound the production of gas by this mechanism.

A.4   Computer Programs for Analysis

A.4.1   Current Thermal-Hydaulic Programs for Primary System Analysis

The reactor coolant system response to accident initiating events is studied using advanced thermal hydraulic computer codes. These software tools numerically approximate the continuity, momentum, and energy equations for a series of interconnected control volumes. Those most frequently used are the one dimensional RELAP 4(6-10) , RETRAN (6-2), and in the case of Sequoyah, the Westinghouse WFLASH (4-1), and Westinghouse SATAN (4-2) codes. These programs predict the behavior of fluid systems when subcooled liquid, satuarated vapor, superheated vapor, and when the more physically complex two-phase (liquid and vapor) mixtures exist. The codes approximate the reactor system as a series of fluid control volumes and flow paths.

Homogeneity of the fluid is assumed which means that the liquid and vapor phases in a given volume are intimately mixed and flow in the same direction with the same velocity. Thermal equilibrium is also assumed between fluid phases indicating that the liquid and vapor in a volume are at the same temperature.

Some of these codes include additional models that relax some of the

above assumptions, particularly the assumption that the vapor and
liquid move at the same velocity. This is necessary for the proper
modeling of small break loss-of-coolant accidents where the vapor
produced during depressurization has sufficient time to bubble through
the liquid, slipping past the slower moving fluid. The additional
models only modify the fluid solution which remains basically
homogeneous.

Current research has led to the development of computer programs that
solve a more complete set of equations, describing the liquid and
vapor phases separately with coupling terms between phases to detail;
for example, the drag of the vapor flow on the liquid. These codes
include RELAP5 (A-3) and the advanced multidimensional program TRAC
(6-1). Thermal nonequilibrium is incorporated to accurately describe
injection of cold emergency core cooling water into the primary system
fluid which is at elevated temperature. These codes represent the
state-of-the-art in reactor system mathematical description.

All of the above codes are single component, two-phase approximations
to the fluid behavior. Vapor and liquid phases of water may be
present simultaneously, but other components (e.g., nitrogen,
hydrogen, and other noncondensible gases) cannot be followed using
these programs. This limitation is not a problem since significant
core oxidation followed by a large hydrogen production rate has not
been predicted for the present design basis accident. Because the
computer models do not determine the physics associated with the
presence of hydrogen and noncondensible gases in general, the
distribution of the gas through the primary system cannot be derived
directly from the vendor analyses. For the purpose of containment
analysis, it is assumed that the hydrogen does not collect within
the primary loop, but is released directly into the containment within
two minutes of its production. This is not unreasonable in the design
basis large LOCA due to the rapidity at which mass is released from
the primary system via the break. However, the single component
limitation is a problem for the inadequate core cooling events.

A.4.2   Current Thermal-Hydraulic Program for Containment Analysis

The Sequoyah containment was analyzed using the LOTIC (4-3) computer
code. The Westinghouse LOTIC code (Long-Term Ice Condenser Code)
has the capability to properly describe the postblowdown period in
the ice condenser containment. Not only are the upper, lower, and
ice condenser volumes described, but also the ice condenser is divided
into six circumferential sections, each with two vertical divisions.
Another significant feature of the code is the two sump configuration
(active and stagnant sumps) such that the sump level increase and
temperature history of the containment is accurately modeled. The
code also describes the performance of the air recirculation fan in
returning upper compartment air to the lower compartment. Coupling
of residual and component cooling heat exchangers is provided to give
an accurate indication of performance for these heat exchangers.
The spray heat exchanger performance is also accurately modeled in
the transients. The basic equations used are the standard transient
mass and energy balances and the equations of state used in any
containment transient, but appropriately coupled to the multi-volume

ice condenser containment. The core also considers noncondensible accumulator gas added to the containment and the displacement of free volume by the refueling water storage tank volume.

The LOTIC code, as do the primary system codes, uses the control volume technique to represent the physical geometry of the system. Fundamental mass and energy equations are applied to the appropriate control volumes and solved by suitable numerical procedures. The initial conditions of the containment by compartment is specified before blowdown. Ice melt is calculated for the blowdown period based on the mass and energy released to the containment. After the RCS blowdown, the basic LOTIC code assumption is made that the total pressure in all compartments is uniform. This assumption is justified by the fact that after the initial blowodown of the RCS, the remaining mass and energy released from this system into the containment are small and very slowly changing. The resulting flow rates are unable to maintain significant pressure differentials between the containment compartments.

Tests have been performed by Westinghouse at their Waltz Mill facility to demonstrate the long term performance capability of the ice condenser. Specifically, these tests verified the ability of the ice condenser to reduce the containment pressure within a few minutes following the blowdown, and have provided ice condenser performance parameters for tests simulating the long term addition of residual heat.

The thermodynamic conditions in these compartments are obtained as a function of time by the use of fundamental equations solved through numerical techniques. These equations are solved for three distinct phases of problem time. Each phase corresponds to a distinct physical characteristic of the problem. Each of these phases has a unique set of simplifying assumptions based on test results from the ice condenser test facility. These phases are the blowdown period, the depressurization period, and the long term period.

In the control volumes, which are always assumed to be saturated, steam and air are assumed to be uniformly mixed and at the control volume temperature. When the circulation fan is in operation, the fan flow and the reactor coolant system boiloff are mixed before entering the lower compartment. The air is considered a perfect gas, and the thermodynamic properties of steam are taken from the ASME steam table.

The condensation of steam is assumed to take place in a condensing node located, for the purpose of calculation, between the two control volumes in the ice storage compartment. The exit temperature of the air leaving this node is set equal to a specified value which is equal to the temperature of the ice filled control volume of the ice storage compartment.

A-14

A.5.1 Hydrogen Analyzers

Hydrogen detection and monitoring postaccident is essential to
determine the proper time for mitigation of accumulated hydrogen.
This function is accomplished by the hydrogen analyzer system. The
analyzer system is normally in a standby mode and calibrated, ready
for immediate operation. In addition, sampling may be performed by
removing gas from the containment and analyzing it in the chemistry
lab as a backup.

The analyzers are located in the annulus and are constructed of
materials compatible with the environmental conditions expected
postaccident in the annulus area. Controls are located in the main
control room and on a remote panel located in the auxiliary building
in an area accessible postaccident. The system is designed to sample
the containment atmosphere for the presence and concentration of
hydrogen and is completely redundant (2 per unit). It is also
designed to operate continuously during and following an accident
over the containment conditions of 2 to 50 psig and 40 to
290°F. Concentration indicators and system alarms for both analyzer
systems are located in the main control room.

The accurate measurement of hydrogen in the presence of nitrogen,
oxygen, and water vapor is possible because the thermal conductivity
of hydrogen is approximately seven times greater than nitrogen,
oxygen, or water vapor, which have nearly identical thermal
conductivities at the filament operation temperature of approximately
500°F. Hydrogen measurement is accomplished by using a thermal
conductivity measurement cell and a catalytic reactor. The sample
first flows through the reference section of the cell, then passes
through the catalytic converter where free oxygen is catalytically
recombined with hydrogen to form water vapor, and finally, flows
through the sample section of the measuring cell. The hydrogen
content is indicated by the difference in thermal conductivity between
the sample and reference sides of the cell.

The system currently analyzes H_2 concentrations over the range of
0 to 10 percent, with 1 percent of full scale accuracy; however,
through inquiries with the manufacturer, it has been determined that
the analyzer could be altered to read over the scale 0 to 25 percent
wth only minor modifications.

A.5.2 Electric Hydrogen Recombiners

The accumulation of hydrogen in the primary containment building is
controlled via the hydrogen recombiners. The recombine.'s are manually
initiated by the operator based on information from the hydrogen
detection system. After an initial warmup time of five hours, the
recombiners will begin depleting the hydrogen at a relatively slow
rate. At the maximum design hydrogen concentration of 4 percent by
volume, the recombiners will process 1.35 pounds hydrogen per hour
for each operating recombiner (two available). This recombination
rate is approximately equal to the hydrogen production from sump and
core water radiolysis in the first few hours of the accident and is
sufficient to equal or exceed sump radiolysis, aluminum corrosion,
and zinc corrosion at times later in the accident. The short-term

hydrogen generation by the zirconium-water reaction in the core is accommodated by dilution into the containment volume.

Recombination of hydrogen and oxygen is accomplished by electric heating in the permanently installed recombiner units. The recombiners are sized to process a sufficient amount of containment atmosphere such that the containment hydrogen concentration remains below flammability limits after the release of hydrogen from a 5-percent metal-water reaction assuming only one recombiner is functioning. As a result of these design parameters, each primary containment building is supplied with two such recombiners, each with a minimum capacity to process 100 SCFM of containment atmosphere even in the severe post-LOCA containment environment.

Containment air is drawn into the unit by natural convection, passing first through the preheater section. This section consists of the annular space between the heater-recombination section duct and the external housing. The temperature of the incoming air is increased by heat losses from the heater section. This reduces external heat loss and results in increased efficiency of the unit. The preheated air passes through an orifice plant and enters the heater-recombination section which consists of a thermally insulated vertical metal duct enclosing five assemblies of metal-sheathed electrical heaters. Each heater assembly contains individual heating elements, and the operation of the unit is virtually unaffected by the failure of a few individual heating elements. The incoming air is heated to a temperature in the range of 1150 to 1400°F, where recombination of hydrogen and oxygen occurs. The air from the heater-recombination section finally enters an exhaust section where it is mixed with cooler containment air before being discharged from the unit.

Tests have verified that the recombination of hydrogen and oxygen in the unit is not the result of a catalytic surface effect but occurs as a result of the increased temperature of the process gases. The performance of the unit is, therefore, unaffected by fission products or other impurities which might poison a catalyst. However, should the hydrogen concentration exceed 6 percent by volume, it may be necessary to terminate recombiner operation to preclude initiation of a hydrogen deflagration by the recombiners.

A.5.3   The Hydrogen Purge System

The hydrogen purge system is a backup to the hydrogen recombiners should they be unavailable. The system consists of two subsystems; an exhaust subsystem and a dilutent air supply.

The hydrogen purge exhaust subsystem consists of a single penetration in the primary containment wall equipped with two normally closed, remote manually operated isolation valves, one on either side of the containment wall; one pneumatically operated annulus purge exhaust valve located within the annulus; and two 1/2-inch leakoff nipples located between the outboard isolation valve and the annulus purge exhaust valve. With the containment isolation valves open and the annulus purge exhaust valve closed, a flow path is established from the primary containment through the leakoffs and into the annulus,

which will permit purging of the containment for hydrogen control
subsequent to a LOCA. The impetus for flow is provided by the
differential pressure between the primary containment and annulus.
If the concentration cannot be maintained below 4 percent through
the leakoff path, the annulus purge valve will be opened to supply
diluted air for a minimum time sufficient to maintain the hydrogen
concentration below the flammable limit. The containment effluent,
purged for hydrogen control, will flow directly to the annulus where
it will mix with the annulus atmosphere and be filtered by the air
cleanup system prior to discharge to the outside environment.

Dilution air flow is introduced into the containment from the service
air system. The service air system has provisions enabling it to
receive diesel power. The dilution supply subsystem consists of a
single 2-inch penetration in the primary containment wall equipped
with provisions for containment isolation. The inboard containment
isolation feature is a check valve located in the primary
containment. The outboard containment isolation feature is a double
O-ring sealed flange located in the auxiliary building. A pressure
hose will be required to provide a flow path from a service air
flange. Operation of this subsystem is accomplished by removing the
flange and coupling a service air hose to the pipe penetration. The
system is sized to provide 60 SCFM of dilution air at a service air
pressure of 60 psig.

A.5.4   Containment Mixing System

Mixing of the containment atmosphere to minimize localized
concentration of hydrogen is accomplished by the containment air
return fan system. The associated ductwork consists of: two 12-inch
ducts (one connected to each air return fan housing) which draw air
in from the containment dome region; one 8-inch duct which circles
the containment removing air from accumulator rooms and other dead-
ended spaces and terminates at each air return fan housing; one 8-inch
duct which circles the crane wall, removing air from steam generator
and pressurizer compartments and terminates at each air return fan
housing; two 8-inch pipes (one connected to each air return fan
housing) which remove air from above the spent fuel pool; and a main
duct between the upper and lower compartment through the divider deck
that includes the nonreturn dampers.

The ductwork described above is embedded in concrete, where possible,
to prevent damage from buildup of pressure during a LOCA. Ductwork
not protected by embedment is designed to withstand the LOCA
environment. Rapid pressure buildup in ductwork in the upper
containment compartment is precluded by nonreturn dampers which
prevent the high pressure LOCA effluent flowing from lower to upper
compartment.

APPENDIX B
RISK ANALYSIS BACKGROUND AND
IDENTIFICATION OF ACCIDENT SEQUENCES

The safety of nuclear power plants can be addressed in terms of risk. Risk of a specific event is defined as the product of (1) the probability of occurrence for that event and (2) the consequences of that event. Both entities must be considered simultaneously to obtain the overall plant safety.

The first extensive evaluation of nuclear power plant risk was released in 1975 by a group headed by Dr. Norman Rasmussen of MIT. Known as the Rasmussen Report or the Reactor Safety Study, WASH-1400 was the first concentrated effort to provide an analytical assessment of the safety of a nuclear power plant using probabilistic methodologies. The WASH-1400 study addressed plant, system, and component failures, radiological assessments of accidents, and comparisons of risks associated with other common risks (such as airline accidents, natural phenomenon, etc.). The basic concept used in the WASH-1400 study was to estimate the risk of a nuclear power plant through an extensive evaluation of key and dominant sequences. Key sequences are plant failures which are representative of many other combinations of system and containment failure states. Dominant sequences are those sequences which are considered to have the highest probability of occurrence.

The WASH-1400 study provided an estimate of plant risks based on determining the probability for a particular sequence of system failures given an initiating event and then determining the consequences of such a failure sequence. The initiating events chosen for WASH-1400 consisted of large and small loss-of-coolant accidents, normal operating transients, and some special case failures. The study selected the Surry Nuclear Plant as the representative pressurized water reactor design. The Surry plant uses a Westinghouse nuclear steam supply system enclosed in a dry containment.

For each initiating event a decision was made as to what safety systems would be required to mitigate the consequences of the initiating event. An event tree was then developed for each initiating event in order to determine all the possible system failure sequences which could occur (see figure B-1 for an example). Many potential sequences in the event tree were eliminated from detailed consideration because the failure of certain systems would make immaterial the success of other systems needed later in the sequence.

Each of the safety systems identified for use in mitigating the initiating events were then qualitatively and quantitatively analyzed in order to obtain an estimate of the probability that the system would fail to accomplish its intended function. These system failures will, in general, be independent (dependence being accounted for in the construction of the event tree).

Having found the probability of a particular sequence, the appropriate containment failure mode must be determined for each sequence. When
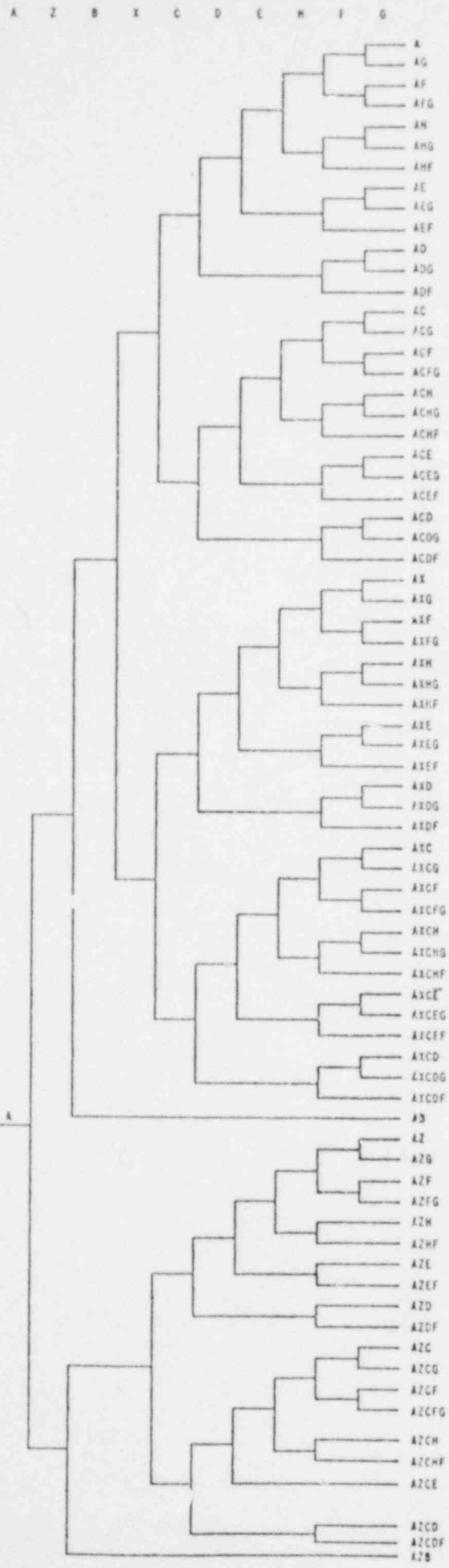
S24 EVENT TREE - LARGE LOCA (D > 6")

FIGURE B-1

a particular containment failure mode was selected for a sequence (a sequence may have more than one possible failure mode), the probability of that containment failure mode was estimated based on quantitative and qualitative analysis.

The next step was to estimate the magnitude of the consequences which could result from nuclear plant accidents. WASH-1400 did this by defining nine release categories for a PWR which distributed by magnitude the fission product releases of the accidents and by assigning each key sequence to one of the nine categories. Category one was defined to have the most severe fission product releases and category nine the least severe. Categories one through seven were core melt sequences.

The selection of a release category for a particular Sequoyah accident sequence was made using WASH-1400 as a guide. However, the uncertainties in the analysis techniques, computer codes, and data require that the possibility be considered that a sequence has been placed in an incorrect category. The WASH-1400 study accounted for this possibility through the use of sequence smoothing. This method was also used in the Sequoyah analysis. Smoothing is a method for incorporation of possible release variations, and it assumes that the release category chosen for a particular sequence was the correct choice with a 78 percent probability. The method further assumes if the sequence was placed in the incorrect category, it most likely was a borderline case and the correct category would be the category immediately before and immediately after the one originally chosen with a 10 percent probability. A 1 percent probability exists that the sequence actually belongs in a release category two before or two after the one chosen.

All relevant sequences were then compared to the key sequences in order to determine in which category they were to be placed. The probability of each sequence was calculated by multiplying the initiating event probability by the system failure probabilities by the containment failure mode probability. By summing the probabilities in each release category, one can obtain an estimate of risk of the plant; that is, the probability of the various sequences (occurrence) in that release category (consequence). WASH-1400 carried the analysis all the way to the ultimate consequence by analyzing the effects of the release categories on a representative population around the nuclear plant and estimating the human and environmental impacts due to the fission product releases.

The objective of the Sequoyah analysis was to determine a reasonable estimate for the risk associated with reactor core damage and core melt accidents at Sequoyah. As in WASH-1400, noncore damage sequences were considered as inconsequential to the risk of the plant and therefore ignored. The risk analysis for Sequoyah was carried to the point of estimating the total probability of accidents in each release category.

A format similar to the WASH-1400 (Surry) study was used to establish the key and dominant sequences for the Sequoyah Nuclear Plant. Similar initiating events (large and small LOCA, etc.) were used for

Sequoyah as were used for the WASH-1400 study. However, the transients evaluated for Sequoyah were the loss of normal feedwater and the loss of ac power. The Sequoyah event trees showing the possible failure sequences for each of the initiating events are shown in figures B-1 through B-5. Table 5-4 provides an explanation of the symbols and provides failure probabilities for the containment failure modes and those systems affected by the study. The assumption was made for Sequoyah that if the core melted, the containment would fail either by steam explosion(α), hydrogen burn (δ), containment leakage (β), or containment overpressurization (ζ). Basemat penetration (ϵ) was not considered. These assumptions resulted in reducing the release categories to five for this study. Sequoyah dominant sequences and their probabilities for each release category are shown in table 5-2.

The Sequoyah analysis was based on a limited number of detailed evaluations of accident sequences and other qualitative evaluations. Three detailed evaluations were obtained from the Battelle, Columbus Laboratories and these were:

1. A large LOCA followed by failure of the emergency core injection system (AD).

2. A small LOCA less than 2 inches followed by failure of the emergency core injection system (S_2D).

3. A transient which was loss of all alternating current power followed by loss of the auxiliary feedwater system (TMLB').

TVA used these three sequences to perform a detailed evaluation of a fourth sequence: a small LOCA followed by the failure of the emergency core recirculation system (S_2H).
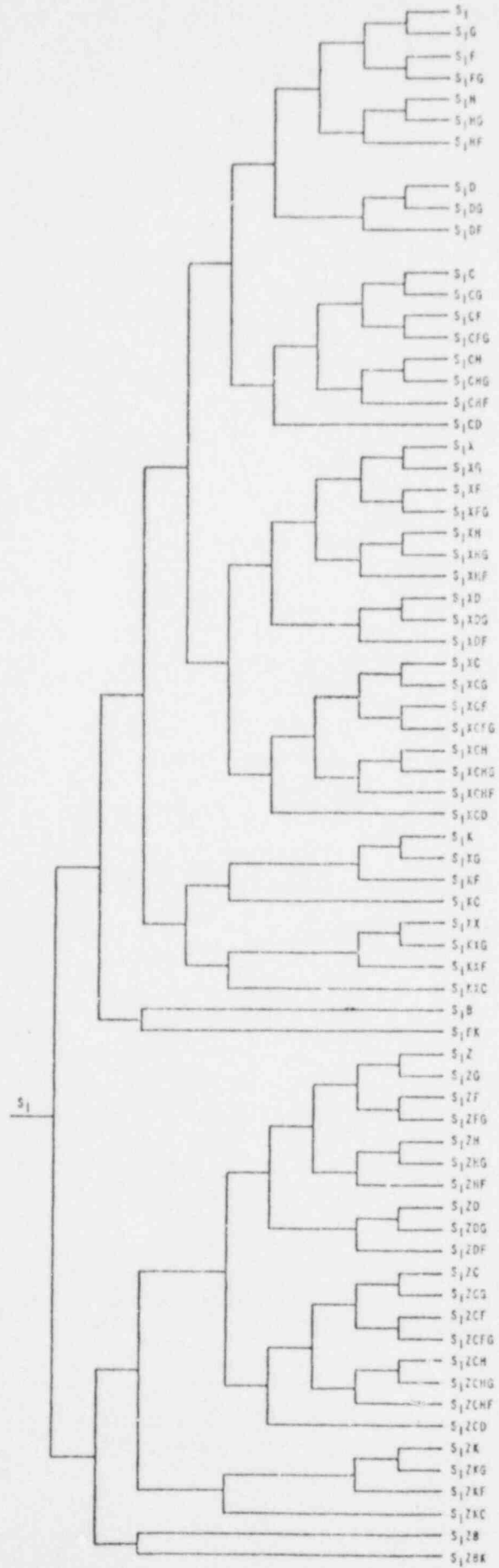
FIGURE B-2   B-5

FIGURE B-3

SQN EVENT TREE - TRANSIENT
(LOSS OF AC POWER)

T_A K B_2 L M P Q

T_A
T_AQ
T_AM
T_AMQ
T_AMP
T_AL
T_ALQ
T_ALP
T_AB_2
T_AB_2Q
T_AB_2P
T_AK
T_AKQ
T_AKP
T_AKB_2
T_AKB_2Q
T_AKB_2P
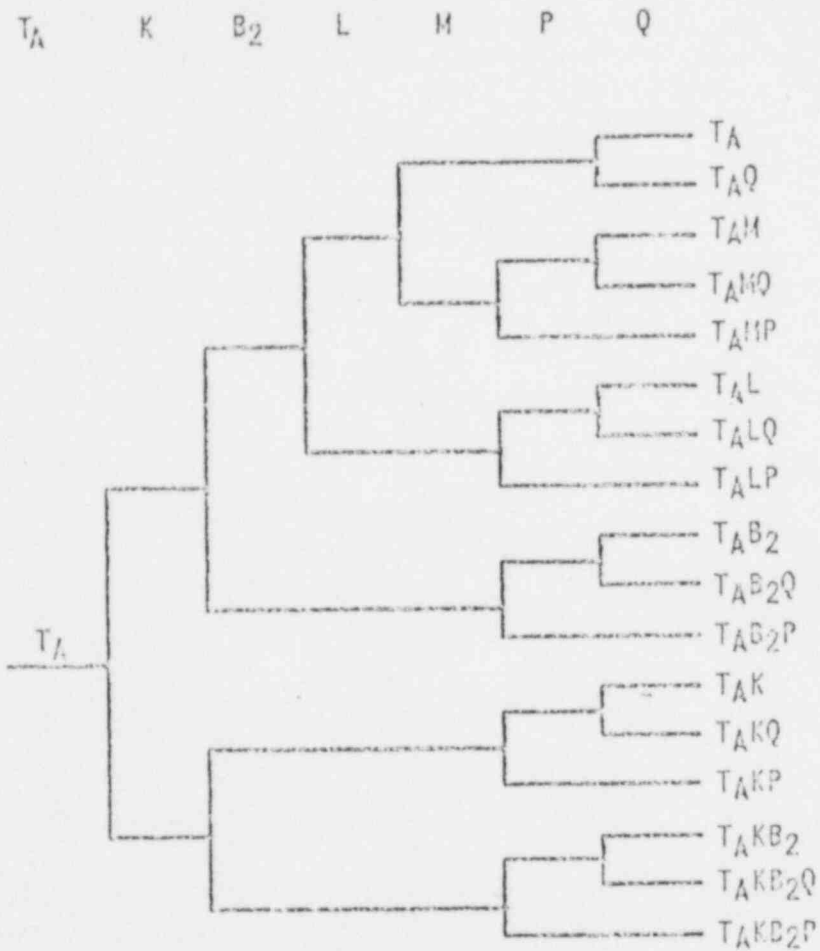
T_A

FIGURE B-4

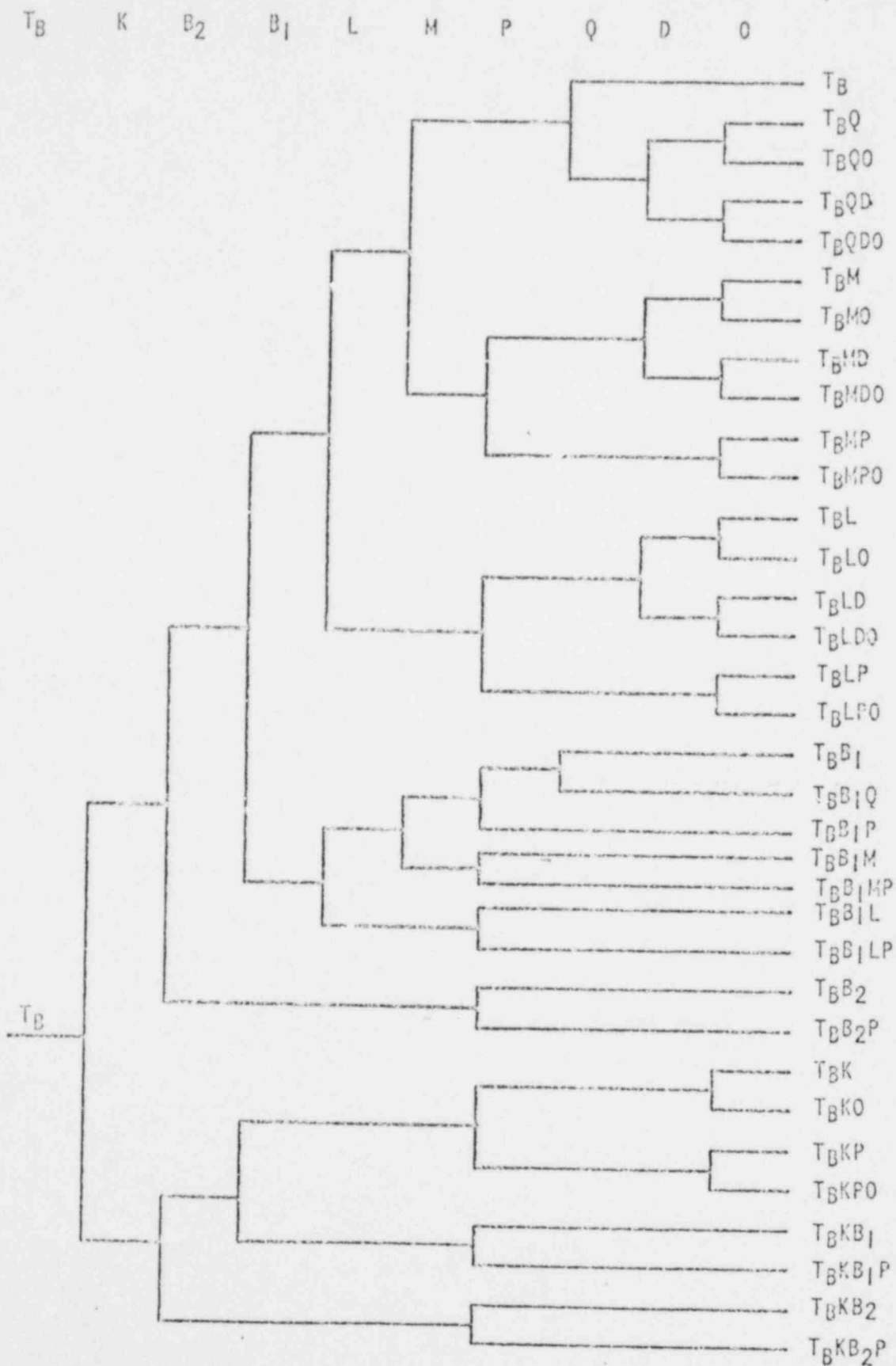SQN EVENT TREE - TRANSIENT
(LOSS OF MAIN FEEDWATER)

FIGURE B-5

APPENDIX C
MARCH COMPUTER CODE

C.1 General Description

The MARCH modules necessary to cover the accident physics are shown in figure C-1 and described below.

Subroutine BOIL is a modification of the BOIL program which was written for the reactor safety study (WASH-1400). The subroutine calculates primary system mass, energy balances, and primary system heat transfer. In combination with subroutine PRIMP, which calculates primary system pressure variation and leakage rates, the subroutine solves the necessary equations of energy and continuity to simulate a one volume RELAP type model for the core. A level swell or bubble size model is included in the core heat transfer nodes. Core fuel elements transfer their heat to the water and thereby, as an option, to a steam generator. In areas where only steam is present, convection to the steam, as well as radiation to the upper internal structures and to the water below, is modeled. Hydrogen is generated by oxidation of the hot cladding with steam, consuming the steam in the reaction. MARCH appropriately limits this reaction based on the availability of steam. Core slumping is calculated once the melting point of the fuel is reached. Cladding metal and relocation is not treated separately from core melt. A melting temperature of $4130^{\circ}F$ is used. Three options are included for the slumping behavior, however, due to large uncertainties it is not known whether some of these are truly mechanistic. Other models which are simplified include the critical flow discharge model that permits only single phase steam or water to be discharged but not to a two-phase mixture.

Subroutine MACE determines the containment response to energy and mass released from the primary system. Ice beds, containment fans, sprays, and air cooler unit models are included in this subroutine. Concrete decomposition and hydrogen burning effects are considered in the containment model. MACE functions as a simplified BEACON or LOTIC code.

Additional coding is used to determine meltdown specific phenomena. Subroutine HEAD determines the time of failure of the vessel due to meltthrough once fuel assembly failure and slumping has occurred. Following vessel melt, molten debris is expected to drop into the water remaining in the reactor cavity and to boil any fluid present. Subroutine HOTDROP determines the boilaway rate. Finally, subroutine INTER calculates the decomposition and penetration of the containment basemat by the molten core debris.

Many of the physical problems modeled by MARCH have been found to be phenomenologically complex during reactor accident research. The

FIGURE C-1
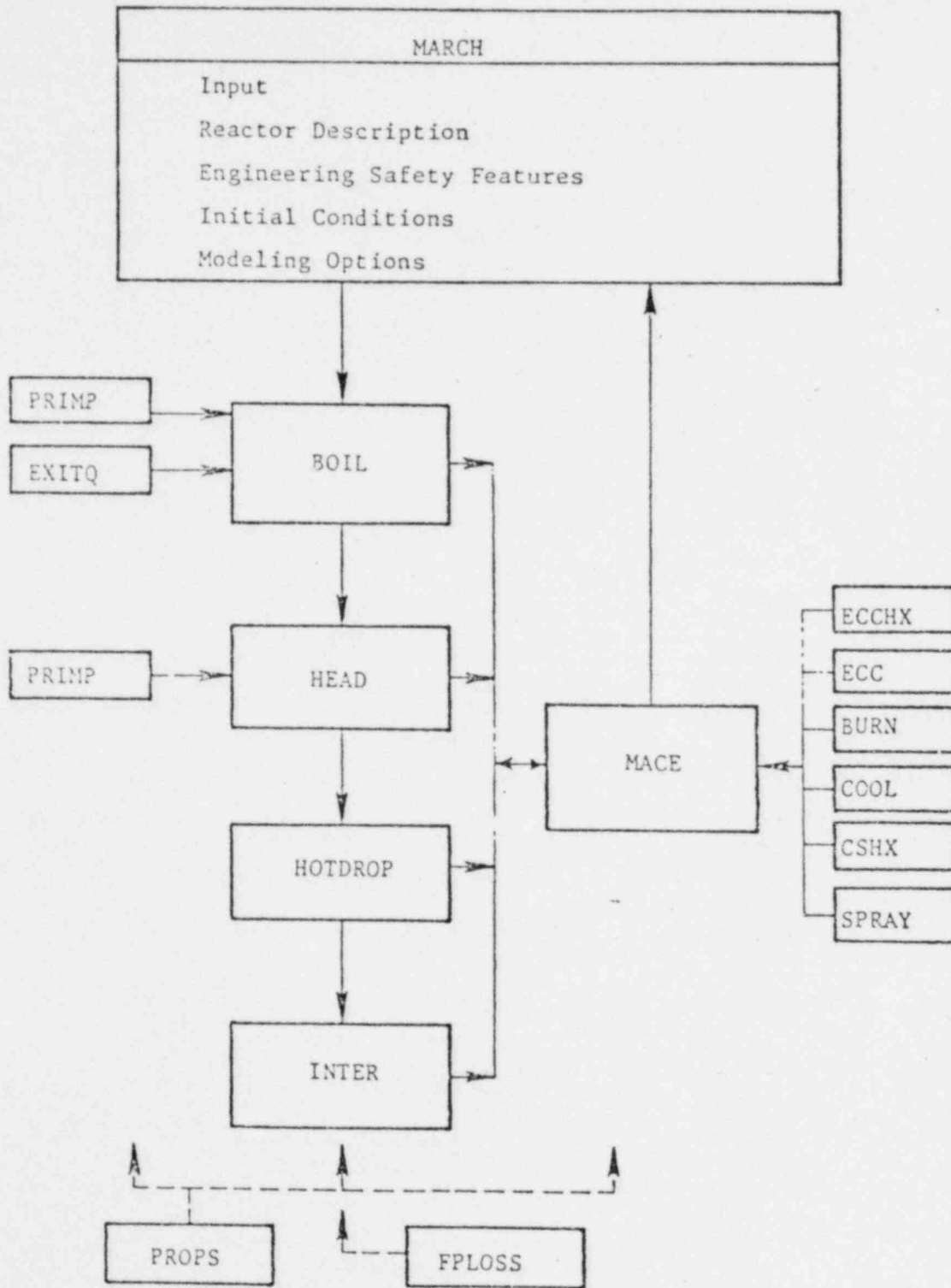
(Ref. 6-3)

melting of the core to a pool of steel, uranium, cladding, control rod material in the possible presence of some remaining water is a complex heat transfer problem. If the fuel is able to heat to extremely high temperatures in the range of 5000 to 5500°F (note that some literature suggests the fuel may not melt until this temperature is achieved) the stainless steel and other metals may begin to boil forming a frothing metallic pool.

Boiling of the steel would raise the pool level and increase heat transfer from the sides of the molten debris. This could delay vessel melt-through, possible radiating more heat from the outer vessel walls to the reactor cavity liquid, boiling a larger quantity of this liquid, and therefore reducing the probability of steam explosion following vessel failure. It is also unlikely that symmetrical melting of the core, vessel, and other components will occur. A more likely scenario would be that portions of the core may melt, dripping into the remaining vessel water causing a slower boiling of this water than predicted by MARCH. It should be noted, in summary, that uncertainties exist in the calculational models employed in this code, that the models may be too conservative in some areas, that some phenomena may not be treated properly, but that the overall code is a significant extension to the current analytical tools and that no other codes are currently available to study the total core melt sequence.

C.2 MARCH Code Check

A simplified inhouse computer code named H2GEN was created to examine adiabatic heatup of the Sequoyah core fuel assemblies following a major accident with inadequate core cooling. Adiabatic heating of the fuel, once uncovered, results in the most conservative rate of temperature increase. Fuel rod oxidation rates and consequently hydrogen rates are maximized, allowing an upper bound on total hydrogen production to be estimated. Comparison of H2GEN results with MARCH output can be used to determine the conservatism in the more advanced code.

H2GEN considers positional variation in the average power of core fuel assemblies. Higher average power assemblies generate more heat following uncovery and achieve larger hydrogen production rates earlier in the transient. Axial variation in the individual rod power has been neglected in H2GEN. Core boildown is not treated explicitly, but is entered as user input from emergency core cooling system studies. The fuel assemblies are modeled as axial segments with twelve nodes per rod. Heat transfer to the core mixture adequately cools a node until it is uncovered, at which time heat transfer from the node is terminated and heatup begins. The program tracks the oxide interface as the cladding reacts with the steam present. Once all the cladding in a rod has corroded, no more hydrogen is generated by that rod. Several models are included as a user option to approximate the oxidation rate including the more conservative Baker-Just correlation. Unlimited quantities of steam are assumed available for cladding oxidation unlike the physical event where oxidation of the lower rations of cladding may consume all steam available, or water boilaway rates, influenced by radiant heat transfer, may not

be sufficient to oxidize cladding at the rates predicted. Rod heatup
is based on volumetric heat generation at ANS decay heat power levels
with no excess conservatism included. Thermal energy is also supplied
by the oxidation reaction in each node where oxidation is in
progress. This contribution can be very significant at large
oxidation rates.

Comparison of H2GEN results with MARCH output for small break
inadequate core cooling events demonstrates that for the portion of
the transient immediately following core uncovery, the MARCH results
are similar but slightly below the strictly adiabatic H2GEN results.
Deviation is noted in later periods as the adiabatic generation
accelerates in the presence of unlimited steam, whereas the MARCH
results exhibit steam limiting. The H2GEN hydrogen production
terminates after all the zircaloy cladding is oxidized, while MARCH
hydrogen generation exceeds the maximum cladding reaction for some
events. However, this is not unanticipated, since MARCH also
estimates the quantity of hydrogen produced by the steel corrosion
reaction. MARCH appears to be adequately conservative over the
inadequate cooling transient based on these limited studies.

APPENDIX D

DETAILED DESIGN CRITERIA

D.1 Filtered Vented Containment

We propose two design cases for evaluation:

I   Core melt event (large LOCA, no hydrogen burn)
II Degraded core event (small LOCA, partial hydrogen burn)

CASE I

1. Maximum vent volume flow rate - 400,000 cfm at $600^{o}F$, 34 psia. This lasts approximately 600 seconds, then drops.

2. Vent Temperature - $600^{o}F$ - Design Temperature - $750^{o}F$

3. Vent Driving Pressure - 35 psia - Design Pressure - 55 psia

4. Constituents - Air   - Swept Out   (88,000 lb)
   after peak
   presssure H_2 - 8%   3,400 lb
   has swept
   most air   Steam   - 47%   450,000 lb
   out

                  CO   - 23%   35,000 lb[1]

                  CO_2 - 22%   128,000 lb[1]

   Average Density 0.04 lb/ft^3

5. Radioactivity - see Table D-1

6. Duration - 12 hours

   Total Heat Input - 600×10^6 Btu

7. Operator will cycle valves - no auto on pressure.

8. Vent size - approximately 7.5 ft^2 - Vent flow rate is to be controlling factor.

CASE II

1. Maximum volume flow rate - 467,000 cfm at 700° F and 55 psia. This lasts approximately 100 seconds. The flow rates are much less at other times, eg 20,000 cfm for previous vent periods.

2. Vent temperature - 700° F Design Temperature - 750° F

TABLE D-1

CASE I

FISSION PRODUCT RELEASES TO CONTAINMENT

ATMOSPHERE DURING CORE MELT ACCIDENT

Number	Name	Source (Curies x 10^8)
1	KR-85	.006
2	KR-85*	.26
3	KR-87	.52
4	KR-88	.76
5	SR-89	.12
6	SR-90	.0057
7	SR-91	.14
8	Y-90	.00068
9	Y-91	.018
10	ZR-95	.021
11	ZR-97	.021
12	NB-95	.021
13	MO-99	.13
14	TC-99*	.11
15	RU-103	.08
16	RU-105	.046
17	RU-106	.015
18	RH-105	.046
19	TE-129	.28
20	TE-129*	.10
21	TE-131	.15
22	TE-132	1.2
23	I-131	.85
24	MI-131	.85
25	I-132	1.2
26	I-133	1.7
27	I-134	2.0
28	I-135	1.5
29	MI-135	1.5
30	XE-133	1.7
31	XE-135	.26
32	CS-134	.017
33	CS-136	.06
34	CS-137	.058
35	BA-140	.18
36	LA-140	.021
37	CE-141	.021
38	CE-143	.020
39	CE-144	.014
40	PR-143	.020
41	ND-147	.0078
42	PM-147	.0022
43	PM-149	.0052
44	PU-238	.000013
45	PU-239	.0000013

1 of 1

3. Vent Driving Pressure - 55 psia (design pressure). Initiate venting at 35 psia.

4. Constituents - several hydrogen burns from 5 percent to 4 percent until last big burst:

Constituent	After First Burn	After Last Burn
N_2	72%	55%
O_2	15%	6%
H_2	5%	4%
Steam	8%	35%

Minimum vent open or driving pressure 35 psia.

5. Radioactivity - see Table D-2.

6. Duration - 10 to 12 burns (ventings) over 2-3 hours. Total heat input - 200×10^6 Btu over 2-3 hours.

7. Operator will cycle valves - no auto on pressure.

8. Vent size - approximately 10 ft^2 - Vent flow rate is to be controlling factor.

9. Vent line length - 500 ft total run - assumed 15 ft below grade for yard pipe run.

10. Decontamination Factor - 100 for particulates
                             100 for iodine
                               1 for noble gases

11. Condensate - should be collected as part of conceptual design of filtered vent system.

12. Seismic - to withstand OBE (0.09g).

13. Tornado - no pressure or missile requirements.

14. Redundant - no redundancy requirements (except isolation valves).

15. Qualification - Mechanical Quality Group "C"

16. Passive versus active - design should be passive as far as possible. Active components can be used when necessary or required to minimize sizes of components.

17. Cooling Water - 5,000 gpm of 100° F cooling water available if necessary.

CASE II

FISSION PRODUCT RELEASES TO CONTAINMENT DURING

DEGRADED CORE ACCIDENT

(CURIES)

	NUCLIDE	PRIMARY CONTAINMENT
1	I-131	7.685E 06
2	I-132	9.222E 06
3	I-133	1.680E 07
4	I-134	1.077E 07
5	I-135	1.443E 07
6	I-131*	8.774E 05
7	I-132	1.053E 06
8	I-133*	1.918E 06
9	I-134*	1.230E 06
10	I-135*	1.647E 06
11	KR- 85M	3.481F 07
12	KR- 85	9.989E 05
13	KR- 87	4.898E 07
14	KR- 88	8.858E 07
15	KR- 89	4.045E 03
16	XE-131M	9.211E 05
17	XE-133M	5.107E 06
18	XE-133	2.021E 08
19	XE-135M	6.570E 06
20	XE-135	5.223E 07
21	XE-137	2.997E 04
22	XE-138	1.719E 07

1 of 1

18. Isolation – three remote manual valves, one inside, one in annulus, and one outside secondary containment – followed by a 35 psia rupture disk outside secondary containment.

19. Vent stack – 400 ft. height

D.2 Additional Containment Volume

1. Design temperature – 750° F

2. Design pressure – 55 psia

3. Vent size – 48 in.

4. Additional containment volume – 1.5×10^6 ft.^{3}

5. Additional containment design pressure – 20 psig

6. Vent line – below yard piping (approximately 15 ft below grade)

7. Seismic – to withstand OBE

8. Redundant – no redundancy requirements (except isolation valves)

9. Isolation – three remote manual valves, one inside, one in annulus, and one outside secondary containment – followed by a 35 psia rupture disk outside secondary containment

10. Actuation – operator will cycle valves – no automatic on pressure

D.3 Coupled Containment

1. Vent line design temperature – 750° F

2. Vent line design pressure – 55 psia

3. Vent size – 24 in

4. Vent line – below yard piping (approximately 15 feet below grade)

5. Seismic – to withstand OBE

6. Redundant – no redundancy requirements (except isolation valves)

7. Isolation – three remote manual valves, one inside, one in annulus, and one outside secondary containment – followed by a rupture disk outside secondary containment

8. Actuation – operator will cycle valves – no automatic on pressure

D.4 Augmented Containment Cooling

1. Environmental design temperature - 342° F

2. Heat removal rate - 9×10^6 Btu/hr

3. Seismic - Category I(L)

4. Redundant - no redundancy requirements

5. Actuation - automatic on high containment pressure and temperature

D.5 Containment Inerting

1. Maximum O_2 for inerting - 4 percent

2. Minimum O_2 for deinerting - 16 percent

3. Nitrogen supply - present 24-inch purge

4. Nitrogen exhaust - present 24-inch purge

5. Seismic (inside containment) - Category I(L)

6. Redundant - no redundancy requirements

D.6 Halon Injection

1. Minimum halon for inerting - 3 percent

2. Halon supply - 3 ring headers in upper compartment, one ring header in lower compartment, at least one spray nozzle every 2,000 ft^3 of free volume

3. Minimum halon discharge pressure - 260 psig

4. Seismic (inside containment) - Category I(L)

5. Redundancy - no redundancy requirements

6. Containment isolation - standard

7. Actuation - containment isolation valves open automatically on high containment pressure followed by low reactor vessel level; manual isolation at halon tank

D.7 Ignition Sources

1. Hydrogen release rates - see Table D-3

2. Hydrogen release locations - small/large LOCA; pressurizer relief tank; reactor vessel head vent (assume it can be

TABLE D-3

S_{2D} - HYDROGEN RELEASE

TIME (SEC)	MASS ($1b_m$/SEC)	TEMPERATURE $^\circ F$
0.0	0.0	61.24
3480.	0.0	61.24
3804.	0.0413	66.56
4116.	0.2600	1582.29
4428.	0.7400	795.45
4752.	1.0700	771.47
5700.	0.4300	661.53
6330.	0.2233	555.39
6648.	0.1600	535.22
6960.	0.1167	519.43
8070.	0.0367	519.43
9252.	0.0043	519.43
9384.	0.0	61.40
9552.	0.0	61.24
9630.	0.1667	226.68
9636.	0.0	61.24
9651.	0.6667	61.24
9654.	39.0667	4251.30
9660.	100.20	1670.01
9660.1	0.0	550.42
9672.	0.0	61.24
22374.	0.0	61.24
24690.	0.0267	2676.15
30138.	0.0417	2678.20
31608.	0.0450	2681.12
35202.	0.0533	2684.05
38808.	0.0617	2681.12

42408.	0.0717	2675.27
46002.	0.0800	2675.27
49614.	0.0	61.24
6.0×10^{10}	0.0	61.24

routed or burned as desired)

3. Mixing - at least one air return fan is running

4. Seismic - Category I(L)

5. Redundant - no redundancy requirements

6. Actuation - actuate at low concentrations (approximately 4%) of H_2 and monitor controlled burning

DA02:APPDCY.01

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